PNNL-30086

Fresh Fuel Transportation of Accident Tolerant Fuel Concepts

FeCrAl Cladding

July 2020

Kenneth J Geelhood

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

Abstract

The U.S. Nuclear Regulatory Commission (NRC) is preparing for anticipated licensing applications and commercial use of accident tolerant fuel (ATF) in U.S. 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 (up to 5 wt% U-235) with iron-chromium-aluminum alloy (FeCrAl) cladding.

The NRC is specifically interested in FeCrAl alloys currently under consideration by U.S. fuel vendors. The only U.S. fuel vendor with FeCrAl cladding in their near term plans is Global Nuclear Fuels (GNF), which is developing a proprietary alloy, IronClad, that is designated alloy C26M. This report provides the current state of industry information on material properties and fuel performance considerations for FeCrAl cladding in fresh fuel transportation conditions. To support the agency's readiness efforts, this report will identify and discuss the implications of substitution of Zr-alloy cladding with FeCrAl cladding on the material properties of the cladding at the relevant conditions to fresh fuel transportation. This report will also discuss any characteristics of FeCrAl cladding that may not be addressed within existing regulatory documents.

This report will provide specific material properties for IronClad/C26M cladding as it is the only FeCrAl cladding alloy under near term development. Even though this particular cladding is the focus, the general conclusions are relevant to any FeCrAl cladding alloy and specific properties for another alloy would be necessary to license transportation of fresh fuel with that alloy.

Acronyms and Abbreviations

ATF	Accident Tolerant Fuel
BWR	boiling water reactor
CFR	Code of Federal Regulations
DOE	U.S. Department of Energy
FeCrAl	Iron-Chromium-Aluminum Alloy
GNF	Global Nuclear Fuels
k_{eff}	effective neutron multiplication factor
LWR	Light Water Reactor
MOX	Mixed Oxide Fuel (U, PuO_2)
NCT	Normal Conditions of Transport
NRC	U.S. Nuclear Regulatory Commission
ODS	oxide dispersion strengthened
ORNL	Oak Ridge National Laboratory
PNNL	Pacific Northwest National Laboratory
PWR	pressurized water reactor
RXA	recrystallized annealed
SRA	stress relief annealed
SRP	Standard Review Plan

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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 d-isilicide (U_3Si_2) pellets, and metallic fuels. These designs represent evolutions and deviations from the standard zirconium alloy clad, UO_2 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 FeCrAl cladding concepts in fresh fuel transportation conditions.

To support the agency's efforts, this report identifies and discusses substituting Zr-alloy cladding with FeCrAl cladding, and the effect of that substitution on the material properties of the cladding at the conditions relevant to the transportation of fresh fuel. This report also discusses any characteristics of FeCrAl cladding that may not be addressed within existing regulatory documents.

The scope of this report includes FeCrAl cladding alloys that are in development for ATF claddings. The NRC is specifically interested in Global Nuclear Fuel's (GNF) IronClad alloy C26M (Fe-12Cr-6Al-2Mo, see Table 1). This report provides an assessment of the impact of the substitution of typical Zr-alloy cladding with FeCrAl cladding on the requirements that have been placed on the transport of fresh nuclear fuel. This report will provide specific material properties for IronClad/C26M cladding as it is the only FeCrAl cladding alloy under near term development. Even though this is the focus of this report, the general conclusions are relevant to any FeCrAl cladding alloy and specific properties for another alloy would be necessary to license transportation of fresh fuel with that alloy.

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 FeCrAl cladding on fresh fuel transport. Section 3.0 describes criticality considerations for substitution of Zr-alloy cladding with FeCrAl cladding relative to Zr-alloy cladding. Section 4.0 discusses material property differences that should be considered for substitution of Zr-alloy cladding with FeCrAl cladding relative to Zr-alloy cladding. Overall conclusions are given in Section 5.0.

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™[1] for their PWR fuel, while retaining Zircaloy-2 for BWR fuel. Framatome uses M5®[2] for their PWR fuel, while also retaining Zircaloy-2 for BWR fuel. GNF only supplies BWR fuel and has recently received approval for their GNF-Ziron cladding.

FeCrAl alloys have historically been used in industrial applications where high temperature oxidation resistance is needed. As part of the ATF development, development of FeCrAl alloys has been performed by commercial entities, national laboratories, and universities with collaborations between the different research sectors. Both wrought FeCrAl and powder-metallurgy based FeCrAl alloys are currently under development. Within the nuclear industry, focus has been on the wrought FeCrAl alloys which are to be considered "nuclear grade." In this context, "nuclear grade" means an optimized composition to perform within the full range of reactor operating conditions. Much of the development of wrought FeCrAl alloy and the investigation of these have been performed at Oak Ridge National Laboratory (ORNL). ORNL has developed alloys and classifies them as two different generations of alloys. "Generation I" alloys are simple alloys, similar in composition and structure to early model alloys, while "Generation II" alloys are derived from Generation I alloys but include minor alloying additions to increase specific performance factors.

GNF has tested several different FeCrAl alloys including Kanthal APMT, C26M, and a FeCrAl oxide dispersion strengthened (ODS) alloy, MA956. GNF has not publicly stated which FeCrAl alloy will be used as their IronClad alloy, but two unfueled C26M alloy rods were inserted into Hatch 1 (Edwin Irby Hatch Nuclear Power Plant, Baxley, Georgia, operated by Southern Nuclear) and eight fueled C26M rods were inserted into Clinton (Clinton Power Station, Unit 1, Clinton, IL, operated by Exelon Generation Co., LLC). At the same time 16 unfueled C26M, APMT, and MA956 rods were inserted into Clinton. Based on this, it is likely that GNF will proceed with C26M as their IronClad alloy, but is still considering APMT and MA956. The composition of these three alloys are shown in Table 1.

[1] 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.
[2] M5® is a trademark or registered trademark of Framatome or its affiliates, in the USA or other countries.

Table 1. Composition of GNF FeCrAl alloys (compositions in wt%)

	Fe	Cr	Al	Mo	Ti	C	Si	Mn	Y	S	P
C26M[1]	Balance	12	6.0	2.0			0.2		0.05		
Kanthal APMT[2]	Balance	20.5-23.5	5.0	3.0		0.08 max	0.7 max	0.4 max		<0.01	<0.04
MA956[3]	Balance	18.5-21.5	3.75-5.75		0.2-0.6				0.3-0.7[(1)]		

[(1)] Values for Y_2O_3 in MA956

Figure 1 shows issues found in various FeCrAl alloys (Yamamoto, Field, Pint, Rebak, & Fawcett, 2020). Based on this figure, APMT and MA956 likely will exhibit α' embrittlement, while C26M does not exhibit any known issues, making it the most likely candidate for GNF IronClad.

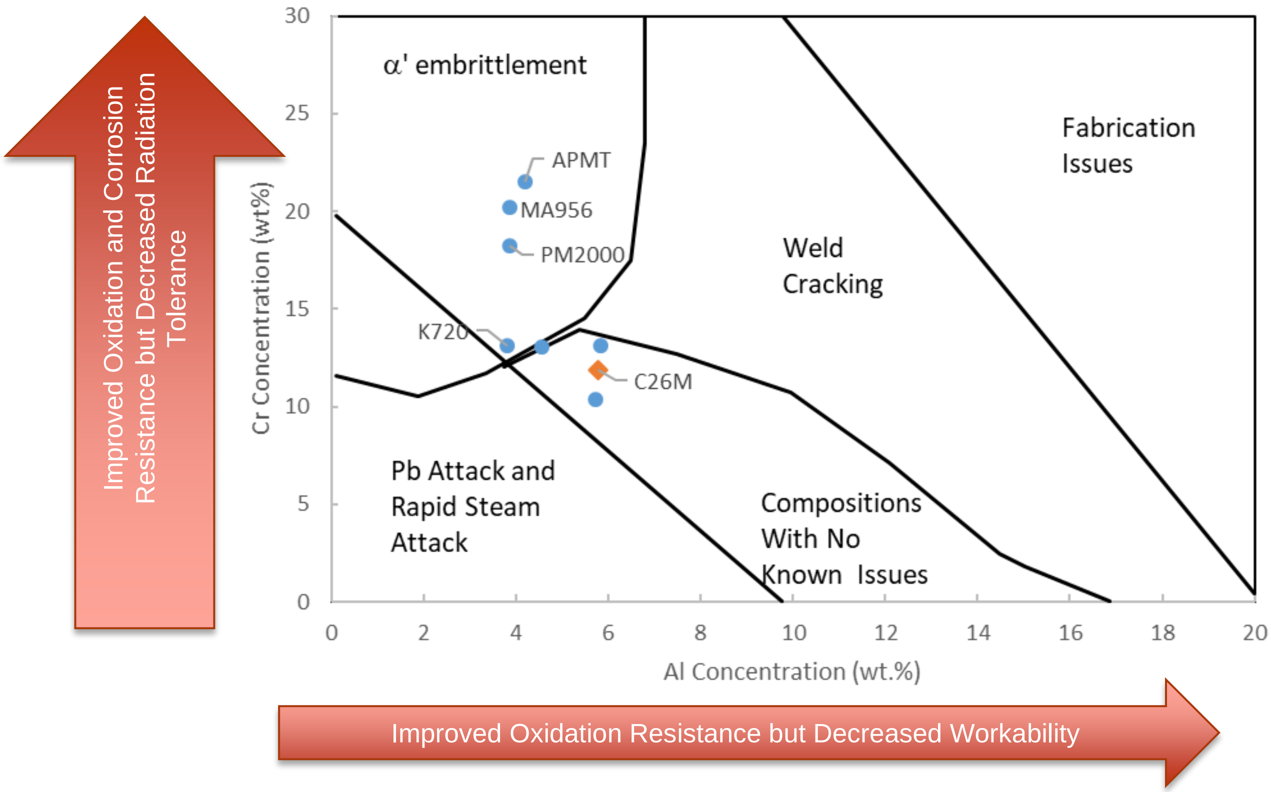

Figure 1. Impact of chromium and aluminum concentration in FeCrAl alloys (Yamamoto, Field, Pint, Rebak, & Fawcett, 2020)

[1] (Yamamoto, Kane, Pint, Trofimov, & Wang, 2019)
[2] https://www.kanthal.com/en/products/material-datasheets/tube/kanthal-apmt/
[3] http://www.pccforgedproducts.com/web/user_content/files/wyman/Incoloy%20alloy%20MA956.pdf

ATF cladding is being developed primarily to give an advantage during high temperature oxidation that may occur following a 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 subcritical 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 sufficient to guide the transportation of fresh fuel clad in FeCrAl. These regulations and guidance are discussed in the following section.

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 radioactive material: Type A and Type B.

A Type A package 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.

A Type B package 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. Hypothetical accident conditions are simulated by performance testing and engineering analysis.

Except for MOX fuel and UO_2 fuel fabricated from recycled or down-blended high-enriched uranium, 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 requirements for Type A and Type B packages, as well as the additional requirements for a Type AF packages. Fresh fuel could also be shipped in a Type BF container as the requirements are more restrictive for BF than AF.

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* 10^5 A_2*"), 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 (in part): *(f) 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.

71.55 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.

71.59 Standards for arrays of fissile material packages

(a) A fissile material package must be controlled by either the shipper or the carrier during transport to assure that an array of such packages remains subcritical. To enable this control, the designer of a fissile material package shall derive a number "N" based

on all the following conditions being satisfied, assuming packages are stacked together in any arrangement and with close full reflection on all sides of the stack by water:

(1) Five times "N" undamaged packages with nothing between the packages would be subcritical;

(2) Two times "N" damaged packages, if each package were subjected to the tests specified in § 71.73 ("Hypothetical accident conditions") would be subcritical with optimum interspersed hydrogenous moderation; and

(3) The value of "N" cannot be less than 0.5.

(b) The CSI must be determined by dividing the number 50 by the value of "N" derived using the procedures specified in paragraph (a) of this section. The value of the CSI may be zero provided that an unlimited number of packages are subcritical, such that the value of "N" is effectively equal to infinity under the procedures specified in paragraph (a) of this section. Any CSI greater than zero must be rounded up to the first decimal place.

(c) For a fissile material package which is assigned a CSI value--

(1) Less than or equal to 50, that package may be shipped by a carrier in a nonexclusive use conveyance, provided the sum of the CSIs is limited to less than or equal to 50.

(2) Less than or equal to 50, that package may be shipped by a carrier in an exclusive use conveyance, provided the sum of the CSIs is limited to less than or equal to 100.

(3) Greater than 50, that package must be shipped by a carrier in an exclusive use conveyance, provided the sum of the CSIs is limited to less than or equal to 100.

71.71 Normal Conditions of Transport

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

71.73 Hypothetical Accident Conditions

This section defines the conditions and tests used to represent normal conditions of transport. 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 AF Packages 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 IronClad cladding beyond what has been previously been done for Zr-alloy cladding.

2.0 Impact of FeCrAl Cladding on Fresh Fuel Transportation

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 FeCrAl 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 FeCrAl cladding.

In general, any analysis that relies on cladding properties will have to be updated for application to FeCrAl cladding. FeCrAl alloys are completely different from Zr-alloys and the thermophysical and mechanical properties are likewise completely different. These will be discussed in Section 4.0. In addition, the cladding thickness will likely be different from the cladding thickness used in a Zr-alloy fuel design. To compensate for the increased neutron cross section of FeCrAl relative to Zr-alloys, designers have opted to rely on the increased strength of FeCrAl and use a design with thinner cladding.

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 FeCrAl. This table shows that before performing analysis of normal conditions of transport using the cladding design information, the thermophysical or mechanical properties of the cladding should be re-evaluated using relevant rod design information and properties specific for the FeCrAl alloy in question. Additionally, a fatigue lifetime curve from representative cladding should be developed and used for the vibration analysis.

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

Requirement	Impact of changing from Zr-alloy cladding to FeCrAl cladding
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 will be impacted. Representative FeCrAl design information and material properties should be used.

Requirement	Impact of changing from Zr-alloy cladding to FeCrAl cladding
Heat: An ambient temperature of 38°C (100°F) in still air.	Fuel analysis will be impacted. Representative FeCrAl design information and material properties should be used.
Cold: An ambient temperature of -40°C (-40°F) in still air and shade.	Fuel analysis will be impacted. Representative FeCrAl design information and material properties should be used. Additional concerns remain regarding ductile to brittle transition temperature. Zr does not exhibit a ductile to brittle transition. However, FeCrAl alloys do. Tests should be performed to demonstrate adequate ductility over the full temperature range (-40°F to 100°F).
Reduced external pressure: An external pressure of 25 kPa (3.5 lbf/in^2) absolute.	No impact of this requirement on the fuel if it is inside a sealed cask. If not, analysis will be necessary to show the cladding can withstand the reduced external pressure.
Increased external pressure: An external pressure of 140 kPa (20 lbf/in^2) absolute.	No impact of this requirement on the fuel if it is inside a sealed cask. If not, analysis will be necessary to show the cladding can withstand the increased external pressure.
Vibration: Vibration normally incident to transport.	Fuel analysis will be impacted. Representative FeCrAl design information and material properties should be used. Fatigue lifetime curve for representative FeCrAl cladding tubes 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 fresh fuel packages can demonstrate no water in-leakage under water spray conditions.

Requirement	Impact of changing from Zr-alloy cladding to FeCrAl cladding
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 analysis will be impacted. Representative FeCrAl design information and material properties should be used. 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 FeCrAl cladding to make assessments regarding the acceptability of FeCrAl 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 analysis will be impacted. Representative FeCrAl design information and material properties should be used. 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 FeCrAl cladding to make assessments regarding the acceptability of FeCrAl cladding in a strain-based approach.
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 package 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, Fuel analysis will be impacted. Representative FeCrAl design information and material properties should be used.

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 FeCrAl cladding. This table shows that before performing analysis of normal conditions or hypothetical accident conditions using the cladding design information, the thermophysical or mechanical properties of the cladding should be re-evaluated using relevant rod design information and properties specific for the FeCrAl alloy in question. Also, the impact of the FeCrAl neutron absorption cross section should be included in the criticality assessment.

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

Requirement	Impact of changing from Zr-alloy cladding to FeCrAl cladding
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 analysis will be impacted. Representative FeCrAl design information and material properties should be used.

Requirement	Impact of changing from Zr-alloy cladding to FeCrAl cladding
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 analysis will be impacted. Representative FeCrAl design information and material properties should be used. 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 FeCrAl cladding to make assessments regarding the acceptability of FeCrAl 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 package requirement. If a package design is such that these loads cause deformation in the fuel, Fuel analysis will be impacted. Representative FeCrAl design information and material properties should be used.
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 package requirement. If a package design is such that these loads cause deformation in the fuel, Fuel analysis will be impacted. Representative FeCrAl design information and material properties should be used.
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 analysis will be impacted. Representative FeCrAl design information and material properties should be used. Applicants may wish to disposition this based on greater strength of FeCrAl at 800°C and superior corrosion resistance. However, if the cladding design is also changed, (e.g., thinner cladding), the existing analysis would not be applicable as the conversion between rod internal pressure and cladding hoop stress would not be the same.
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 will be impacted. The geometry of the cladding is likely different as well as the neutron absorption cross section of FeCrAl and the mechanical response of the cladding to various events.

Requirement	Impact of changing from Zr-alloy cladding to FeCrAl cladding
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 package requirement. If the package is not leek tight, analysis will be necessary to show the cladding can withstand the water pressure.

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. It is likely that the geometry and mechanical response of the FeCrAl cladding will not be the same as the Zr-alloy cladding. Additionally, the neutron cross section of FeCrAl will not be the same as the Zr-alloy cladding.

It is recommended that criticality assessment be performed specifically for fresh fuel transportation of FeCrAl. However, applicants may wish to reference existing criticality assessments for fuel with Zr-alloy cladding. If it can be demonstrated that the mechanical response for the specific events is the same or less severe (smaller rod deformation) and that the effective neutron absorption cross section of the cladding (including the effect of thinner cladding) is greater than or equal to that of the reference Zr-alloy case, then it would be reasonable to conclude that the k_{eff} of the fuel system will be bounded by current Zr-alloy reference case.

In the case an applicant is transporting an array of fresh fuel packages, the requirements of 10 CFR 71.59 also apply, and the applicant must perform array calculations under normal conditions of transport and hypothetical accident conditions in order to determine the package criticality safety index (CSI) for accumulation control on conveyances.

4.0 Material Properties for Fresh Fuel Transportation

ORNL has published a *Handbook on the Material Properties of FeCrAl Alloys for Nuclear Power Applications* (Field, Snead, Yamamoto, & Terrani, 2018). Unfortunately, the latest version of the handbook has limited properties for C26M other than creep properties, which are not relevant to transportation of fresh nuclear fuel. ORNL has also published a *Report on Exploration of New FeCrAl Heat Variants with Improved Properties* (Yamamoto, Kane, Pint, Trofimov, & Wang, 2019). This report presents recent data taken on C26M. Using the data from these references, this section will show comparisons between material properties for FeCrAl and Zr-alloy cladding. This information will be useful for an NRC reviewer to understand the magnitude of the difference in cladding properties between FeCrAl and Zr-alloy.

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

Figure 2 shows the thermal conductivity of Zircaloy taken from the FAST fuel performance code (Geelhood, et al., 2020) as well as the thermal conductivity of various FeCrAl alloys (Field, Snead, Yamamoto, & Terrani, 2018). It can be seen from this figure that Zircaloy and FeCrAl have similar thermal conductivity up to 1370K. This plot does not include alloy C26M, but recent thermal diffusivity data (Yamamoto, Kane, Pint, Trofimov, & Wang, 2019) indicates that C26M will have similar thermal conductivity to these other alloys.

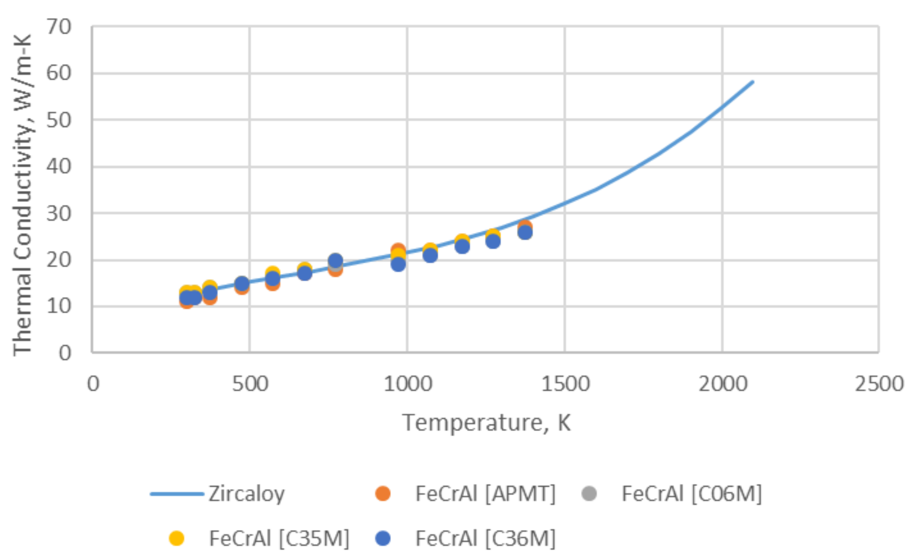

Figure 2. Thermal Conductivity of Zircaloy and various FeCrAl alloys

4.1.2 Thermal Expansion

Figure 3 shows the thermal expansion of Zircaloy taken from the FAST fuel performance code (Geelhood, et al., 2020) as well as the thermal expansion of various FeCrAl alloys (Field, Snead, Yamamoto, & Terrani, 2018). Zircaloy tubes are processed in such a way that the tubes exhibit a large degree of microstructural texture. This results in different thermal expansion in different directions as can be seen in this figure. Figure 3 also shows that Zircaloy has a lower thermal expansion than FeCrAl. This plot includes recent data from alloy C26M (Yamamoto, Kane, Pint, Trofimov, & Wang, 2019) that shows it to be in agreement with other FeCrAl alloys. The C26M exhibits some difference on heating and cooling, but the magnitude of this difference is not large.

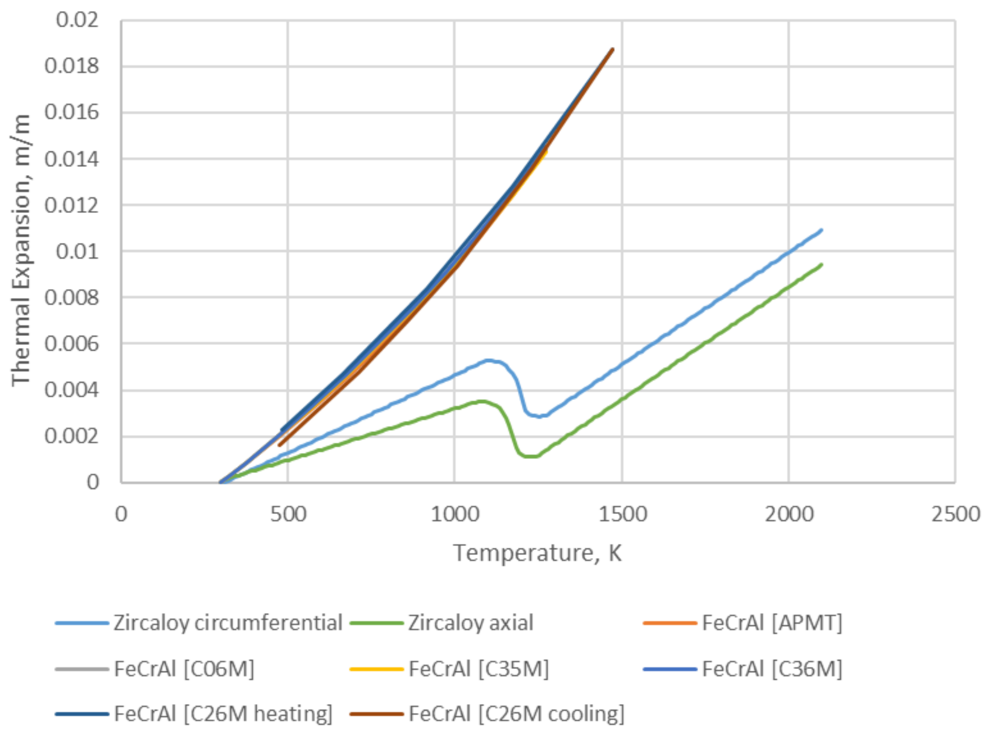

Figure 3. Thermal Expansion of Zircaloy and various FeCrAl alloys

4.2 Cladding Mechanical Properties

This section describes the mechanical properties of the cladding including elastic modulus, yield stress, and ductility.

4.2.1 Elastic Modulus

Figure 4 shows the elastic modulus of unirradiated Zircaloy taken from the FAST fuel performance code (Geelhood, et al., 2020) as well as the elastic modulus of various FeCrAl alloys (Field, Snead, Yamamoto, & Terrani, 2018). It can be seen from this figure that Zircaloy has a considerably lower elastic modulus than FeCrAl. This plot does not include alloy C26M, but given scatter shown here between various FeCrAl alloys, it is not expected that C26M will be significantly different.

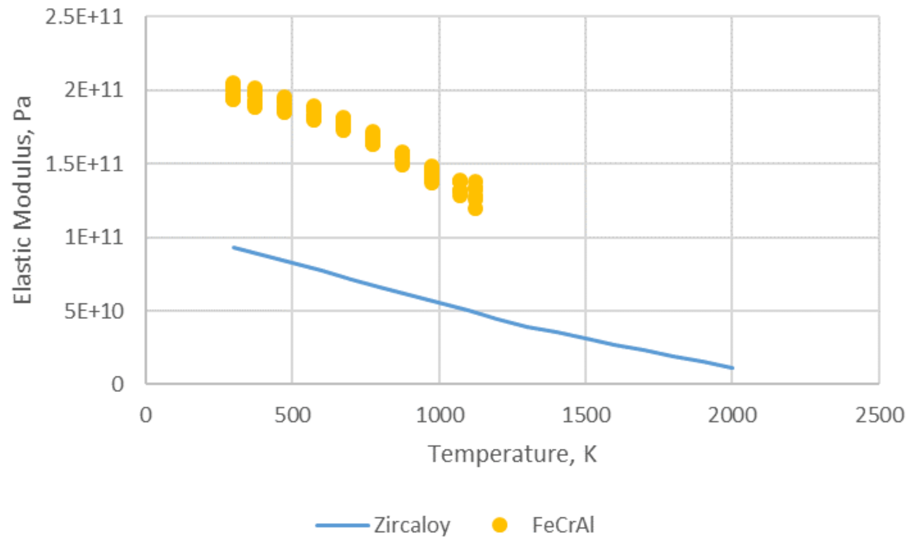

Figure 4. Unirradiated elastic modulus of Zircaloy and various FeCrAl alloys

4.2.2 Yield Stress

Figure 5 shows the yield stress of unirradiated Zircaloy taken from the FAST fuel performance code (Geelhood, et al., 2020) as well as the yield stress of various FeCrAl alloys (Field, Snead, Yamamoto, & Terrani, 2018). Zircaloy tubes are typically provided either in a stress relief annealed (SRA) condition or in a fully recrystallized (RXA) condition. The expected unirradiated yield stress in each of these conditions is shown. It can be seen from this figure that there is considerable variation in FeCrAl yield stress depending on the alloy. This plot includes room temperature yield stress for alloy C26M (Yamamoto, Kane, Pint, Trofimov, & Wang, 2019). Given the scatter in FeCrAl yield stress, temperature dependent yield stress data is necessary to perform mechanical calculations to support fresh fuel transport.

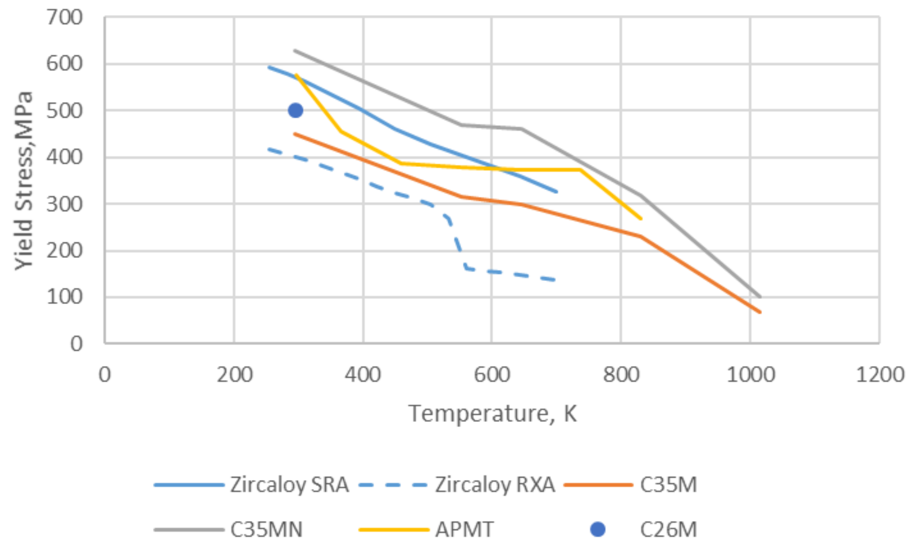

Figure 5. Unirradiated yield stress of Zircaloy and various FeCrAl alloys

4.2.3 Ductility

Unirradiated Zircaloy shows adequate ductility over the entire temperature range of interest for fresh fuel transportation (-40°C to 100°C). However, FeCrAl exhibits a ductile to brittle transition temperature below which the alloy exhibits brittle failure and almost no ductility. This temperature can range from 0°C to 150°C based on aluminum content between 3 wt% and 6 wt% (Field, Snead, Yamamoto, & Terrani, 2018). See Figure 6. For other FeCrAl alloys, the ductile to brittle transition was between 119 and 318°C which resulted in those FeCrAl specimens showing fully brittle characteristics at room temperature (Field, Snead, Yamamoto, & Terrani, 2018).

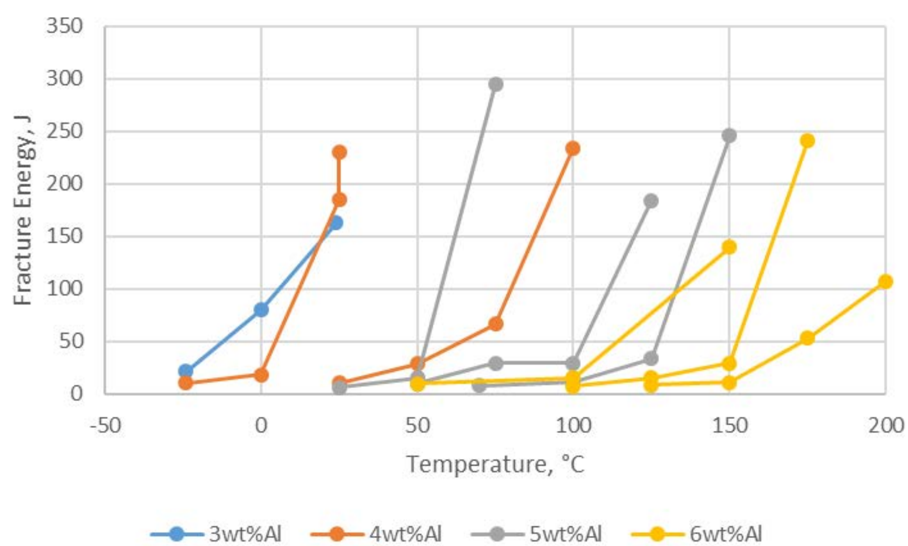

Figure 6. Charpy impact toughness for FeCrAl alloys with various Al content

However, recent data on C26M shows adequate ductility down to -40°C. (Yamamoto, Field, Pint, Rebak, & Fawcett, 2020) If this data can be confirmed by the applicant, then the specific C26M alloy would be acceptable for the temperature range of interest. These data are shown in Table 4. It can be seen that there is some variation between the test sample geometry, but in general the ductility is about the same at room temperature and -40°C.

Table 4. C26M ductility data (Yamamoto, Field, Pint, Rebak, & Fawcett, 2020)

Sample	Uniform Elongation%	Total Elongation, %
Ring tensile test at 20°C	1.9	15.1
Axial tube tensile test at 20°C	7.7	15.8
Ring tensile test at -40°C	1.8	12.3
Axial tube tensile test at -40°C	5.2	10.5

Under the traditional application involving a stress-based approach, it may not be necessary to ensure cladding ductility if it is demonstrated that the cladding stress never exceeds the yield stress. 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 and a FeCrAl alloy that exhibits no ductility would likely not be acceptable. For a strain-based approach, the special tests should be performed on the specific FeCrAl cladding to make assessments regarding the acceptability of this alloy.

4.3 Cladding Fatigue

Cladding fatigue is necessary to evaluate the impact of vibration during NCT on FeCrAl 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 for Zircaloy are typically found relative to the O’Donnell and Langer unirradiated Zircaloy fatigue design curve (O'Donnell & Langer, 1964). Figure 7 shows the typical unirradiated Zircaloy fatigue design curve as well as some fatigue data from a particular FeCrAl alloy (Field, Snead, Yamamoto, & Terrani, 2018). It can be seen from these data that the fatigue lifetime for this FeCrAl alloy is considerably different than the Zircaloy fatigue lifetime. These data indicate a significant temperature dependence. No fatigue data from C26M are available. Temperature dependent fatigue data from this alloy or the specific alloy being considered are necessary to perform vibration calculations to support fresh fuel transport. 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.

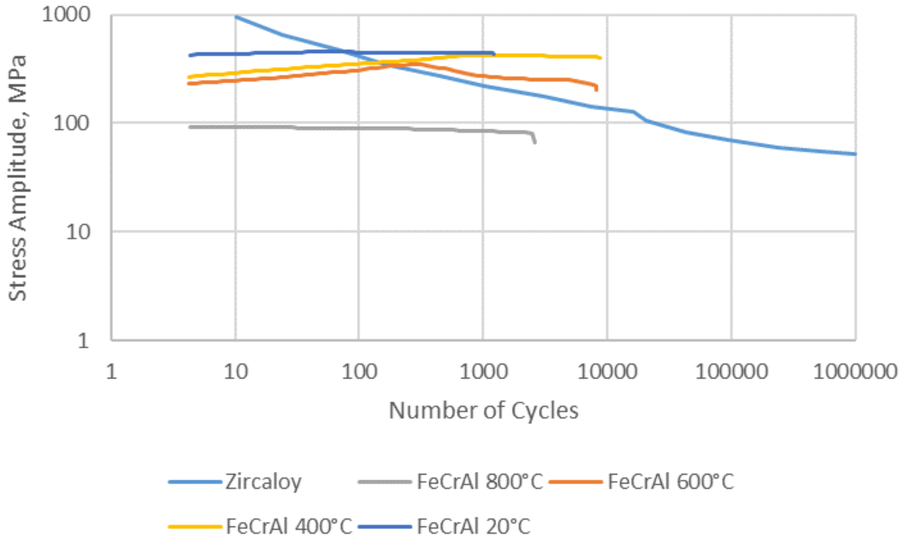

Figure 7. Fatigue lifetime curve for unirradiated Zircaloy and fatigue data from FeCrAl (Fe-23.85Cr-3.89Al)

4.4 Cladding Nuclear Properties

The nuclear properties of the cladding (neutron absorption cross section) are important in the determination of the k_{eff} of the fuel assemblies for transportation of fresh fuel. Table 5 shows the neutron absorption cross section for the primary elements in Zr-alloy cladding and FeCrAl. Zr-alloy cladding also contains other minor elements such as iron (Fe), tin (Sn), and niobium (Nb), but these do not likely impact the overall cross section. Likewise, Generation 2 FeCrAl alloys contain other minor elements such as molybdenum (Mo) and titanium (Ti), but these do not likely impact the overall cross section. FeCrAl has higher overall thermal neutron absorption than Zr, but that the cross sections of FeCrAl components are well known, and modern computer codes can easily account for the associated change in k-eff due to cladding cross section differences.

Table 5. Neutron thermal absorption cross section of various cladding elements

Element	Neutron Cross Section, Barns
Zr	0.184
Fe	2.56
Cr	3.1
Al	0.233

5.0 Conclusions

This report provides an assessment of the shipment of fresh UO_2 fuel with FeCrAl cladding. The U.S. NRC is specifically interested in GNF's IronClad alloy with is alloy C26M. This assessment concludes:

- Fresh UO_2 fuel with FeCrAl cladding may be shipped in a Type A fissile package because the FeCrAl cladding 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 FeCrAl 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 thermal conductivity, cladding thermal expansion, cladding yield stress, and cladding elastic modulus for a stress-based performance analysis[2]. If any analysis requires that cladding be intact beyond cladding yield stress, cladding ductility will also be required. Expected FeCrAl properties including, where available, C26M properties, are shown and compared to Zr-alloy cladding properties.
  - These comparisons are useful to assist NRC in evaluating claims by applicants regarding FeCrAl properties and will also highlight were C26M specific data are necessary.
  - C26M specific data are lacking for yield stress as a function of temperature, ductility, and fatigue. Data is sparse for thermal conductivity and elastic modulus, but is not expected to be significantly different from other FeCrAl alloy data that are available.
- It is recommended that criticality assessment be performed specifically for fresh fuel transportation of FeCrAl.

[1]A Type A Fissile package is not acceptable for transport of fresh MOX fuel and recycled or down-blended UO_2. A Type BF package is required for these.

[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 FeCrAl cladding to make assessments regarding the acceptability of FeCrAl cladding in a strain-based approach.

6.0 References

Field, K., Snead, M., Yamamoto, Y., & Terrani, K. (2018). *Handbook on the Material Properties of FeCrAl Alloys for Nuclear Power Production Applications (FY18 Version: Revision 1.1), ORNL.SPR-2018/905 Rev. 1.* Oak Ridge, TN: Oak Ridge National Laboratory.

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

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

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.

Yamamoto, Y., Field, K., Pint, B., Rebak, R., & Fawcett, R. (2020). Development of Nuclear Grade Wrought FeCrAl Alloys for Accident Tolerant Fuel Cladding. *TMS 2020 149th Annual Meeting and Exhibition.* San Diego, CA: TMS.

Yamamoto, Y., Kane, K., Pint, B., Trofimov, A., & Wang, H. (2019). *Report on Exploration of New FeCrAl Heat Variants with Improved Properties, ORNL/TM-2019/1290.* Oak Ridge, TN: Oak Ridge National Laboratory.

Pacific Northwest National Laboratory

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

www.pnnl.gov