

Spent Fuel Storage and Transportation of Accident Tolerant Fuel Concepts

Cr-Coated Zirconium Alloy and FeCrAl
Cladding

September 2020

Joseph M Brusky
Kenneth J Geelhood
Christine E Goodson
Walter G Luscher



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Pacific Northwest National Laboratory
Richland, Washington 99354

Abstract

The U.S. Nuclear Regulatory Commission (NRC) is anticipating licensing applications and use of accident tolerant fuel (ATF) in U.S. commercial nuclear power reactors. Pacific Northwest National Laboratory is providing technical assistance to the NRC related to the newly proposed nuclear fuel and cladding designs.

This report focuses specifically on near-term ATF cladding concepts being investigated to replace zirconium-based alloys for fuel cladding currently in use and provides current state-of-the-industry information on material properties and fuel performance considerations under storage and transportation conditions.

Currently, three U.S. nuclear fuel market suppliers are developing near-term ATF designs: GNF has tested several different iron-chromium-aluminum (FeCrAl) alloys, including Kanthal APMT, C26M, and MA956 as well as Abrasion Resistant, More Oxidation Resistant (ARMOR) cladding, a coated zirconium alloy cladding with UO_2 fuel; Westinghouse has tested Cr-coated ZIRLO[®] cladding and standard UO_2 and chromium oxide-aluminum oxide ($\text{Cr}_2\text{O}_3+\text{Al}_2\text{O}_3$)-doped UO_2 fuels; and Framatome has tested Cr-coated M5[®] cladding and Cr_2O_3 -doped UO_2 fuel.

To support the NRC's readiness efforts, this report will identify and discuss degradation and failure modes of near-term ATF cladding concepts, including fuel performance characteristics that may not be addressed within existing regulatory documents.

Acronyms and Abbreviations

AOO	anticipated operational occurrence
ARMOR	Abrasion Resistant More Oxidation Resistant
ATF	accident tolerant fuel
ATR	Advanced Test Reactor
BDBA	beyond design basis accident
BWR	boiling water reactor
CEA	French Alternative Energies and Atomic Energy Commission
CFR	Code of Federal Regulations
CRUD	Chalk River Unknown Deposit (generic term for deposits on fuel cladding)
DBA	design basis accident
DNB	departure from nucleate boiling
dpa	displacements per atom
DOE	U.S. Department of Energy
DSS	dry storage system
EPRI	Electric Power Research Institute
GDC	General Design Criteria
GNF	Global Nuclear Fuels
INL	Idaho National Laboratory
ISG	Interim Staff Guidance
KAERI	Korea Atomic Energy Research Institute
KEPCO	Korea Electric Power Corporation
LOCA	loss-of-coolant accident
LTA	lead test assembly
LWR	light water reactor
MBE	molecular beam epitaxy
MIT	Massachusetts Institute of Technology
MITR	Massachusetts Institute of Technology Reactor
NCS	normal conditions of storage
NCT	normal conditions of transport
NEA	Nuclear Energy Agency
NRC	U.S. Nuclear Regulatory Commission
ODS	oxide dispersion strengthened
OECD	Organization for Economic Cooperation and Development
OECD-NEA	Organization for Economic Cooperation and Development- Nuclear Energy Agency
ORNL	Oak Ridge National Laboratory

PCT	peak cladding temperature
PIE	post irradiation examination
PNNL	Pacific Northwest National Laboratory
PVD	physical vapor deposition
PWR	pressurized water reactor
R&D	research and development
RIA	reactivity-initiated accident
SNF	spent nuclear fuel
SRP	standard review plan
TIG	tungsten inert gas
TRISO	TRi-structural ISOtropic particle fuel
Zry	Zircaloy
Zry-2	Zircaloy-2
Zry-4	Zircaloy-4

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1.0 Introduction

The accident at the Fukushima Daiichi power plant led to a worldwide interest in the development of nuclear fuel systems with enhanced accident tolerance that, in turn, led to the start of accident tolerant fuel (ATF) programs among industry teams and across many research institutions. A new fuel system alone is insufficient to completely mitigate accident consequences; however, new fuel in combination with other systems may provide some margin in responding to such rare events while providing additional benefits during more frequent off-normal events or during normal operations.

The U.S. Nuclear Regulatory Commission (NRC) is expecting license applications for and commercial use of ATF. ATF is being developed to “improve safety in the event of accidents in the reactor or spent fuel pools” (U.S. Congress 2011) while maintaining or exceeding normal reactor operational expectations compared to current fuel technologies. The current ATF designs under development fall into one of two categories: cladding and fuel. Cladding developments include coated Zircaloy cladding, silicon-carbide (SiC) cladding, and iron-chromium-aluminum (FeCrAl) cladding; fuel developments include doped UO_2 , high-density fuels (e.g., U_3Si_2), and metallic fuels.

As most of the NRC’s regulatory framework was developed for the zirconium alloy-clad, UO_2 -fueled system, Pacific Northwest National Laboratory (PNNL) is providing technical assistance related to the new proposed fuel and cladding designs to enhance the staff’s knowledge base and ultimately support the NRC’s efforts to develop and review the required regulatory infrastructure for commercial use of ATF.

This report provides current state of the industry information on material properties and fuel performance considerations for Cr-coated and FeCrAl cladding concepts in spent fuel storage and transportation conditions. To support the agency’s efforts, this report will identify and discuss degradation and failure modes of Cr-coated and FeCrAl cladding spent fuel concepts, including fuel performance characteristics that may not be addressed within existing regulatory documents (e.g., regulations, regulatory guidance, NUREG-2215).

The scope of this report includes metallic and ceramic coatings of chromium that are in development for ATF cladding as well as the most likely near-term candidates from the group of FeCrAl concepts. This entire class of concepts is generically referred to as “near-term ATF concepts” throughout this report. The near-term Cr-coated cladding concepts being tested, and thus closest to license request, consist of Cr-coating currently licensed Zr alloy claddings and are referred to generically as “Cr-coated Zr cladding” in this report. The near-term FeCrAl concept that is the focus of most research, development, and testing is C26M, however, unless otherwise specified, the family of FeCrAl cladding alloys is generically described as “FeCrAl cladding”.

This report provides an overview of the ATF cladding development. A general overview of the Cr-coated Zr clad and FeCrAl clad is provided. An assessment is made of the applicability of current regulations and guidance for spent fuel storage and transportation to these near-term accident tolerant fuel concepts. Based on this, an assessment of the critical data needs for spent fuel storage and transportation for each ATF concept is made. A recent literature review has been performed and is documented regarding the current status of the required data.

1.1 Background

Cladding for light water reactors (LWRs) has historically been fabricated from zirconium alloys; Zircaloy-2 has been used for boiling water reactors (BWRs) and Zircaloy-4 has been used for pressurized water reactors (PWRs). In-reactor cladding corrosion became an issue as demand for higher burnup levels of LWR fuels grew. To reduce the issue and maintain (or improve) the creep properties of the cladding, nuclear fuel vendors developed proprietary Zr-based alloy claddings that have mostly replaced the traditional Zr-based alloys. Currently, Westinghouse uses ZIRLO^{®1} and Optimized ZIRLO[™] for PWR fuel and Zircaloy-2 for BWR fuel; Framatome uses M5^{®2} for PWR fuel and Zircaloy-2 for BWR fuel; Global Nuclear Fuels (GNF), only supplying BWR fuel, recently received approval for GNF-Ziron. LWR cladding is typically between 0.56 and 0.75 mm thick.

ATF cladding is being developed primarily to give an advantage during high-temperature steam oxidation that can occur following a design basis accident (DBA) or in a situation considered to be beyond the fuel design basis; however, there is a general set of requirements placed on nuclear fuel cladding to retain shape, pellets, and fission products and effectively transfer heat to the coolant (10 CFR Part 50, Appendix A). Prior NRC and PNNL experience in review and approval of advanced Zr-based alloy cladding and with steel tubes in nuclear applications will be used in the development of material for review and approval of ATF cladding.

The following subsections provide an overview of the normal conditions that fuel assemblies are subjected to during storage and transportation conditions and the limits associated with each condition.

1.1.1 Wet Storage Conditions

Immediately after discharge from the reactor, fuel assemblies are placed in the spent fuel storage pool. Active cooling is provided to this pool to remove the excess decay heat from the spent fuel assemblies. This pool is not pressurized, so the temperature is limited to less than 100°C to prevent the water from boiling. However operating limits are around 50-60°C and normal temperatures are 30-35°C.

In general, the condition of the rods has not been observed during their time in the spent fuel pool.

1.1.2 Normal Dry Storage Conditions

After spent fuel assemblies have spent some period in the spent fuel pool (typically more than 5 years), they can be moved out of the reactor containment building and into dry storage. A dry storage system (DSS) is a term inclusive of designs for both storage casks and storage containers. Fuel assemblies are typically subjected to at least one drying operation when they are removed from the spent fuel pool to remove any water from the assemblies. Drying operations typically involve drawing a rough vacuum on the fuel assemblies, which leads the fuel rods to heat up due to the lack of convection heating and active heating from radioactive decay. Per NRC review guidance in NUREG-2215 and NUREG-2216, the length of these drying operations is controlled such that the cladding will not exceed 400°C at any axial location of the fuel rod. Additionally, per the same NRC review guidance, the cladding should not experience

¹ ZIRLO[®] and Optimized ZIRLO[™] are registered trademarks of Westinghouse Electric Company LLC.

² M5[®] is a registered trademark of Framatome.

more than 10 thermal cycles each not exceeding a temperature variance of 65°C in order to minimize the impacts of hydride reorientation in the cladding (See Section 5.4.2 of NUREG-2215) (US Nuclear Regulatory Commission 2020a; US Nuclear Regulatory Commission 2020b).

Following the vacuum drying, fuel assemblies are placed in DSS that are passively cooled with helium (may be oriented vertically or horizontally). The design of each DSS accommodates a limited amount of heat removal and the number and type of fuel assemblies placed in each DSS are limited such that this heat limit is not exceeded by the total heat produced by radioactive decay in each fuel assembly. The passive cooling in each of these DSS is designed such that the maximum cladding temperature will not exceed 400°C at the initial loading if the design basis is consistent with NUREG-2215 and NUREG-2216. As time progresses, the fuel temperature will decrease as the decay heat produced in each assembly decreases (US Nuclear Regulatory Commission 2020a; US Nuclear Regulatory Commission 2020b).

Analyses are performed to certify that during any identified accident condition, the temperature will not exceed 570°C (See Section 5.4.2 of NUREG-2215).

1.1.3 Transportation Conditions

Many DSS currently in use are dual-use storage and transportation containers. Other utilities plan to move spent fuel from DSS to shipping containers prior to transport to final disposition. Regardless, the boundary conditions for the fuel assemblies are the same. They will be in helium (or another inert gas) under passive cooling with a maximum normal temperature of 400°C and a maximum temperature under accident conditions of 570°C (See Section 7.4.14.2 of NUREG-2216). There are additional mechanical requirements placed on the transportation package and the spent fuel contents regarding transportation loads such as shock and vibration as well as different accident scenarios (US Nuclear Regulatory Commission 2020b).

1.2 Previous Reviews

Three publications have been identified as providing a reasonable overview of the work that has been done to support the development of ATF:

1. The Organization for Economic Cooperation and Development – Nuclear Energy Agency (OECD-NEA) has published a state-of-the-art report on LWR ATF (OECD-NEA 2018).
2. Oak Ridge National Laboratory (ORNL) published a review paper in *Journal of Nuclear Materials* summarizing the status and challenges associated with ATF (Terrani 2018).
3. The Electric Power Research Institute (EPRI) has published a gap analysis on coated cladding being developed for accident tolerant fuels (EPRI 2018).
4. The Electric Power Research Institute (EPRI) has published a report evaluating the performance of ATF under beyond design basis accident (BDBA), DBA, and anticipated operational occurrence (AOO) scenarios, with specific reference to the U.S. fleet and regulations (EPRI 2019).

OECD-NEA Report

The OECD-NEA state-of-the-art report (OECD-NEA 2018) discusses the work being done on all ATF concepts, including some development and data collection activities that have been performed. Chapter 10 of that report describes the coated cladding concepts and chapter 11 of that report describes FeCrAl.

OECD-NEA Report: Coated Cladding

The report (OECD-NEA 2018) summarizes the main advantages of and the challenges to be monitored for coated cladding as:

Main advantages:

- Low neutronic penalty if coating is sufficiently thin ($<20\text{ }\mu\text{m}$)
- Similar mechanical behavior as uncoated cladding if coating is sufficiently thin ($<20\text{ }\mu\text{m}$)
- Significant reduction in corrosion kinetics for metallic coatings (Cr, Cr-Al, FeCrAl) and for some ceramic coatings (CrN and TiN) → increased margins and longer exposure times expected
- Significantly reduced hydrogen pickup and therefore hydrogen embrittlement for these same coatings → increased margins and longer exposure times expected
- Increased wear resistance → reduced fuel rod failures due to fretting are expected (but needs further assessment in representative irradiation conditions up to high burn-up).

Challenges to be monitored:

- Coating thickness
- Dissolution of Al-containing coatings (TiAlN, CrAlN, and to a significantly lower extent FeCrAl)
- Irradiation impact on coatings, which may lead to cracks or local removal of the coating
- Lack of out-of-pile data on the mechanical behavior of ceramic coatings
- Lack of in-pile mechanical behavior data in representative LWR conditions, especially at high burn-up
- Lack of out-of-pile corrosion behavior of MAX phase coatings in normal operating conditions.

OECD-NEA Report: FeCrAl Cladding

The report (OECD-NEA 2018) summarizes the main advantages of and the challenges to be monitored for FeCrAl cladding as:

Main advantages:

- Superior resistance to fragmentation upon reflooding in a DBA
- Increased wear resistance
- Increased reactor coping time in accident conditions
- Enhanced ability to maintain a coolable geometry in accident conditions
- Improved coolant oxidation reaction kinetics in accident conditions → significant reduction in heat generation and hydrogen generation during accident conditions
- Increased allowable peak cladding temperature (PCT) during normal operations and AOOs and in accident conditions

- Similar or better ballooning and perforation characteristics than zircaloy in accident conditions → improvement in fission product retention.

Challenges to be monitored:

- Increased parasitic neutron absorption relative to zirconium alloys
- Increased fuel pellet diameter with a reduction in cladding thickness to ~300 μm at a constant fuel enrichment of 4.9 w/o can maintain current cycle length → increased fuel cycle costs
 - No increased costs related to handling, storage, and cooling are anticipated (Rebak, Terrani, and Fawcett 2016)
- Increased permeability of hydrogen through the cladding → increased release of tritium into the reactor coolant during normal operations and AOOs
- Lack of some irradiated material properties and integral tests.

PNNL staff generally agree with these conclusions, however this report will produce its own conclusions regarding lack of data and challenges.

Review Article in *Journal of Nuclear Materials*

The article in *Journal of Nuclear Materials* (Terrani 2018) discusses the work being done on all ATF concepts, including the development status of and challenges facing the use of both coated claddings and FeCrAl cladding.

This article reviews coatings of Cr, CrN, CrAlN, TiAlN, TiN/TiAlN, Ti_2AlC , Ti_3SiC_2 , and CrAlC. In general, it was concluded that in terms of corrosion resistance and neutron stability, the Cr and CrN are the most promising. In the case of Cr-coating and CrN-coatings, it concludes that both coatings are resistant to corrosion in LWR coolant and stable under neutron irradiation at expected temperatures. It concludes that Cr-coatings provide increased resistance to high temperature steam oxidation while CrN does not.

Terrani (2018) reviewed systematic studies that have been performed on FeCrAl cladding to determine the critical quantities of Cr and Al in the alloy system to avoid embrittlement as a result of the α' -phase precipitation that occurs after irradiation at 300°C to 400°C and to increase resistance to high-temperature steam oxidation. Normal operation and AOO behavior of FeCrAl cladding is expected to be superior to that of Zr-based cladding. However, FeCrAl cladding has a poor thermal neutron utilization factor and a potential for increased tritium release.

These conclusions help in determining if a concept should be evaluated for ATF research and do not consider the requirements for licensing of such fuel.

EPRI Cr-Coated Cladding

The EPRI gap analysis report (EPRI 2018) attempts to identify gaps related to the licensing of Cr-coated cladding. To do this, the report identifies gaps in three general areas; 1) fuel performance phenomena and modeling gaps, 2) material and behavior model gaps, and 3) technical licensing/regulatory gap analysis. The following gaps were identified in each area:

Fuel performance phenomena and modeling gaps

- Simulation meshing capabilities
- Material interfaces
- Material model implementation
- Validation of the computer code
- Problem initialization.

Material and behavior model gaps

- Material properties (thermal)
- Material properties (mechanical)
- Diffusion of Cr coating into Zr substrate
- Cracking and/or delamination of coating.

Technical licensing/regulatory gaps

- Damage at the substrate/coating interface related to microcracking, localized embrittlement, and system effects
- Fretting damage to grid components from hard coatings on cladding
- CRUD deposition affecting heat transfer during AOOs and DBAs
- Coating spallation leading to coolability issues with pump screen clogging.

These gaps, as well as additional gaps that have been identified, will be discussed throughout this report.

EPRI ATF Report

The EPRI report (EPRI 2019) evaluated the performance of ATF under BDBA, DBA, and AOO scenarios, with specific reference to the U.S. fleet and regulations. The report presented the following potential safety benefits of FeCrAl cladding:

- Reduced fuel fragmentation and dispersal, which reduces gap and in-vessel releases;
- Improved fuel reliability;
- Reduced oxidation;
- Reduced corrosion and hydrogen pickup;
- Additional coping times;
- PCT and departure from nucleate boiling (DNB) benefits, which enable improved thermal margins and increased burnups;
- Tolerance to CRUD-induced localized corrosion;
- Improved fission product barrier in accident (DBA and BDBA) conditions, which reduces equipment qualification demands;
- Potential replacement of DNB limits with dryout; and

- Improved DBA margins, which enables thermal limit relaxation by relaxing emergency core cooling system injection.

2.0 Overview of ATF Cladding Concept Development

This section provides an overview of concepts that are currently being developed for ATF, with focus on those concepts that are near-term ATF candidates. Sections 2.1 to 2.3 summarize concepts being developed by fuel manufacturers for the U.S. market; Sections 2.4 to 2.6 summarize concepts being developed in other countries; Sections 2.7 to 2.8 summarize concepts being developed and tested at laboratories. Although the concepts being developed outside the U.S. may not have a short-term path to U.S. licensing, the research and development (R&D) may identify relevant degradation mechanisms or data that can be applied to U.S.-license-capable concepts.

2.1 Westinghouse

Westinghouse is working toward commercializing two near term ATF designs (Karoutas 2019). In the near term, Westinghouse is testing chromium-coated zirconium cladding with chromium oxide-aluminum oxide ($\text{Cr}_2\text{O}_3+\text{Al}_2\text{O}_3$)-doped UO_2 (ADOPT[™]) fuel and chromium-coated zirconium cladding with uranium silicide (U_3Si_2) fuel. In the long term, Westinghouse is developing SiC fiber reinforced SiC matrix (SiC/SiC) composite cladding with U_3Si_2 fuel and other pellet designs including uranium nitride (UN). All concepts are referred to as EnCore^{®1} fuel (Westinghouse Nuclear 2019; Lahoda and Boylan 2019).

The cladding concepts have been tested in the Massachusetts Institute of Technology Reactor (MITR); the U_3Si_2 fuel pellets have been tested in the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL) and at Westinghouse.

Lead test rods (12 rods with Cr-coated ZIRLO[®] cladding and standard UO_2 fuel, 4 rods with standard cladding and segmented U_3Si_2 fuel, and 4 rods with Cr-coated ZIRLO[®] cladding and ADOPT^{™2} fuel) were inserted at Byron 2 in April 2019 (Westinghouse Nuclear 2019; Avali and Lahoda 2018) with a plan for licensing for region quantities as early as 2023 (Karoutas 2019). Further lead test assemblies (LTAs) containing Cr-coated cladding with ADOPT[™], UO_2 , and U_3Si_2 fuels and SiC cladding with U_3Si_2 fuel are planned for irradiation in 2022 (Avali and Lahoda 2018).

Westinghouse is not developing any FeCrAl cladding concepts.

2.2 Framatome

As part of the CEA-Framatome-EDF French nuclear fuel joint program, Framatome is working toward commercializing two ATF designs (Reed and Boman 2019). In the near term: Cr-coated zirconium alloy cladding (M5[®]) with Cr_2O_3 -doped UO_2 fuel; in the long term: SiC/SiC composite cladding with Cr_2O_3 -doped UO_2 fuel.

Cr-coated M5[®] cladding has been tested at Gösgen and ORNL; Cr-coated M5[®] cladding with Cr_2O_3 -doped UO_2 fuel has been tested at ATR and Halden. SiC test rodlets are planned for insertion into ATF-2 at ATR in 2020 (Reed and Boman 2019).

¹ EnCore[®] is a registered trademark of Westinghouse Electric Company LLC.

² ADOPT[™] is a registered trademark of Westinghouse Electric Company LLC.

Sixteen lead test rods with Cr-coated M5[®] cladding and Cr₂O₃-doped UO₂ fuel were inserted at Vogtle Unit 2 (Framatome 2019) in spring 2019. Thirty-two Cr-coated lead test rods were inserted in fall 2019 at ANO Unit 1. Two full assemblies with Cr-coated M5[®] cladding and Cr₂O₃-doped UO₂ fuel are planned for insertion at Calvert Cliffs in spring 2021 (Reed and Boman 2019). Framatome is on track for a 2025 batch reload (Reed 2019). Lead test rods with SiC/SiC cladding and Cr₂O₃-doped UO₂ fuel are planned by 2022 (Framatome 2020).

Framatome is not developing any FeCrAl cladding concepts.

2.3 Global Nuclear Fuels

Together with General Electric, Global Nuclear Fuels (GNF) is working toward commercializing two near-term ATF designs: Abrasion Resistant, More Oxidation Resistant (ARMOR) cladding, a coated zirconium alloy cladding with UO₂ fuel, and FeCrAl cladding called IronClad (Fawcett 2019).

Both ARMOR and IronClad have been tested at ATR. LTAs containing unfueled, IronClad-segmented rods and ARMOR-segmented rods were irradiated at Plant Hatch, discharged in February 2020 (though one rod will go through two additional cycles). Post-irradiation examination (PIE) results are expected by January 2021. LTAs with both ARMOR and three varieties of IronClad clad rods have been installed at Clinton (GNF 2020).

2.4 Japan

The Japanese ATF R&D program is developing two ATF concepts: FeCrAl strengthened by the dispersion of fine oxide particles (FeCrAl-ODS) cladding, SiC/SiC composite cladding, and doped-UO₂ fuel (Yamashita, et al. 2019).

Experimental studies have been conducted to evaluate key material properties for FeCrAl-ODS cladding, including strength and ductility, corrosion resistance, tritium permeability, wear resistance, iodine stress corrosion cracking resistance, and weldability (Sato, et al. 2018; Takahatake, et al. 2018; Kimura, et al. 2018). Sheet specimens have been tested at ORNL; loss-of-coolant accident tests have been conducted at ORNL as well, with FeCrAl-ODS showing excellent resistance to high-temperature steam oxidation, water quenching, and burst (Sakamoto, et al. 2019).

2.5 China

The Chinese ATF R&D program, led by China General Nuclear, has developed several ATF concepts including both cladding (coated Zr alloy, FeCrAl alloys, coated molybdenum alloy, and SiC) and fuel (high thermal conductivity UO₂) (Liu, et al. 2018). Ex-reactor testing has determined some thermal and mechanical properties of these concepts.

No plans for irradiation tests on these concepts have been indicated.

2.6 South Korea

The Korean Atomic Energy Research Institute (KAERI) and Korea Electric Power Corporation (KEPCO) R&D programs are moving forward with developing a number of ATF concepts including surface-modified Zr [Cr alloy-coated and oxide dispersion strengthened (ODS)], SiC

cladding, Fe-based alloy cladding, doped UO_2 fuel, microcell-, microplate- UO_2 (high thermal conductivity ceramic and metallic) fuel, and TRISO-SiC composite fuel (Yang, et al. 2019; Jang 2019). FeCrAl is being investigated as a coating for Zr-based alloy cladding.

CrAl-coated Zircaloy-4 cladding, CrAl-coated FeCrAl, ceramic microcell UO_2 fuel, and metallic microcell UO_2 fuel have been tested at Halden (Szöke, McGrath and Bennet 2017). PIE is expected for these samples (Kim, et al. 2019).

2.7 Oak Ridge National Laboratory

ORNL is researching and developing several ATF concepts: coated Zr-based cladding, FeCrAl cladding, and SiC/SiC cladding (Goldner, et al. 2019).

The lab has explored the high-temperature steam oxidation resistance of commercially available FeCrAl alloys (Kanthal APMT and Alloy 33) but has also conducted many studies to optimize the chromium and aluminum contents of new FeCrAl alloys. In recent years, work has continued to not only further alloy optimization for fabricability and baseline property assessment, but to also understand the effects of irradiation on the mechanical properties of FeCrAl alloys (Goldner et al. 2019).

An LTA containing C26M, a FeCrAl alloy developed by ORNL, and fabricated by GNF was inserted in Hatch Unit 1 in February 2018, discharged in February 2020. Additional rods will undergo a second cycle of irradiation. PIE, refabrication, and out-of-pile testing is planned at ORNL.

2.8 Idaho National Laboratory

Two irradiation testing campaigns are underway in the ATR at INL, testing fuel rodlets in the ATR reflector region (ATF-1 campaign) and under PWR conditions (ATF-2 campaign). Test specimens come from all industry teams. Transient testing in the Transient Reactor Test (TREAT) facility is planned for ATF concepts from all industry teams as well (Goldner et al. 2019).

3.0 Overview of Cr-Coated Zr Cladding

LWR fuel has traditionally been clad in zirconium alloys (Zircaloy-2, Zircaloy-4, ZIRLO®, and M5®). One of the responses to the development effort for ATF was to reduce the high-temperature corrosion effects of water and steam by adding Cr-coating to existing licensed Zr-based cladding. Table 3.1 provides a comparison of these near-term Cr-coated Zr ATF concepts.

This section provides an overview of the coating techniques that are being used by U.S. vendors. This section also discusses the possible interactions that can occur between chromium and zirconium, starting with the phase diagram and discussing the possibility of low-temperature eutectics and brittle phases.

Table 3.1. Comparison of Cr-coated concepts being pursued by U.S. nuclear fuel vendors.

Vendor	Coating	Application Process	Coating Thickness ^(a)
Westinghouse	Cr-coated ZIRLO®	Cold spray and polishing	20-30 µm
Framatome	Cr-coated M5®	Physical vapor deposition (PVD)	8-22 µm
GNF	ARMOR ^(b) coated Zircaloy-2	Proprietary	Proprietary

^(a) May change by the time of application. Typical cladding thickness is 600-750 µm.

^(b) 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.

3.1 Overview of Cr-Coating Techniques

This section provides an overview of the two Cr-coating techniques that have been identified for the Westinghouse and Framatome Cr-coated Zr cladding near-term ATF candidates. The technique used by GNF is proprietary.

Each coating technique has different processes for application of the coating. Specific coating technique processes used by the manufacturers may be partially or fully proprietary. The following techniques and processes are provided as an overview of potential processes being used by manufacturers of near-term ATF candidates. Details regarding other coating techniques can be found elsewhere (Geelhood and Luscher 2019).

3.1.1 Cold Spray

Cold spraying is a fairly new technology and has attracted serious attention since unique coating properties can be obtained by the process that are not achievable by conventional thermal spraying. This uniqueness is because coating deposition takes place without exposing the spray or substrate material to high temperature and without melting the sprayed particles.

Thus, oxidation and other undesired reactions such as interdiffusion between the substrate and coating can be avoided. Spray particles adhere to the substrate only because of their high kinetic energy on impact. For successful bonding, powder particles must exceed a critical

velocity on impact, which is dependent on the properties of the spray material. This requires new concepts for the description of coating formation but also indicates applications beyond the market for typical thermal spray coatings (Gärtner et al. 2006; Maier et al. 2018, 2019; Yeom et al. 2019).

3.1.1.1 Conventional High-Pressure Cold Spray

This process is “a kinetic spray process utilizing supersonic jets of a compressed gas to accelerate, at or near-room temperature, powder particles to ultra-high velocities (up to 1,500 m/s). The unmolten particles traveling at speed between 500 and 1,500 m/s plastically deform and consolidate on impact with their substrate to create a coating” (Fauchais, Heberlein and Boulos 2014).

The basis of the cold spray process is the gas-dynamic acceleration of particles to supersonic velocities and hence high kinetic energies. This is achieved using Laval (convergent-divergent) nozzles. The upstream pressure is between 2 and 2.5 MPa for typical nozzle through diameters in the range of 2-3 mm. Gases used are N₂, He, or their mixtures at very high flow rates (up to 5 m³/min). For stable conditions, typically the mass flow rate of the gas must be 10 times that of the entrained powder. For a powder flow rate of 6 kg/h, this means a volumetric flow rate of 336 m³/h for helium and 52.3 m³/h for nitrogen. Gases introduced (nitrogen or helium) are preheated up to 700-800°C to avoid their liquefaction under expansion and increase their velocity. With the highest gas flow rate, this means that the heating device must be capable of heating 90 m³/min to temperature up to 700-800°C. As particles are injected upstream of the nozzle throat, the powder feeder must be at slightly higher pressure compared to the upstream chamber pressure. Especially when spraying with He, the spray is performed within an enclosure to recycle gas (Fauchais et al. 2014).

Particles adhere to the substrate only if their impact velocity is above a critical value. Depending on the sprayed material, this may vary between about 500 and 900 m/s. The spray pattern covers an area of roughly 20-60 mm², and spray rates are about 3-6 kg/h. Feed stock particle sizes are typically between 1 and 50 µm and deposition efficiencies reach easily 70-90%. Only ductile metals or alloys are sprayed owing to the impact-fusion coating buildup. Blends of ductile materials (>50 vol%) with brittle metals or ceramics are also used. It is important to emphasize that the substrate is not really heated by the gas exiting the gun (up to 200°C at the maximum). Current and expected application for cold spray coatings are electronic and electrical coatings, and coatings for the aircraft and automotive industries for localized corrosion protection (Fauchais et al. 2014).

3.1.1.2 Low-Pressure Cold Spray

The cost of high-pressure spray processes is rather high, especially because of the quantity of gases consumed (especially if it is helium), and the process is not adapted at all to spraying for onsite repair of parts. Portable equipment using air has been developed where the upstream pressure is below 1 MPa, typically 0.5 MPa, with a much lower gas consumption of about 0.4 m³/min (Fauchais et al. 2014).

Less power is required to preheat air, which is less expensive than N₂ or He. The powder feeder can be simplified because pressurization is not necessary, and the powder can be injected downstream of the nozzle throat. With gas velocities in the 300-400 m/s range, the critical velocity of particles is not attained. To achieve coatings, small metal particles are mixed with bigger ones, which can be ceramics. Big particles mostly rebound upon impact, their role being

to “press” the small metal particles onto the substrate and the previously deposited layers (shot peening effect). Powder recuperation is hardly possible for multicomponent powder mixtures. The deposition efficiency is also much lower than with high-pressure guns, but the process is attractive for short production runs (Fauchais et al. 2014).

3.1.2 Physical Vapor Deposition

Physical vapor deposition (PVD) is a broad term used to describe the deposition of atoms, molecules, or the combination of atoms and molecules via condensation. In general, the term PVD encompasses evaporation, sputtering, and ion plating processes. These three processes are described in the following subsections (Pierson 1999; Grainger 1998).

3.1.2.1 Evaporation

Evaporative coatings are applied by heating the coating material (i.e., source) above the boiling point under low pressure ($<10^{-3}$ Pa). This sends atoms or molecules through a cosine distribution of trajectories in a straight line to the substrate where they condense and form a thin film. At these low pressures, the mean-free path is large relative to the distance between the source and substrate and few collisions occur before the species condense on the substrate. This may lead to uneven coating thickness because the thickest part will be closest to the source. Uneven coatings may be avoided by employing planetary substrate holders and multiple sources. Evaporative coatings offer relatively high deposition rates (up to 75 $\mu\text{m}/\text{min}$) but complex shapes are difficult to accommodate, and the coatings often exhibit poor adhesion. A variety of different techniques are available for depositing evaporative coatings, such as reactive evaporation, plasma assisted or activated reactive evaporation, and molecular beam epitaxy. These techniques are described briefly in this section (Pierson 1999).

Reactive evaporation can be used to deposit refractory carbides, nitrides, and oxides, which have extremely high boiling points and tend to dissociate during evaporation. During reactive evaporation, the nonmetallic element of the coating (e.g., carbon, nitrogen, or oxygen) is introduced into the gas phase and a pure metal source is used (e.g., nitrogen used with a titanium source can produce titanium nitride) (Pierson 1999).

Plasma assisted or reactive evaporation can be used to enhance deposition rates using a plasma. A plasma is an ionized gas that is formed in the presence of an electromagnetic field under vacuum. The presence of the plasma enhances reactions in the gas phases and the growth kinetics of the deposit (Pierson 1999).

Molecular beam epitaxy (MBE) is another form of evaporative coating and can be used to produce extremely pure and very thin films with abrupt composition changes. The deposition rate for MBE is very slow and the process is still considered experimental. Nevertheless, MBE is considered for extremely exacting electronic and optoelectronic applications (Pierson 1999).

3.1.2.2 Sputtering

Sputtering is a technique used to create thin films. It is extensively used in the hard coating industry. High-quality coatings of refractory compounds and metals can be readily produced with good adhesion and composition control. In addition, since sputtering is not a thermally activated process, it is not associated with high temperature requirements like other coating processes (Pierson 1999).

During the sputtering process, a source (or target) is placed in a high vacuum and bombarded with gas ions (typically argon) that have been accelerated by high voltage, producing a glow discharge or plasma. Atoms from the target are physically ejected by the momentum transfer and travel across the vacuum chamber and are deposited on a substrate surface. Since the process is performed under low pressure, the mean-free path of the target atoms is relatively long, thus permitting the ejected atoms to condense on the intended surface (Pierson 1999).

Sputtering requires low pressure to remove all traces of background and contaminant gases, which could degrade the coating. This is typically achieved by cryogenic pumps capable of producing a vacuum of about 10^{-5} Pa with good pumping speed. After evacuation, the system is refilled with argon to a partial pressure of 0.1 to 10 Pa. Higher pressure, by placing too many argon atoms in the path of the ions and ejected atoms, would not allow these atoms or molecules to travel unimpeded by collision, effectively reducing the mean-free path and reducing the deposition rate. Sputtering can also be performed in the presence of a small partial pressure of hydrocarbons, nitrogen, or oxygen to react with ejected atoms and form carbide, nitride, or oxide coatings in a process called reactive sputtering. It is important to note, however, that reaction between the target material and the reactive species can poison the target and interfere with deposition (Pierson 1999).

The general disadvantages of sputtering include a relatively low deposition rate and a line-of-sight deposition characteristic that make the coating of deep holes and trenches difficult. This can be overcome to some extent by operating at higher pressure (but at some sacrifice in deposition rate) or by using three-dimensional grids. However, an advantage of sputtering is that the high energy of sputtered particles improves adhesion and produces a denser and more homogenous coating than evaporation (Pierson 1999).

The following is a list of commonly used sputtering techniques and a description of their attributes (Pierson 1999):

- Diode sputtering – Simplest sputtering technique but requires an electrically conductive target, has low energy efficiency, and electron bombardment may cause significant damage of the substrate.
- Radio-frequency sputtering – Frequencies above 50 kHz can sputter insulators, but the process has low deposition rates.
- Triode sputtering – An additional cathode is used to sustain the plasma, but this configuration is more complicated and may contaminate the deposit.
- Magnetron sputtering – Magnetically enhanced cathodes (magnetrons) have considerably expanded the potential of sputtering. The magnetron sends the electrons into spiral paths to increase collision frequency and ionization. Deposition rates are high, and the process does not cause electron radiation damage.

3.1.2.3 Ion Plating

In ion-plating deposition, the substrate and deposited film (as it forms) are subjected to bombardment by particles (ions, atoms, molecules) that alter the formation process and properties of the coating. The process is also called ion-beam assisted deposition (Pierson 1999).

Two basic versions of the ion beam plating process exist: plasma-based ion plating and vacuum-based ion plating. The coating material is vaporized in a manner similar to evaporation.

Typically, the plasma is obtained by biasing the substrate to a high negative potential (5 kV) at low pressure. The constant ion bombardment of the substrate sputters off some of the surface atoms, which results in improved adhesion and reduced impurities relative to other ion plating techniques. Surface coverage of discontinuities is also improved (Pierson 1999).

Reactive ion plating is similar to reactive sputtering and evaporation with applications in optical, wear, abrasion, lubrication, and decorative coating (Pierson 1999).

Another variant of ion plating is vacuum arc, cathodic arc, and multi-arc plating. This technique uses an electric arc to generate a plasma of target atoms. It is typically used for hard coatings like CrN and TiN. It typically forms dense, adherent films, but can result in macro-particle formation.

3.2 The Cr-Zr Phase Diagram

The substrate-coating interface must provide a sufficiently strong and stable bond throughout service to prevent delamination and benefit from the addition of the engineered coating. While some coating processes may result in a sufficiently adherent mechanical bond, many processes are conducted at elevated temperatures that result in a chemical bond at the substrate-coating interface. While chemical bonds are generally stronger and more robust than mechanical bonds, the characteristics of chemical bonds are ultimately dependent on the solid-state reactions at the interface. Cold spray and PVD are both examples of mechanically bonded coatings. Surface preparation is one of the most important aspects of any coating process. This is particularly critical for mechanically bonded coatings.

Interfacial solid-state reactions between the substrate and coating material occur in a manner similar to a diffusion couple. The resulting crystalline phase assemblage at the substrate-coating interface will influence the overall performance of the coated component and, as a result, the strength and stability of these interfacial phases is a critical aspect of an engineered coating.

Equilibrium phase diagrams of substrates and coating materials provide a convenient means of illustrating the range of crystalline phase assemblages that may be present at the interface. In addition to identifying interfacial phases, the phase diagram also provides transformation temperatures (e.g., melting, eutectic, various crystallographic morphologies, liquid phase) that may place additional limitations on the service environment of the coated component.

For chromium-coated zirconium substrates, the Zr-Cr phase diagram illustrates the equilibrium phases that may exist at the substrate-coating interface (see Figure 4.1; Arias and Abriata 1986). For ceramic coatings, such as CrN or Cr₂O₃, it would be necessary to examine a ternary phase diagram to determine the equilibrium phases that could be present. There is currently limited information regarding ternary phase diagrams of Cr-Zr-N or Cr-Zr-O, and none were located during this review.

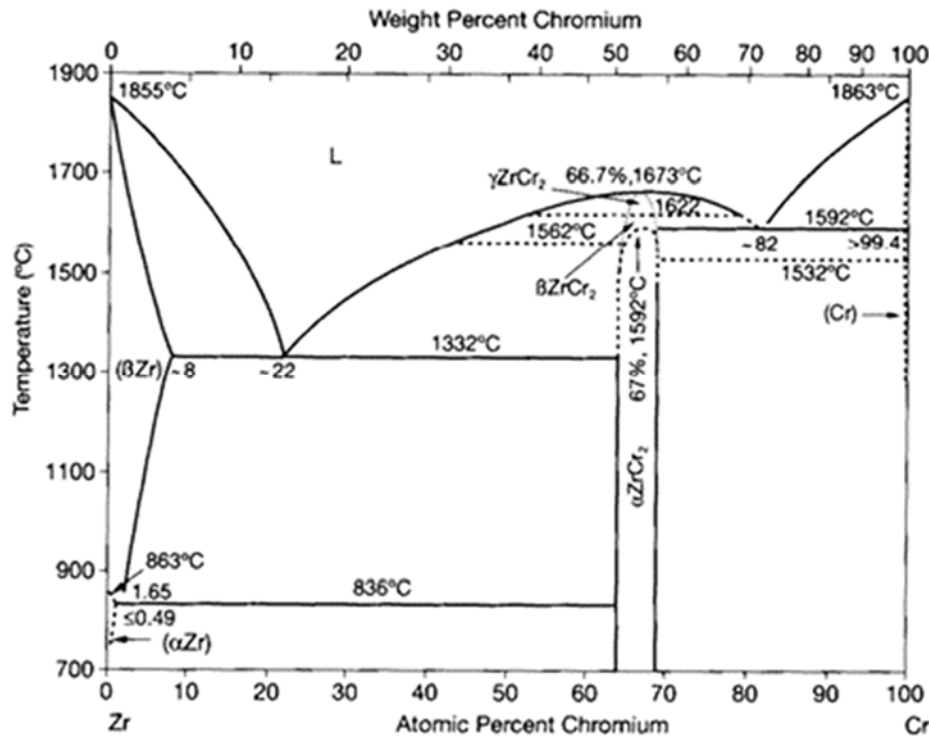


Figure 3.1. Zr-Cr phase diagram (Arias and Abriata 1986).

3.3 Eutectics

A eutectic isotherm is present on each side of the ZrCr_2 intermetallic in the Cr-Zr phase diagram. These isotherms represent an equilibrium between the liquid phase and two solid phases. As the liquid phase cools below the eutectic isotherm, it decomposes into two solids. Conversely, the two solid phases, when in contact, will form a liquid phase when heated above the eutectic isotherm. Clearly, the formation of a liquid phase below the melting points of a component's constituents has direct implications for temperature limitations on service environment, and hence understanding the temperatures of these eutectics is critical.

Of the two eutectics in the Zr-Cr phase diagram, the higher eutectic temperature occurs at 1592°C. This is also the temperature of the maximum solid solubility for Zr (< 0.6 at%) in Cr. The lower eutectic temperature occurs at 1332°C and corresponds to the maximum solubility limit for Cr (8 at%) in $\beta\text{-Zr}$ (Arias and Abriata 1986). This lower eutectic temperature (1332°C) is of greater importance for establishing temperature limits for the Zr-Cr system as it represents the more limiting condition. In an engineered coating, such as a Cr-coated Zr substrate, it is possible for all phases to be present following fabrication and, depending on the service environment, these phases may form in service. Consequently, the limiting temperature to avoid liquid phase formation is set by the lower eutectic isotherm.

While it is important to understand that eutectics exist for Cr-coated Zr cladding, conditions associated with spent fuel storage and transportation (400°C for normal conditions and 570°C for accident conditions) do not approach the eutectic temperatures identified. Furthermore, the

inner surface of the cladding is not coated, so there is no additional concern regarding the fuel and cladding interaction.

3.4 Brittle Phases

An intermetallic compound, ZrCr_2 , is present in the Zr-Cr phase diagram. Intermetallic compounds generally have high melting points and low densities and exhibit superior corrosion and oxidation resistance that make them candidates for high-temperature structural materials. The ZrCr_2 intermetallic is a topologically close-packed Laves phase of the form AB_2 . Intermetallic compounds such as these are typically brittle at low temperature due to their complex crystal structure.

The intermetallic compound, ZrCr_2 , is present in three stable phases on the Zr-Cr phase diagram. These include α , β , and γ - ZrCr_2 , which are the low, intermediate, and high-temperature phases of the compound. The α -phase has a composition range of 64-69 at% Cr at 900°C and transforms to the β -phase at about 1592°C. The high temperature γ -phases is only stable between 1622°C (β to γ transformation) and its congruent melting temperature (1673°C). Note that severe experimental difficulties are found in the ZrCr_2 compositional range due to the long times and high temperature required to attain stable equilibrium. Consequently, the details of the $\beta + \gamma$ phase region of the Cr-Zr phase diagram is incomplete and rather speculative (Arias and Abriata 1986). This would only be a concern above 1592°C

The intermetallic compound, ZrCr_2 , can form due to diffusion of atoms in the Zr substrate and Cr atoms in the coating. The formation of this intermetallic has been observed with a thickness of 4 μm after 66 hours at 775°C (Sweeney and Batt 1964). The formation of this intermetallic has also been observed with a thickness between 1 and 4.5 μm after 49 to 225 hours at 750°C to 850°C (Wenxin and Shihao 2001). As with the eutectics previously discussed, the conditions associated with spent fuel storage and transportation (400°C for normal conditions and 570°C for accident conditions) do not approach the temperatures where significant interdiffusion would occur beyond what may have occurred in-reactor.

3.5 Cr-Coated Zr Cladding Degradation and Failure Modes

Unlike in-reactor safety analysis, there are no specific safety limits put on the fuel cladding for spent fuel storage and transportation analysis. However, to certify that DSS or transportation package is safe, a number of safety analyses are performed and an accurate knowledge of the thermal and mechanical state of the fuel cladding is necessary. Therefore, it is critical to understand how these properties have changed relative to the fresh fuel condition following irradiation.

It is known that irradiation of Zr-alloy cladding will cause degradation to the cladding such that there is a possibility of failure either in-reactor or out of reactor during storage and transportation.

In-reactor, the following changes to Cr-coated Zr tubes are possible relative to the unirradiated conditions:

- Increase in yield stress
- Decrease in ductility

- Decrease in fatigue life
- Coating cracking or delamination
- Cr-Zr interdiffusion
- Radiation effects on Cr
- Galvanic corrosion

Additionally, a number of aging-related damage mechanisms that could impact the cladding during long-term storage have been identified for Zr-alloy tubes and are likely applicable to Cr-coated Zr cladding as well. These include:

- Embrittlement (typically caused by hydride reorientation in Zr-alloy tubes)
- Delayed hydride cracking
- Thermal and athermal creep

The regulations and guidance for spent fuel analysis are discussed in greater detail in Section 5.0, including a discussion of information that would be required to adequately model Cr-coated Zr cladding.

4.0 Overview of GNF FeCrAl Alloys

FeCrAl alloys have historically been used in industrial applications where high-temperature oxidation resistance is needed. Development of FeCrAl alloys has been performed by commercial entities, national laboratories, and universities, with collaboration between the different research sectors. Both wrought FeCrAl and powder metallurgy based FeCrAl alloys are under development. Within the nuclear industry, focus has been on the wrought alloys, considered to be “nuclear grade,” which in this context means an optimized composition to perform within the full range of reactor operating conditions.

FeCrAl alloys consist of iron (Fe), chromium (Cr), and aluminum (Al), with minor alloying additions for various purposes. There are commercially available variants; however, the main focus of U.S. R&D programs is to develop a wrought oxidation-resistant alloy variant; the Japanese effort intends to greatly improve on the strength by pursuing ODS FeCrAl alloys (Terrani 2018).

GNF remains the only fuel vendor in the U.S. with FeCrAl cladding planned for the near term. GNF has tested several different FeCrAl alloys, including Kanthal APMT, C26M, and MA956. While GNF has not publicly stated which FeCrAl alloy will be used for IronClad, the 2 unfueled IronClad rods irradiated at Hatch were C26M, 8 fueled IronClad rods inserted in Clinton were C26M, and 16 unfueled rods inserted in Clinton were C26M, APMT, and MA956. The compositions of these three alloys are shown in Table 4.1. More detail on these three alloys is provided below.

Table 4.1. Compositions (by weight percent) of C26M, Kanthal APMT, and MA956 FeCrAl alloys.

Alloy	Fe	Cr	Al	Mo	Ti	C	Si	Mn	Y	Cu	Co	Ni	P
C26M ^(a)	Balance	12	6.0	2.0	-	-	0.2	-	0.03	-	-	-	-
Kanthal APMT ^(b)	Balance	20.5-23.5	5.0	3.0	-	0.08 max	0.7 max	0.4 max	-	-	-	-	-
MA956 ^(c)	Balance	18.5-21.5	3.75-5.75	-	0.2-0.6	0.1 max	-	0.30 max	0.3-0.7 ^(d)	0.15 max	0.3 max	0.50 max	0.02 max

^(a) (Yamamoto et al. 2019)
^(b) (Kanthal 2019)
^(c) (Special Metals 2004)
^(d) Values given are for Y₂O₃

A detailed evaluation of each alloy variant not being considered by GNF is outside the scope of this report.

4.1 FeCrAl Design

FeCrAl alloys are fully ferritic (body-centric-cubic structure) with typically no phase transformation to or from austenite (face-centered-cubic structure) between liquidus temperature and room temperature due to the Cr and Al additive effects on Fe-based alloys (Field et al. 2018). Cr additions contribute to corrosion resistance by forming a layer of chromium oxide (or chromia) under normal conditions; Al additions improve high-temperature oxidation resistance by forming an aluminum oxide (or alumina) layer under accident conditions (Rebak 2018a). The Cr additions further stabilize the alumina layer in high-temperature steam.

Figure 4.1 shows the Fe-Cr binary phase diagram and indicates the formation of the brittle Cr-rich α' phase at relatively low temperatures where LWRs are operated. Al addition reduces the driving force of Cr-rich α' phase formation even in alloys with relatively high Cr contents (Field et al. 2018; Wukusick 1966).

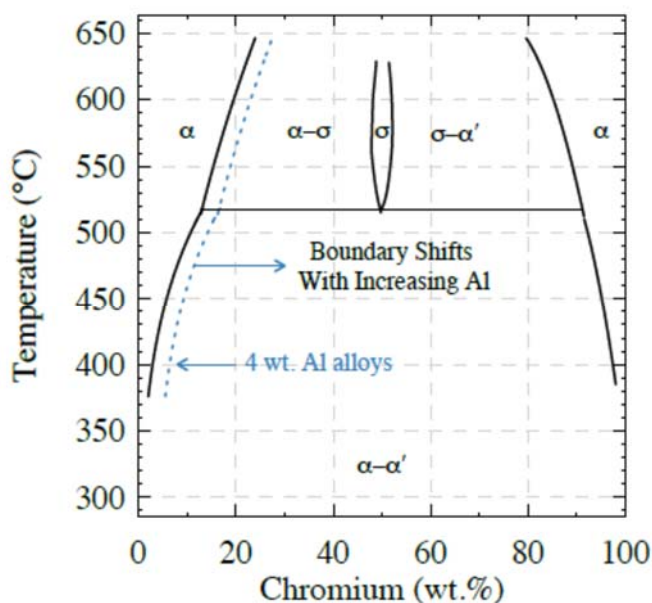


Figure 4.1. Fe-Cr binary alloy phase diagram showing phase boundaries of α -Fe, α' -Cr, and σ -FeCr. The effect of a 4 w/o Al addition on the alpha- α' phase boundary is also shown as example (Field 2018; Wukusick 1966).

The Cr and Al contents need to be balanced not only for surface protection but also for property control. If the Cr content is too high, it can lead to embrittlement as a result of the α' phase precipitation (Field et al. 2018), as shown in Figure 4.2. Small additions of yttrium can enhance the oxidation resistance of the alloy (Kim et al. 2019). ODS variants can have a higher strength and increased high-temperature creep resistance due to the dispersion of fine oxide particles (Yano et al. 2017).

At ORNL, alloying additions of molybdenum and niobium have been made to recent FeCrAl alloys¹, intended to increase alloy strength. Mo addition increases alloy strength through solid-

¹ FeCrAl alloys developed by ORNL are referred to as “model” alloys in literature.

solution hardening; Nb addition increases alloy strength through precipitate strengthening by the formation of Fe₂Nb-type Laves phase particles (Raiman et al. 2020).

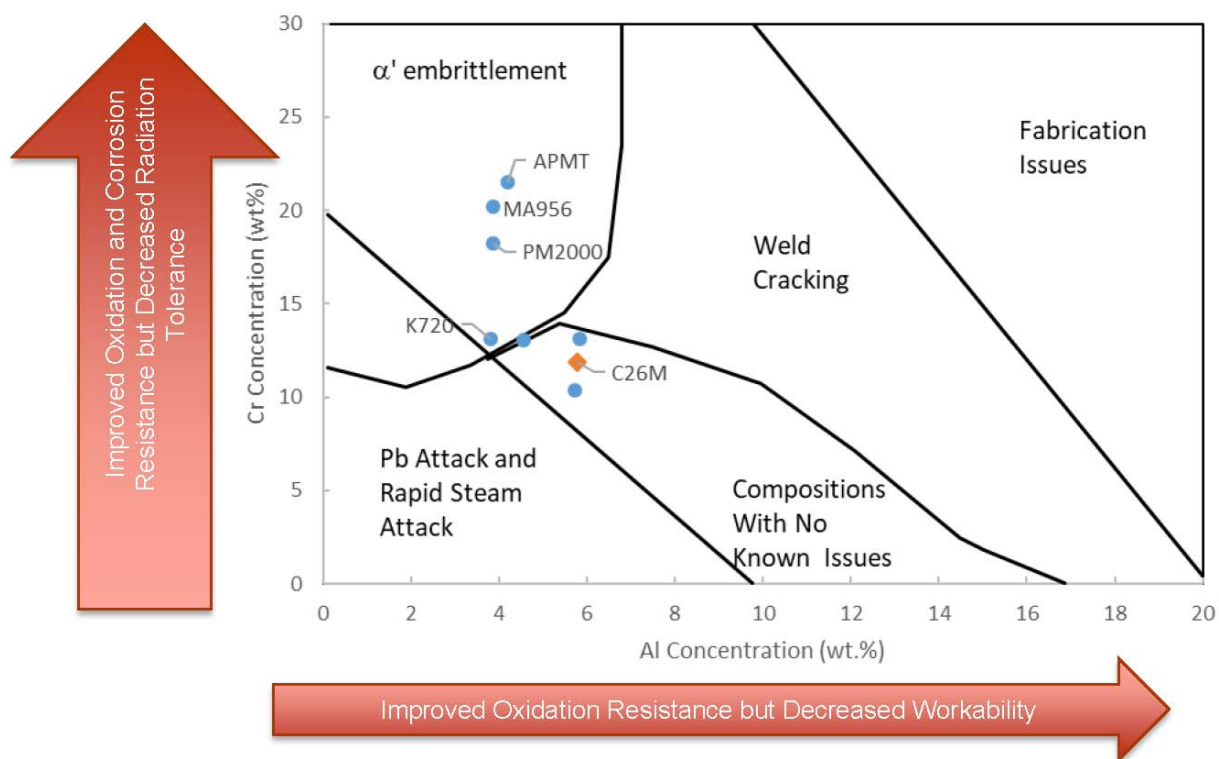


Figure 4.2. Impact of chromium and aluminum concentration in FeCrAl alloys (Yamamoto, Field, et al. 2020).

The three alloys being considered by GNF are C26M (developed by ORNL), Kanthal APMT (commercially available), and MA956 (commercially available). As seen in Figure 4.2, APMT and MA956 will likely exhibit α' embrittlement, unlike C26M which has no known issues.

4.2 C26M

C26M is a FeCrAl alloy developed by ORNL and the subject of continual study to determine the effects of minor alloying additions, including yttrium, zirconium, cesium, carbon, and manganese (Yamamoto et al. 2019). The nominal composition is listed in Table 4.1. C26M is manufactured by traditional melting and has been fabricated into rodlets and segmented full-length rods at the industrial GNF facilities in Wilmington, NC (Richardson and Medema 2019).

The alloy is weldable by the tungsten inert gas (TIG) method, without cracks, porosity, or internal oxidation in the weld seam and no grain-boundary sensitization (Rebak et al. 2018b).

C26M has been irradiated in the ATR and Plants Hatch and Clinton (Richardson and Medema 2019).

4.3 Kanthal APMT

Kanthal APMT is a pre-oxidized commercially available ODS powder metallurgical FeCrAl alloy (Kanthal 2019) with a higher chromium content (see Table 4.1) and finer grain size (Rebak et al. 2019) than C26M.

The pre-oxidation treatment introduces a thin layer of chromia in the outer part of the alumina layer (Schuster, Crawford, and Rebak 2017). The high Cr content significantly improves the alloy's corrosion rate compared to other FeCrAl alloys (Wang et al. 2020; Rebak, Jurewicz, and Kim 2017) but as seen in Figure 4.2, Kanthal APMT is likely to exhibit α' -phase embrittlement due to the Cr and Al contents. The alloy has been exposed to high-temperature water simulating conditions of both BWR and PWR coolant conditions; the thin surface oxide layer was seen to be approximately ten times thinner than the oxide produced for Zircaloy-2 (Rebak, Terrani, and Fawcett 2016).

The alloy is weldable by the TIG method, without cracks, porosity, or internal oxidation in the weld seam and no grain-boundary sensitization (Rebak et al. 2018b).

Segmented full-length rods (non-fueled) and rodlets have been irradiated in Clinton Cycle 20 and the ATR, respectively (Richardson and Medema 2019; Harp, Cappia, and Capriotti 2018).

4.4 MA956

MA956 is a commercially available ODS FeCrAl alloy produced by mechanical alloying (Special Metals 2004). As seen in Figure 4.2, MA956 is likely to exhibit α' -phase embrittlement due to the Cr and Al contents.

Conventional TIG welding is possible but produces relatively low-strength joints (Special Metals 2004).

MA956 has been irradiated in the ATR (Zhang et al. 2020).

4.5 Possible Eutectics

Interactions between materials may occur during a severe accident and can contribute significantly to the general progression of the accident. Several materials must be considered for possible materials interaction with FeCrAl cladding:

- Fuel (UO_2);
- Control materials (Ag-In-Cd and B_4C);
- Burnable absorbers (e.g., Gd); and
- Various hardware such as springs, grids, and sheaths (Inconel and/or SS-304).

The interactions of FeCrAl (composition Fe-11.9Cr-6.2Al-0.50Ti-0.57Ce-0.20O) with UO_2 and B_4C have been studied at 1573 K (1300°C) and 1673 K (1400°C), temperatures relevant to DBA conditions (Sakamoto et al. 2016). Compared to Zircaloy-4, there was no distinct reaction between the FeCrAl- UO_2 couple. A uniform alumina layer and no clear ingress of uranium were observed. Similarly, the FeCrAl- B_4C couple showed excellent resistance to materials interaction.

The interactions of FeCrAl (Alloy B136Y, with composition Fe-13Cr-6.2Al-0.03Y) with SS-304, Inconel, and B₄C have been studied at temperatures ranging from 1300 °C to 1500 °C (Robb, Howell, and Ott 2018). These tests did not show signs of interaction for test temperatures up to 1400 °C for the FeCrAl/SS-304 combination and up to 1450 °C for the FeCrAl/Inconel and FeCrAl/B₄C combinations.

The FeCrAl/B₄C test conducted at 1500 °C appeared to have some melting of the FeCrAl; however, part of the testing apparatus fell during the experiment so the test will be repeated in the future. Fe/B₄C is known to form a low melting eutectic at approximately 1150 °C. It is postulated that a thin oxide layer protects the FeCrAl from the B₄C, which can be confirmed by future by cross-sectional micrographs.

During the QUENCH-19 test performed at Karlsruhe Institute of Technology, FeCrAl (Alloy B136Y, with composition Fe-13Cr-6.2Al-0.03Y) cladding was damaged due to probable melting or by interaction with molten SS-304 thermocouples, which have a melting temperature in the range of 1400 to 1450 °C (Stuckert et al. 2019). This could indicate eutectic interaction between FeCrAl and SS-304; however, it is possible that the cladding reached the melting point of FeCrAl as there were uncertainties in the temperature measurements and several thermocouples fail.

While it is important to understand that eutectics exist for FeCrAl cladding, conditions associated with spent fuel storage and transportation (400°C for normal conditions and 570°C for accident conditions) do not approach the eutectic temperatures identified. Furthermore, the inner surface of the cladding is not coated, so there is no additional concern regarding the fuel and cladding interaction.

4.6 FeCrAl Cladding Degradation and Failure Modes

Unlike in-reactor safety analysis, there are no specific safety limits put on the fuel cladding for spent fuel storage and transportation analysis. However, to certify that a DSS or transportation package is safe, several safety analyses are performed and an accurate knowledge of the thermal and mechanical state of the fuel cladding is necessary. Therefore, it is critical to understand how these properties have changed relative to the fresh fuel condition following irradiation.

It is known that irradiation of Zr-alloy cladding will cause degradation to the cladding such that there is a possibility of failure either in-reactor or out of reactor due during storage and transportation. It is likely that similar degradation will occur in FeCrAl cladding.

In-reactor, the following changes to FeCrAl tubes are possible relative to the unirradiated conditions:

- Increase in yield stress
- Decrease in ductility
- Decrease in fatigue life
- Radiation Effects on FeCrAl
- Galvanic corrosion

Additionally, a number of aging-related damage mechanisms that could impact the cladding during long term storage have been identified for Zr-alloy tubes and may be applicable to FeCrAl cladding as well. These include:

- Embrittlement
- Delayed hydride cracking
- Thermal and athermal creep

The regulations and guidance for spent fuel analysis are discussed in greater detail in Section 5.0, including a discussion of information that would be required to adequately model FeCrAl cladding.

5.0 Storage and Transportation of Spent Nuclear Fuel

The safety analyses for storage and transportation of spent nuclear fuel (SNF) is somewhat different than the in-reactor performance. The storage and transportation of SNF has not been historically considered as a part of the fuel design process. There are currently no in-reactor operating restrictions that are in place because of SNF considerations. However, the peak cladding temperature and average-rod hoop stresses during drying operations are limited via guidance. These limits may need to be revised depending on advances in ATF fuel cladding performance. Additionally, the individual rod power histories are not available for the analysis of SNF during drying, loading, storage, or transportation. Therefore, the safety limits for storage and transportation of SNF must be developed for the most limiting fuel rods at the maximum expected burnup.

For spent fuel storage and transportation, burnup has a profound impact on the safety analysis. Currently licensed cladding is affected in the following ways: As burnup progresses, cladding strength goes up, and ductility down; cladding is thinned by the corrosion process; increased cladding hydrogen concentration causes embrittlement; and increased fission gas release leads to increased rod internal pressure and increased radioactive source term for accident analyses.

It is worth noting that an additional motivation for development of ATF fuels is the side benefit of improving fuel performance in high burnup conditions. Accordingly, burnup-dependent guidance for storage and transportation applications may need to be revised yet again once an assessment of ATF fuel concepts can be made for high burnup operation.

PNNL performed a critical review of the regulations and guidance regarding the storage and transportation of SNF. The general conclusions of this review are that the relevant regulations (10 CFR 72) do not specifically prohibit the use of either of the near-term ATF cladding concepts as the requirements are fairly general. NRC standard review plans (SRP) for dry storage systems (NUREG-1567 and NUREG-1536, replaced by NUREG-2215) and transportation of SNF (NUREG-1617, replaced by NUREG-2216) (US Nuclear Regulatory Commission 2020a; US Nuclear Regulatory Commission 2020b) mention Zr-alloy cladding and stainless steel. There is no mention of either near-term ATF concept in these documents; however, due to the similarity of Cr-coated cladding with Zr-alloy cladding, and FeCrAl cladding with stainless steel cladding, it is possible to assess the data needs and applicability of these alloys to the requirements and recommendations in these documents. The one area that deserves additional scrutiny from these documents is the empirically derived temperature and stress limits in the SRP.

- The integrity of zirconium-based alloy cladding and all fuel burnups (low and high): The maximum calculated fuel-cladding temperature should not exceed 400°C (752 °F) for normal conditions of storage and short-term loading operations, including DSS drying and backfilling.
- A higher temperature limit may only be used for low burnup SNF (less than 45 gigawatt days MTU), as long as the applicant can demonstrate that the best estimate cladding hoop stress is equal to or less than 90 MPa (13.1 ksi) for the proposed temperature limit.
- During loading operations, repeated thermal cycling should be limited to less than 10 cycles when the cladding temperature difference exceeds 65°C (149°F).

- For off-normal and accident conditions the maximum cladding temperature should not exceed 570°C (1,058°F).
- For the integrity of stainless-steel cladding, the maximum calculated fuel cladding temperature should not exceed 570°C (1058°F) for off-normal and accident conditions and the maximum calculated fuel cladding temperature should not exceed 400°C (752°F) for normal conditions of storage and short-term loading operations, including storage container drying and backfilling.
- Hydrogen content: Per NUREG-2214, for burnups above 45 GWd/MTU and up to 62 GWd/MTU, the hydrogen content is noted for Zircaloy-2, Zircaloy-4, ZIRLO®, and M5®. NUREG-2215 (US Nuclear Regulatory Commission 2020a) does not call out a specific hydrogen content

These temperature and stress limits are empirically derived for current fuel designs with current cladding materials. Each of these limits should be assessed to determine if they are applicable to any near-term ATF cladding concept. With the hydrogen content, it is noted that the Cr coating will likely reduce the oxide thickness and subsequent hydrogen content to very low values and should not be an issue for Cr-coated Zr cladding. Embrittlement can be caused by interdiffusion between Cr and Zr, but this is not expected to be significant (see Section 3.4). For FeCrAl, iron is not embrittled by hydrogen as zirconium is and thus hydrogen embrittlement is not an issue (Hales and Gamble 2015). NRC should be made aware of any other environmental factor that could embrittle either ATF cladding concept.

The following subsections provide a critical evaluation of the documents that provide the regulatory framework for the safety analyses regarding the, wet storage (Section 5.1), dry storage (Section 5.2), and transportation (Section 0) of SNF as it relates to near-term ATF concepts.

5.1 Wet Storage of Spent Nuclear Fuel

This section discusses the wet storage of SNF, including the current regulatory framework and its application to near-term ATF concepts.

5.1.1 Current Regulatory Framework

The regulations related to wet storage of SNF are provided in 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities” (US Nuclear Regulatory Commission 2017) as General Design Criteria (GDC) 61, “Fuel Storage and Handling and Radioactivity Control.” Specifically, GDC 61 requires (1) periodic inspections; (2) suitable radiation shielding; (3) appropriate containment, confinement, and filtering systems; (4) residual heat removal capability consistent with its importance to safety; and (5) prevention of significant reduction in fuel storage inventory under accident conditions. To augment those requirements, the spent fuel pool design basis is also covered by GDC 2, “Design Bases for Protection Against Natural Phenomena”; GDC 4, “Environmental and Dynamic Effects Design Bases”; and GDC 63, “Monitoring Fuel and Waste Storage.”¹

To meet the requirements of GDC 61, 2, 4, and 63, it is important that the spent fuel storage pool structures, systems, and components be designed to accomplish the following:

¹ 10 CFR Part 72 can also apply to wet storage, if the wet storage facility is away from a reactor (e.g., GE-Morris spent fuel pool is licensed under Part 72 as a site specific ISFSI)

- Prevent loss of water from the fuel pool that would lead to water levels that are inadequate for cooling or shielding
- Protect the fuel from mechanical damage
- Provide the capability to limit potential offsite exposures in the event of a significant release of radioactivity from the fuel or significant leakage of pool coolant
- Provide adequate cooling to the spent fuel to remove residual heat

5.1.2 Application to Cr-Coated Zr Cladding

For Cr-coated Zr cladding, it is unlikely that the irradiated cladding mechanical properties will be significantly different from the uncoated Zr-alloy irradiated properties. If this is confirmed with test data from irradiated Cr-coated Zr cladding, then it would be reasonable to use existing spent fuel pool analyses for Cr-coated Zr fuel rods.

Additionally, the heat load from Cr-coated Zr clad fuel rods is expected to be the same as that of uncoated Zr-alloy clad fuel rods, so there will likely be no impact on the residual heat analysis assuming current burnup limits remain the same.

5.1.3 Application to FeCrAl Cladding

The mechanical properties of FeCrAl cladding are not the same as Zr-alloy cladding. Additionally, initial FeCrAl cladding designs use thinner cladding than the Zr-alloy cladding they are replacing. An analysis to show that the fuel is protected from mechanical damage should be performed using representative cladding dimensions and mechanical properties taken from irradiated cladding tubes.

Additionally, although the heat load from FeCrAl clad fuel rods is expected to be the same as that of uncoated Zr-alloy clad fuel rods at current burnup limits, there is the possibility of Co-60 activation from Fe that could lead to increased worker dose and thus it should be considered in the spent fuel analysis.

5.2 Dry Storage of Spent Nuclear Fuel

This section discusses the dry storage of SNF, including the current regulatory framework and its application to near-term ATF concepts.

5.2.1 Current Regulatory Framework

The regulations in 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste" (US Nuclear Regulatory Commission 2015), include a number of fuel-specific and dry storage system (DSS)-specific requirements that may be dependent of the design basis condition of the fuel cladding. 10 CFR 72.44(c) states that a specific license for dry storage of SNF is to include technical specifications that, among other things, define limits on the fuel and allowable geometric arrangements. Additionally, 10 CFR 72.236(a) states that, a Certificate of Compliance for a DSS design must include specifications for:

- The type of spent fuel (i.e., BWR, PWR, or both)
- Maximum allowable enrichment of the fuel prior to any irradiation

- Burnup
- Minimum acceptable cooling time of the spent fuel before storage in the spent fuel DSS
- Maximum heat designed to be dissipated
- Maximum spent fuel loading limit
- Condition of the spent fuel (i.e., intact assembly or consolidated fuel rods)
- Inerting atmosphere requirements.

The condition of the SNF cladding is critical to the storage as 10 CFR 72.122(h)(1) states that the SNF cladding is to be protected against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. Additionally, CFR 72.122(l) states that the DSS must be designed to allow ready retrieval of the SNF.¹

In addition to these regulations, the NRC staff have provided NUREG-1536, Rev. 1, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General Licensing Facility” (US Nuclear Regulatory Commission 2010), and NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities” (US Nuclear Regulatory Commission 2000). These guidelines have recently been replaced by NUREG-2215 (US Nuclear Regulatory Commission 2020a). These standard review plans provide guidance to the staff on reviewing applications for DSS Certificates of Compliance or ISFSI site-specific licenses under 10 CFR Part 72. The SRP lays out the following evaluations that should be performed during the safety evaluation of a DSS shown in Figure 5.1.

¹ “Ready retrieval” is defined as an ability to retrieve a fuel assembly intact using its normal lifting hardware

Chapter 1 – General Information Evaluation			
Site Description (SL)	DSS or DSF Description and Operational Features	Engineering Drawings	Contents
Amendment Applications After and During Renewal	Qualifications of the Applicant (SL)	Quality Assurance Program (SL)	Consideration of DSS Transportability (CoC)
Chapter 2 – Site Characteristics Evaluation (SL)			
<ul style="list-style-type: none"> Geography and Demography Surface and Subsurface Hydrology 	<ul style="list-style-type: none"> Nearby Facilities Geology and Seismology 	<ul style="list-style-type: none"> Meteorology 	
Chapter 3 – Principal Design Criteria Evaluation			
<ul style="list-style-type: none"> Classification of SSCs Design Criteria for Safety Protection Systems 	<ul style="list-style-type: none"> Design Bases for SSCs Important to Safety Design Criteria for Other SSCs (SL) 		
Chapter 4 – Structural Evaluation			
<ul style="list-style-type: none"> Description of the SSCs Normal and Off-normal Conditions 	<ul style="list-style-type: none"> Design Criteria Accident Conditions 	<ul style="list-style-type: none"> Loads 	
Chapter 5 – Thermal Evaluation			
<ul style="list-style-type: none"> Decay Heat Removal System Analytical Methods, Models, and Calculations 	<ul style="list-style-type: none"> Material and Design Limits 	<ul style="list-style-type: none"> Thermal Loads and Environmental Conditions Surveillance Requirements 	
Chapter 6 – Shielding Evaluation			
<ul style="list-style-type: none"> Shielding Design Description Shielding Analyses 	<ul style="list-style-type: none"> Radiation Source Definition Reactor-Related GTCCWaste Storage (SL) 	<ul style="list-style-type: none"> Shielding Model Specification 	
Chapter 7 – Criticality Evaluation			
<ul style="list-style-type: none"> Criticality Design Criteria/Features Criticality Analysis 	<ul style="list-style-type: none"> Fuel Specification Burnup Credit 	<ul style="list-style-type: none"> Model Specification Reactor-Related GTCCWaste and HLW (SL) 	
Chapter 8 – Materials Evaluation			
<ul style="list-style-type: none"> General Review Considerations Fuel Cladding Integrity and Retrievability 	<ul style="list-style-type: none"> Material Properties 	<ul style="list-style-type: none"> Environmental Degradation; Chemical and Other Reactions Code Use and Quality Standards 	
Chapter 9 – Confinement Evaluation			
<ul style="list-style-type: none"> Confinement Design Characteristics Nuclides with Potential for Release 	<ul style="list-style-type: none"> Confinement Analyses 	<ul style="list-style-type: none"> Confinement Monitoring Capability Supplemental Information 	
Chapter 10A (SL)/10B (CoC) – Radiation Protection Evaluation			
<ul style="list-style-type: none"> ALARA 	<ul style="list-style-type: none"> Design Features Radiation Exposures 	<ul style="list-style-type: none"> Dose Assessment Health Physics Program (SL) 	
Chapter 11 – Operation Procedures and Systems Evaluation			
<ul style="list-style-type: none"> Operation Description Storage Container Handling and Storage Operations Other Operating Systems (SL) Analytical Sampling (SL) 	<ul style="list-style-type: none"> Storage Container Loading Repair and Maintenance (SL) Operation Support Systems (SL) Fire and Explosion Protection (SL) 	<ul style="list-style-type: none"> Storage Container Unloading Control Room and Control Area (SL) 	
Chapter 12 – Conduct of Operations Evaluation			
<ul style="list-style-type: none"> Organizational Structure (SL) Normal Operations (SL) 	<ul style="list-style-type: none"> Acceptance Tests Personnel Selection (SL) 	<ul style="list-style-type: none"> Maintenance Program Emergency Planning (SL) Preoperational Testing and Startup (SL) Physical Security/Safeguards (SL) 	
Chapter 13 – Waste Management Evaluation (SL)			
<ul style="list-style-type: none"> Waste Sources and Facilities Solid Wastes 	<ul style="list-style-type: none"> Off-Gas Treatment and Ventilation Waste Stream Radiological Characteristics and Dose Analyses 	<ul style="list-style-type: none"> Liquid Waste Treatment/Retention 	
Chapter 14 – Decommissioning Evaluation (SL)			
<ul style="list-style-type: none"> Proposed Decommissioning Plan Operational Features 	<ul style="list-style-type: none"> Design Features Decommissioning Funding Plan 		
Chapter 15 – Quality Assurance Evaluation			
<ul style="list-style-type: none"> Organization and Program Document Control 	<ul style="list-style-type: none"> Design and Nonconformance Procurement and Test Control 	<ul style="list-style-type: none"> Procedures and Drawings Inspections and Audits 	
Chapter 16 – Accident Analysis Evaluation			
<ul style="list-style-type: none"> Cause of Event Detection of Event 	<ul style="list-style-type: none"> Definition of Operating Environment and Physical Parameters Summary of Event Consequences and Regulatory Compliance 	<ul style="list-style-type: none"> Corrective Course of Action 	
Chapter 17 – Technical Specifications Evaluation			
<ul style="list-style-type: none"> Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings Design Features 	<ul style="list-style-type: none"> Surveillance Requirements 	<ul style="list-style-type: none"> Limiting Conditions Administrative Controls 	

Figure 5.1. Overview of safety evaluation of a DSS (Taken from NUREG-2215).

As burnups have progressed and new data have become available, NRC staff published Interim Staff Guidance 11 (ISG-11) regarding cladding considerations for the transportation and storage of spent fuel. This information is used to supplement the guidance in the standard review plan. The standard review plan NUREG-1536, Rev. 1 (US Nuclear Regulatory Commission 2010) was revised in 2010 to reflect the latest guidance in ISG-11, Rev. 3. NUREG-1567 (US Nuclear Regulatory Commission 2000) was replaced by NUREG-2215 (US Nuclear Regulatory Commission 2020a) and incorporates ISG-11, Rev. 3.

ISG-11 has been periodically revised as more data have become available. The following summary lists the main guidance provided in each revision to ISG-11:

ISG-11, Rev.0

- Supplement the standard review plan by addressing potential degradation of high burnup fuel (> 45 GWd/MTU).

ISG-11, Rev. 1

- Incorporate new data
- Give applicant responsibility for demonstrating that the cladding is adequately protected
- Cladding oxidation should not be credited as load-bearing in the fuel cladding structural evaluation
- Defined a 1% creep strain limit on the cladding
- Only accounted for Zircaloy-clad fuel rods (no advanced cladding).

ISG-11, Rev. 2

- Changed the definition of damaged fuel
- Removed the 1-percent creep strain limit
- Discuss criteria to limit hydride reorientation in the cladding
- Applicable to all zirconium-alloy claddings and all burnup levels
- Described calculations to determine the maximum cladding temperature per a justified creep strain limit.

ISG-11, Rev. 3

- Replaced calculation of maximum cladding temperature per a justified creep strain limit with a generic 400°C peak cladding temperature limit for normal conditions of storage (NCS) and normal conditions of transport (NCT).
- Allowed a higher short-term temperature limit for low-burnup fuel if it could be demonstrated that cladding hoop strain does not exceed 90 MPa.
- Generic maximum cladding temperature limit of 570°C for off-normal and accident conditions applicable to all burnups.
- Minimize hydride reorientation by restricting change in cladding temperature during drying operations to 65°C and the cladding should not experience more than 10 thermal cycles each not exceeding 65°C.

NRC staff has recently provided a draft NUREG (NUREG-2224) that is out for comment regarding dry storage and transportation of high burnup SNF (Ahn, et al. 2018). This document seems to consider fuel with burnup up to 65 GWd/MTU. Additionally, NRC staff recently presented their thoughts on the management of high burnup spent fuel (Torres 2018). With regard to storage of SNF, these documents discussed the roles of cladding creep and hydride reorientation. These documents also provided guidance for dry storage for less than 20 years and for dry storage for greater than 20 years.

Regarding operability and safety significance of aging mechanisms of the fuel cladding/assembly hardware on the performance of the fuel for dry storage periods up to 60 years, NRC has issued NUREG-2214 (US Nuclear Regulatory Commission 2019) that provides the technical basis for these issues.

5.2.2 Application to Cr-Coated Zr Cladding

In examining the safety evaluations that should be performed on a DSS shown in Figure 5.1, the items that are expected to be impacted by use of Cr-coated Zr cladding are:

- Structural Evaluation: Component Materials
- Thermal Evaluation: Spent Fuel Cladding, Component Materials
- Confinement Evaluation: Component Materials
- Shielding Evaluation: Component Materials
- Criticality Evaluation: Component Materials
- Materials Evaluation: Cladding Integrity

Based on these needs, the following information is needed to support the safety analysis of a DSS containing Cr-coated clad fuel:

- New cladding mechanical properties (yield stress, ultimate tensile strength, uniform elongation).

As previously discussed, the application of a thin Cr-coating to Zr-alloy cladding is unlikely to impact the mechanical properties or the nuclear properties of the irradiated cladding. Therefore, if data are provided to justify this, the current safety analysis of a DSS containing uncoated Zr-alloy clad fuel could be applied to Cr-coated Zr-alloy fuel.

Necessary updates to in-reactor codes and methods have been discussed elsewhere (Geelhood and Luscher 2019) and will be necessary to provide bounding conditions regarding the following conditions of spent fuel:

- Rod internal pressure (likely not significantly impacted by Cr coating)
- Oxide thickness (likely will be bounded by uncoated cladding)
- Hydrogen content (likely will be bounded by uncoated cladding)

To address the issues of cladding creep and hydride reorientation (embrittlement), the applicant should also justify the following limits that are articulated in ISG-11, Rev. 3 and have historically been used for storage of SNF:

- 400°C peak cladding temperature limit for NCS

- Maximum cladding temperature limit of 570°C for off-normal and accident conditions
- The cladding should not experience more than ten thermal cycles each not exceeding 65°C

Data that could be used to assess these needs are summarized in Section 5.4.

5.2.3 Application to FeCrAl Cladding

In examining the safety evaluations that should be performed on a DSS, shown in Figure 5.1, the items that are expected to be impacted by use of FeCrAl cladding are:

- Principal Design Criteria Evaluation: Spent Fuel Design Basis
- Structural Evaluation: Component Materials, Dimensions and Weights
- Thermal Evaluation: Spent Fuel Cladding, Component Materials, Decay Heat, Dimensions
- Confinement Evaluation: Component Materials, Dimensions
- Shielding Evaluation: Component Materials, Dimensions
- Criticality Evaluation: Component Materials, Dimensions
- Materials Evaluation: Cladding Integrity

Based on these needs, the following information on FeCrAl cladding is needed to support the safety analysis of a DSS containing FeCrAl clad fuel:

- New cladding mechanical properties (yield stress, ultimate tensile strength, uniform elongation)
- New cladding thermal properties (thermal conductivity, specific heat, thermal expansion)
- New cladding dimensions (FeCrAl cladding anticipated to be thinner than Zr-alloy cladding)
- FeCrAl activation following irradiation
- FeCrAl neutron cross sections

Necessary updates to in-reactor codes and methods have been discussed elsewhere (Goodson and Geelhood 2020) and will be necessary to provide bounding conditions regarding the following conditions of spent fuel:

- Rod internal pressure
- Oxide thickness (likely will be minimal)

To address the issues of cladding creep and embrittlement, the applicant should also justify the following limits that are articulated in ISG-11 Rev. 3 and have historically been used for storage of SNF:

- 400°C peak cladding temperature limit for NCS
- Maximum cladding temperature limit of 570°C for off-normal and accident conditions
- The cladding should not experience more than ten thermal cycles each not exceeding 65°C

Data that could be used to assess these needs are summarized in Section 5.4.

5.3 Transportation of Spent Nuclear Fuel

This section discusses the transportation of SNF, including the current regulatory framework and its application to near-term ATF concepts.

5.3.1 Current Regulatory Framework

The regulations in 10 CFR Part 71, “Packaging and Transportation of Radioactive Material” (US Nuclear Regulatory Commission 2015), include a number of fuel-specific and package-specific requirements that may be dependent of the design basis condition of the fuel cladding. 10 CFR 71.31, “Contents of Application,” and 10 CFR 71.33, “Package description,” require an application for a transportation package to describe the proposed package in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation of the package, which includes a description of the chemical and physical form of the allowable contents.

The condition of the SNF cladding is critical to the transportation as 10 CFR 71.55(d)(2) requires that the geometric form of the package contents is not substantially altered under the tests for NCT. 10 CFR 71.55(e) also requires that a package used for the shipment of fissile material is to be designed and constructed and its contents so limited that under the tests for hypothetical accident conditions specified in 10 CFR 71.73, “Hypothetical Accident Conditions,” the package remains subcritical. The requirement assumes that 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 as stated in 10 CFR 71.55(e)(1). Additional criticality requirements are given for package arrays in 10 CFR 71.59, “Standards for arrays of fissile material packages”.

In addition to these regulations, NRC staff have provided NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel” (US Nuclear Regulatory Commission 2000). These guidelines have recently been replaced by NUREG-2216 (US Nuclear Regulatory Commission 2020b). This SRP provides guidance in the preparation by the applicant and review by the staff of a topical report describing a transportation package for SNF. The standard review plan lays out the following evaluations that should be performed during the safety evaluation of a DSS, shown in Figure 5.2.

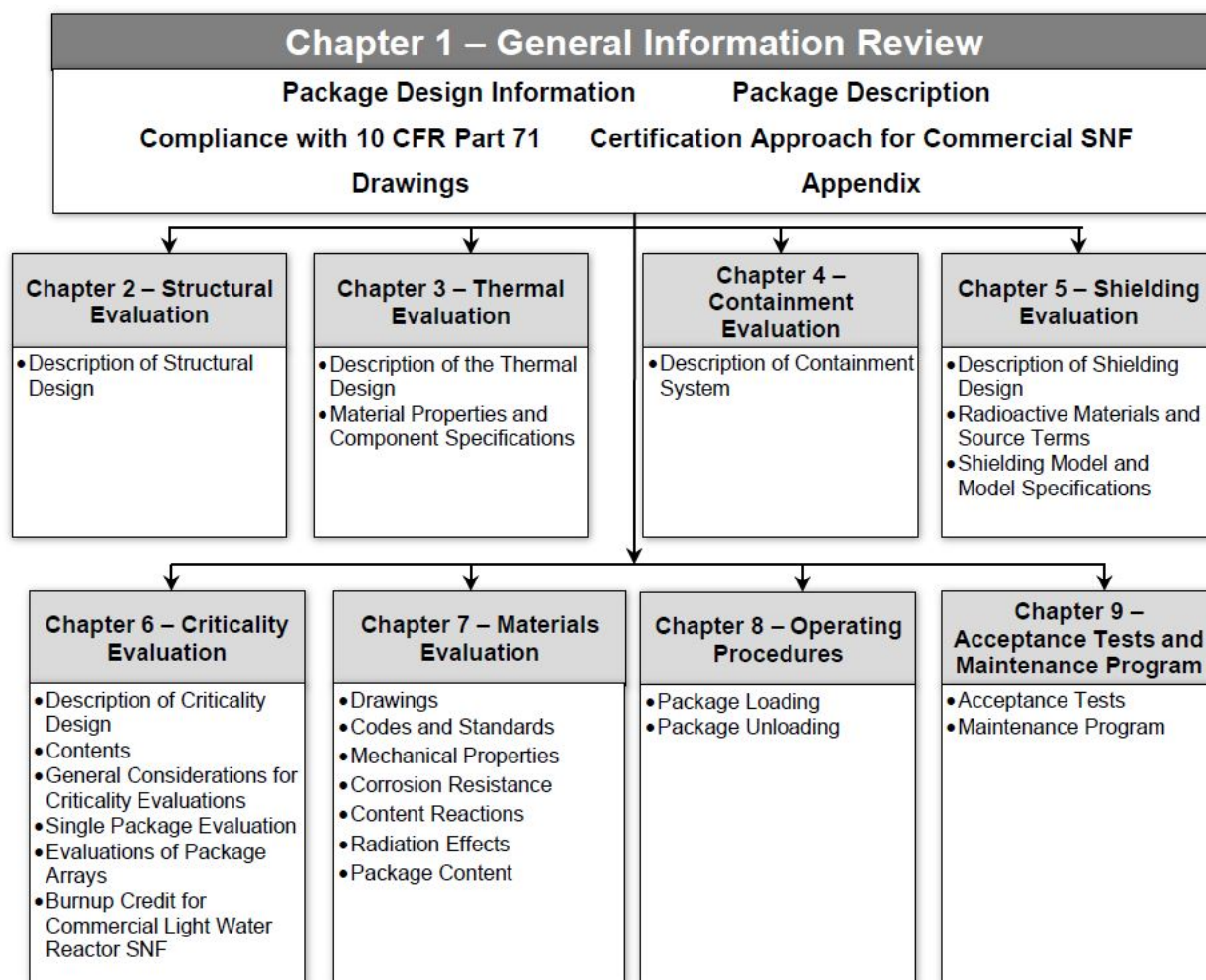


Figure 5.2. Overview of safety evaluation of SNF transportation (from NUREG-2216)

As burnups have increased and new data have become available, NRC staff published ISG-11 regarding cladding considerations for the transportation and storage of spent fuel. This information is used to supplement the guidance in the standard review plan. The standard review plan (NUREG-1617) has not been revised to reflect any of this guidance but was replaced by NUREG-2216 to incorporate ISG-11 rev. 3. ISG-11 has been periodically revised as more data have become available. The summary of revisions is provided in Section 5.2.1.

NRC staff has recently provided a draft NUREG that is out for comment (NUREG-2224; Ahn et al. 2018) regarding dry storage and transportation of high burnup SNF. This document seems to consider fuel with burnup up to 65 GWd/MTU. Additionally, NRC staff recently presented their thoughts on the management of high burnup spent fuel (Torres 2018). Regarding transportation of SNF, these documents discussed the roles of cladding fatigue lifetime and hydride reorientation. These documents also provided guidance for transportation of SNF that has been in dry storage for less than 20 years and for transportation of SNF that has been in dry storage for greater than 20 years.

5.3.2 Application to Cr-Coated Zr Cladding

In examining the safety evaluations that should be performed on a transportation package for SNF shown in Figure 5.2, the items that are expected to be impacted by use of Cr-coated Zr cladding are:

- Structural Evaluation: Component Materials
- Thermal Evaluation: Component Materials
- Confinement Evaluation: Component Materials
- Shielding Evaluation: Component Materials
- Criticality Evaluation: Component Materials

Based on these needs, the following information on cladding is needed to support the safety analysis of a transportation package for SNF Cr-coated Zr clad fuel:

- New cladding mechanical properties (yield stress, ultimate tensile strength, uniform elongation)
- Cladding fatigue lifetime

As previously discussed, the application of a thin Cr-coating to Zr-alloy cladding is unlikely to impact the mechanical properties or the nuclear properties of the irradiated cladding. Therefore, if data are provided to justify this, the current safety analysis of a transportation package containing uncoated Zr-alloy clad fuel could be applied to Cr-coated Zr-alloy fuel.

Necessary updates to in-reactor codes and methods have been discussed elsewhere (Geelhood and Luscher 2019) and will be necessary to provide bounding conditions regarding the following conditions of spent fuel.

- Rod internal pressure (likely not significantly impacted by Cr coating)
- Oxide thickness (likely will be bounded by uncoated cladding)
- Hydrogen content (likely will be bounded by uncoated cladding)

To address the issues of cladding creep and hydride reorientation (embrittlement), the applicant should also justify the following limits that are articulated in ISG-11, Rev. 3 and have historically been used for storage of SNF:

- 400°C peak cladding temperature limit for NCT
- Maximum cladding temperature limit of 570°C for off-normal and accident conditions
- The cladding should not experience more than ten thermal cycles each not exceeding 65°C.

Data that could be used to assess these needs are summarized in Section 5.4

5.3.3 Application to FeCrAl Cladding

In examining the safety evaluations that should be performed on a transportation package for SNF, shown in Figure 5.2, the items that are expected to be impacted by use of FeCrAl cladding are:

- Structural Evaluation: Component Materials, Dimensions, and Weights
- Thermal Evaluation: Component Materials, Dimensions
- Confinement Evaluation: Component Materials, Dimensions
- Shielding Evaluation: Component Materials, Dimensions
- Criticality Evaluation: Component Materials, Dimensions

Based on these needs, the following information on cladding is needed to support the safety analysis of a transportation package for SNF FeCrAl clad fuel:

- New cladding mechanical properties (yield stress, ultimate tensile strength, uniform elongation)
- New cladding thermal properties (thermal conductivity, specific heat, thermal expansion)
- New cladding dimensions (FeCrAl cladding anticipated to be thinner than Zr-alloy cladding)
- Cladding fatigue lifetime
- FeCrAl activation following irradiation
- FeCrAl neutron cross sections

Necessary updates to in-reactor codes and methods have been discussed elsewhere (Goodson and Geelhood 2020) and will be necessary to provide bounding conditions regarding the following conditions of spent fuel.

- Rod internal pressure
- Oxide thickness (likely will be minimal)

To address the issues of cladding creep and embrittlement, the applicant should also justify the following limits that are articulated in ISG-11, Rev. 3 and have historically been used for storage of SNF:

- 400°C peak cladding temperature limit for NCT
- Maximum cladding temperature limit of 570°C for off-normal and accident conditions
- The cladding should not experience more than ten thermal cycles each not exceeding 65°C

Data that could be used to assess these needs are summarized in Section 5.4.

5.4 Data Recommendation for Safety Evaluations

Based on the assessments in the previous sections, the information in Table 5.1 is needed to support safety analyses of a DSS and an SNF transportation package containing fuel with Cr-Coated Zr cladding or FeCrAl cladding. Also shown in this table are the recommended sources of information for each of these items.

Some data necessary to support this information may already be available. Section 6.0 discusses data that is currently available that could support the safety analysis of Cr-coated Zr and FeCrAl clad fuel for SNF storage and transportation.

Table 5.1. Assessment data that could be used to justify the safety evaluation of a DSS and a SNF transportation package containing fuel with Cr-coated Zr cladding and FeCrAl cladding.

Supporting Information	Recommended Source
Cr-coated Zr Cladding	
Mechanical properties (yield stress, ultimate tensile strength, uniform elongation)	Mechanical property tests performed on cladding segments irradiated to target burnup. Note: Mechanical properties used to support in-reactor analysis are typically performed at room temperature and reactor operating temperature (300°C to 350°C). These data should be collected at relevant temperature to storage and transportation.
Separate effects tests to identify phenomena that can lead to gross cladding rupture	Failure limits at high burnup should be confirmed for creep strain capacity and delayed hydride cracking.
Fatigue life	Fatigue tests performed on cladding segments that contain fuel or have been de-fueled irradiated to target burnup.
Justification for peak cladding temperature limits regarding hydride reorientation: <ul style="list-style-type: none"> • 400°C peak cladding temperature limit for NCS and NCT • Maximum cladding temperature limit of 570°C for off-normal and accident conditions • The cladding should not experience more than ten thermal cycles each not exceeding 65°C 	Hydride reorientation and strength tests on cladding segments irradiated to target burnup.
Limiting rod internal pressure, oxide thickness, and hydrogen content.	Thermal-mechanical code approved to target burnup for rod with limiting power history.
FeCrAl Cladding	
Mechanical properties (yield stress, ultimate tensile strength, uniform elongation)	Mechanical property tests performed on cladding segments irradiated to target burnup. Note: Mechanical properties used to support in-reactor analysis are typically performed at room temperature and reactor operating temperature (300°C to 350°C). These data should be collected at relevant temperature to storage and transportation.
Separate effects tests to identify phenomena that can lead to gross cladding rupture	Failure limits at high burnup should be confirmed for creep strain capacity and delayed hydride cracking.
Thermal properties (thermal conductivity, specific heat, thermal expansion)	Thermal property tests performed on cladding segments irradiated to target burnup. Note: Thermal properties used to support in-reactor analysis are typically performed at room temperature and reactor operating temperature (300°C to 350°C). These data should be collected at relevant temperature to storage and transportation.
Fatigue life	Fatigue tests performed on cladding segments that contain fuel or have been de-fueled irradiated to target burnup.
Justification for peak cladding temperature limits regarding hydride reorientation: <ul style="list-style-type: none"> • 400°C peak cladding temperature limit for NCS and NCT • Maximum cladding temperature limit of 570°C for off-normal and accident conditions • The cladding should not experience more than ten thermal cycles each not exceeding 65°C 	Ductility and strength tests on cladding segments irradiated to target burnup.
Limiting rod internal pressure and oxide thickness	Thermal-mechanical code approved to target burnup for rod with limiting power history.
Cladding cross section and activation	Code prediction from code such as ORIGEN.

6.0 Available Data

This section describes the data that are currently available on Cr-coated Zr cladding and FeCrAl data in areas that have been identified for analysis of SNF storage and transportation. The presence of data in any area does not indicate that an applicant would not have to provide data from their specific coated cladding because it has been observed that coating processes and other processes can impact the performance of the cladding and the coatings. Rather, these data are compiled here to give the NRC staff expected performance of coated cladding, as well as areas of concern that should be given additional scrutiny, during the review of one of these concepts.

6.1 Cr-Coated Zr Cladding

The key data needed for analysis of SNF with Cr-coated Zr cladding during storage and transportation are cladding mechanical properties and cladding fatigue. The current available data for Cr-coated Zr cladding in these areas are discussed in the following sections.

6.1.1 Cladding Mechanical Properties

No data showing irradiated mechanical properties of Cr-coated Zr cladding has been found (elastic modulus, yield stress, and uniform elongation). Several sources of unirradiated mechanical properties of Cr-coated Zr have been found. The data are summarized in Table 6.1. In general, the results indicate that, in unirradiated conditions, the mechanical properties at room temperature and normal operating conditions are effectively not impacted by the application of a coating. The data also show that, in the unirradiated conditions, the coating can survive without crack significantly beyond 1% hoop strain. Irradiation causes a dramatic increase in strength and decrease in ductility in Zr-alloys (Geelhood, Beyer and Luscher 2008). For this reason, the impact of the Cr-coatings on the mechanical properties (elastic modulus, yield stress, and ductility) will need to be quantified with irradiated cladding data.

Table 6.1. Summary of unirradiated mechanical properties data for Cr-coated Zr cladding

Organization	Cladding	Test Description	Results
Framatome (Brachet, et al. 2017)	Cr-coated M5 [®] by PVD	Tensile tests at room temperature and 400°C	Elastic modulus, yield stress, ultimate tensile strength and uniform elongation similar for coated and uncoated cladding.
		Thermal creep at 400°C for 240 hours	Thermal creep similar for coated and uncoated cladding.
KAERI (Kim, et al. 2015)	Cr-coated Zry-4 by 3D laser coating	Ring tensile and ring compression tests	Elastic modulus, yield stress, ultimate tensile strength and uniform elongation similar for coated and uncoated cladding.
			No cracking observed at 2% or 4% hoop strain. Cracking observed at 6% strain.
MIT (Shahin, et al. 2018)	Cr-coated Zry-4 by cold spray	Burst test at room temperature	Ultimate tensile strength, burst pressure, and burst strain about the same for coated and uncoated cladding.

6.1.2 Cladding Fatigue

There is little fatigue data for Cr-coated Zr. The data that do exist are for unirradiated Cr-coated Zr. The data are summarized in Table 6.2. For Zr-alloy cladding, the fatigue life has been shown to decrease with irradiation (O'Donnell and Langer 1964). The available data indicate that fatigue failure occurs significantly earlier in Cr-coated samples than in uncoated samples. The authors do note that this contrasts with previous data (Cavaliere and Silvello 2016). This indicates that process parameters and microstructure could have a profound impact on fatigue life. It has been noted (Kvedaras, et al. 2006) that in steels, Cr coating can improve or significantly worsen the fatigue lifetime due to different microstructures produced in the coating.

These data indicate a critical need for an applicant to provide fatigue data from irradiated cladding that they have manufactured to support their safety analysis limits.

Table 6.2. Summary of unirradiated fatigue data for Cr-coated Zr cladding.

Organization	Cladding	Test Description	Results
MIT (Sevecek, et al. 2018)	Cr coated Zry by cold spray	Fatigue cycling in air and in water between 300°C and 312°C	Fatigue failure observed significantly earlier in Cr coated samples (~10,000 cycles) than uncoated samples (100,000 to 500,000) cycles

6.2 FeCrAl Cladding

The key data needed for analysis of SNF with FeCrAl cladding during storage and transportation are cladding mechanical properties, cladding thermal properties, and cladding fatigue. The current available data for FeCrAl cladding in these areas are discussed in the following sections.

6.2.1 Cladding Mechanical Properties

Mechanical properties of irradiated FeCrAl cladding have been studied and are summarized in Table 6.3. Irradiated rods should be investigated further as irradiation hardens the cladding and leads to significant increase in the yield stress and ultimate tensile strength. Fueled rods are preferable, as the in-reactor temperature and heat flux across the cladding can impact the competing creation and annealing of lattice defects that lead to this hardening and this temperature may be different for fueled and unfueled rods.

Table 6.3. Summary of mechanical property testing for irradiated FeCrAl cladding.

Lead	FeCrAl Alloy(s)	Test Description	Results
ORNL (Field et al. 2015)	Fe-10Cr-4.8Al Fe-12Cr-4.4Al Fe-15Cr-3.9Al Fe-18Cr-2.9Al	Tensile tests at room temperature (only one test per sample)	Room temperature engineering stress-strain curves
ORNL (Field et al. 2017)	F1C5AY Kanthal APMT	Tensile tests at room temperature and 320°C	Stress-strain curve, tensile response as a function of dose, 0.2% offset yield strength, ultimate tensile strength, uniform elongation, total elongation
ORNL (Chen et al. 2019)	C06M C36M	Vickers microhardness testing with 1 kg force and 15 s dwell time Transition fracture toughness testing	Master Curve transition temperature; reasonable linear correlation between the Master Curve fracture toughness transition temperature and Vickers microhardness
University of New Mexico (Zhang et al. 2020)	MA956	Tensile tests at room temperature	Yield stress, ultimate tensile strength, uniform elongation, total elongation; hardening and ductility reduction after irradiation were observed

Figure 6.1 shows the elastic modulus of Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; Kanthal 2019; Special Metals 2004).

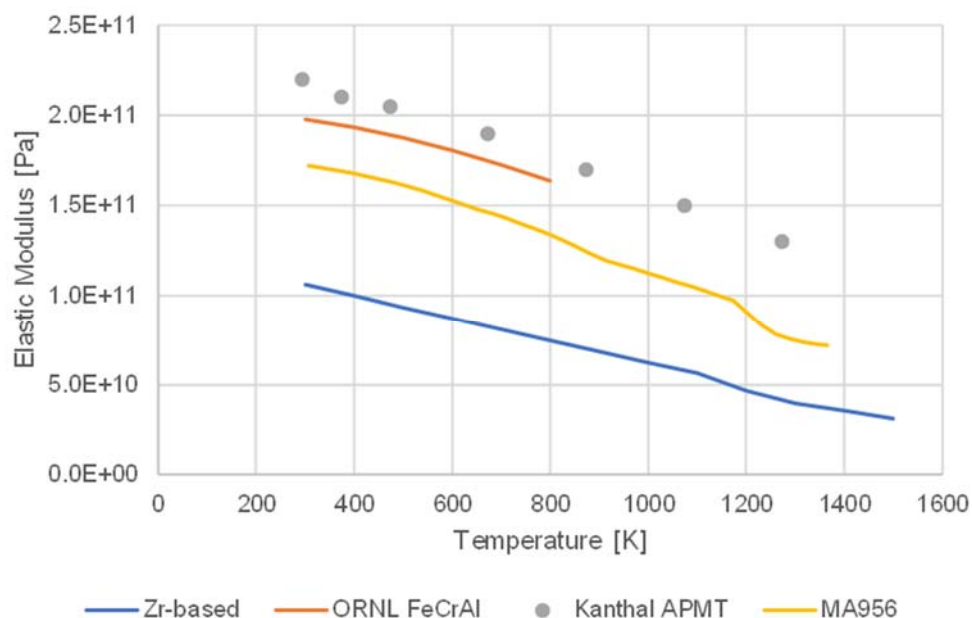


Figure 6.1. Elastic modulus of Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; Kanthal 2019; Special Metals 2004).

The FeCrAl data were obtained from unirradiated samples. Currently no data from irradiated materials exist.

Figure 6.2 shows the yield stress of unirradiated Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys: C25M and Kanthal APMT (Field 2018), MA956 (Special Metals 2004), and C26M (Yamamoto et al. 2019).

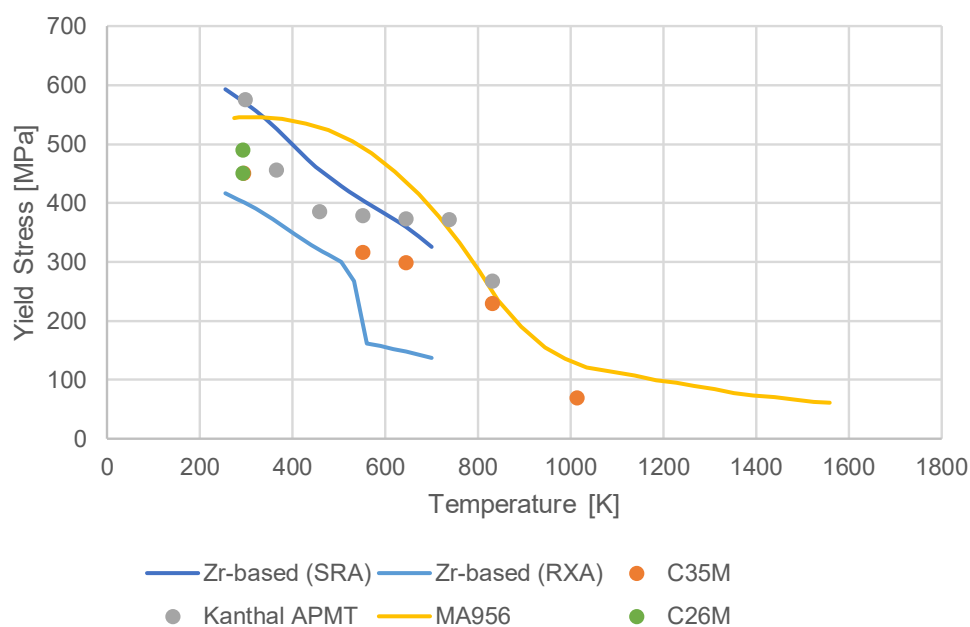


Figure 6.2. Unirradiated yield stress for Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; (Special Metals Corporation 2004); (Yamamoto, Kane, et al. 2019)).

Given the scatter in FeCrAl yield stress, alloy- and temperature-dependent yield stress data is necessary for mechanical data. Figure 6.3 shows the 320°C yield stress for Kanthal APMT after neutron irradiation (Field et al. 2017).

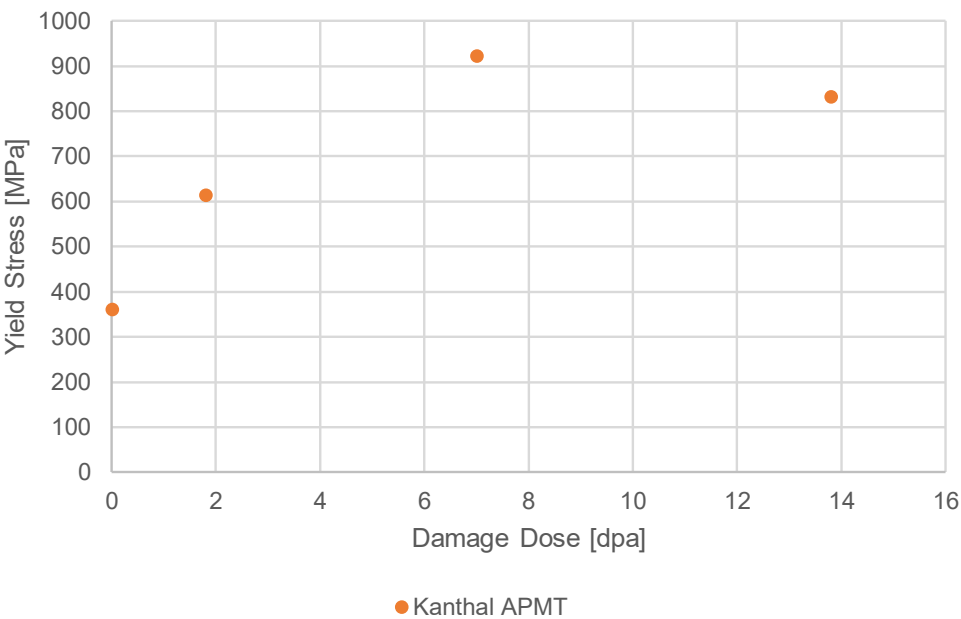


Figure 6.3. Irradiated yield stress for Kanthal APMT at 320 °C (Field et al. 2017).

The yield stress of Kanthal APMT increases up to ~7 displacements per atom (dpa) and then saturation of the hardening occurs above this dose. Lower Cr-content variants of FeCrAl are less susceptible to brittle fracture above 7 dpa when irradiated at near LWR-relevant temperatures. Lower Cr-content variants maintain adequate mechanical performance in the context of tensile properties after neutron irradiation for ATF LWR cladding applications when compared to Zr-based alloys (Field et al. 2017).

6.2.2 Cladding Thermal Properties

Limited studies on the thermal properties of non-commercial FeCrAl alloys, especially lean-Cr content, have been completed. Table 6.4 summarizes these data.

Table 6.4. Summary of unirradiated thermal property testing for FeCrAl cladding.

Lead	FeCrAl Alloy(s)s	Test Description	Results
ORNL (Field 2018)	Kanthal APMT	Differential scanning calorimetry, laser flash testing, dilatometry	Specific heat capacity, thermal diffusivity, thermal expansion
	C06M		
	C35M		
	C36M		
ORNL (Yamamoto et al. 2019)	C26M	Dilatometry, differential scanning calorimetry, and laser flash testing	Thermal expansion, heat capacity, and thermal diffusivity

Figure 6.4 shows the thermal conductivity of Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; Special Metals 2004). This plot does not include C26M, but recent thermal diffusivity data (Yamamoto et al. 2019), used to determine thermal conductivity, indicates that C26M will have similar thermal conductivity to these other alloys.

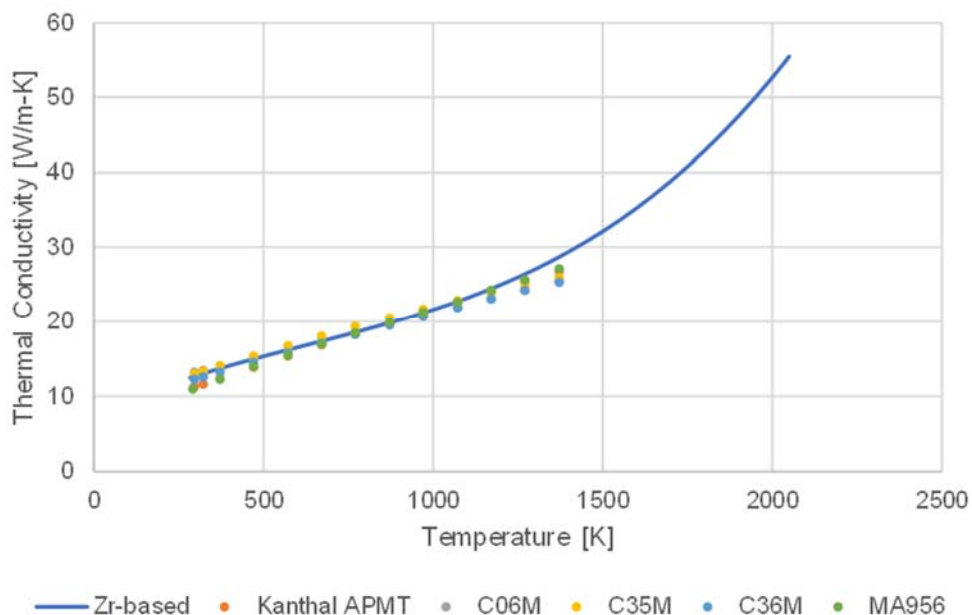


Figure 6.4. Thermal conductivity of Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; Special Metals 2004).

The FeCrAl data (Field 2018) were collected from unirradiated samples and fit to a curve. Currently there are no data from irradiated samples.

Figure 6.5 shows the thermal expansion of Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; Yamamoto et al. 2019; Special Metals 2004). Zr-based alloy tubes are processed in such a way that the tubes exhibit a large degree of microstructural texture, which results in different thermal expansion in different directions (i.e., axial and circumferential). Recent data from C26M is included (Yamamoto et al. 2019) and shows good agreement with other FeCrAl alloys.

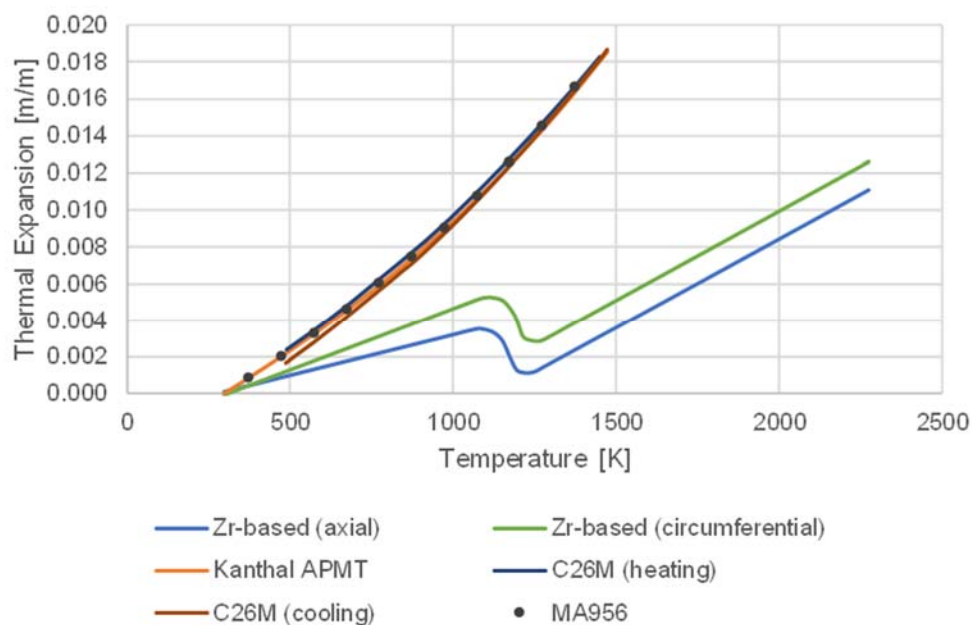


Figure 6.5. Thermal expansion of Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; Yamamoto et al. 2019; Special Metals).

C26M exhibits some differences in heating and cooling but the magnitude of this difference is not large. At lower temperatures (<1000 K), variation in the thermal expansion coefficient of FeCrAl alloys can be observed with composition (Field 2018). The FeCrAl data were collected from unirradiated samples; Kanthal APMT data were fit to a curve, but the C26M and MA956 data reported here are from direct measurements of thermal diffusivity.

Figure 6.6 shows the specific heat of Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; Yamamoto et al. 2019; Special Metals 2004). Recent data from C26M is included (Yamamoto et al. 2019) and shows good agreement with other FeCrAl alloys.

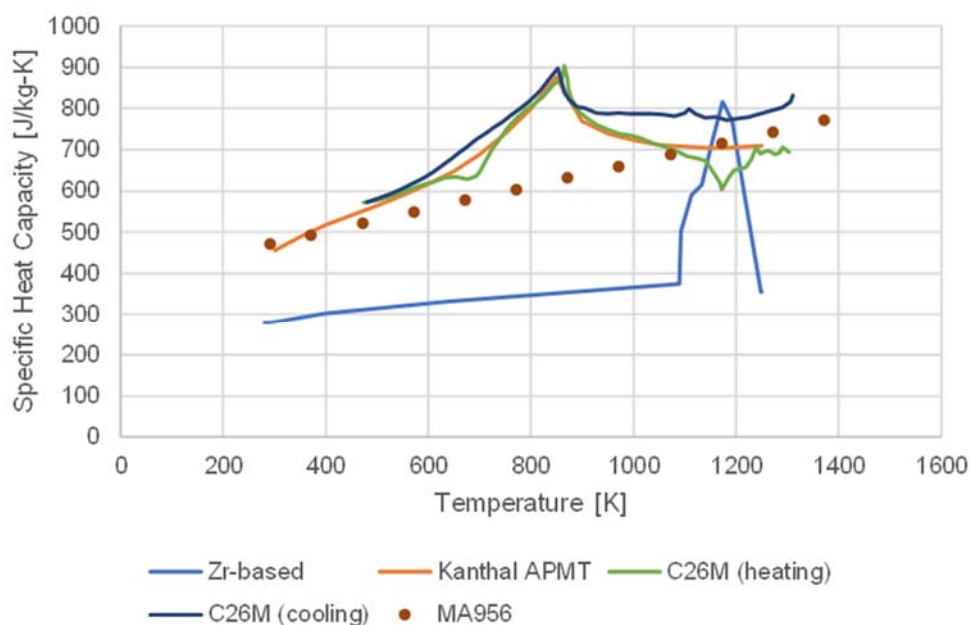


Figure 6.6. Specific heat of Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; Yamamoto et al. 2019; Special Metals).

C26M exhibits some difference between heating and cooling but both peaked around 850 K. The large peaks for C26M and Kanthal APMT correspond to the second order phase transitions from the materials' ferromagnetic to paramagnetic states. The FeCrAl data were collected from unirradiated samples; Kanthal APMT data were fit to a curve but the C26M and MA956 data reported here are from direct measurements.

6.2.3 Cladding Fatigue

Cladding fatigue is necessary to evaluate the impact of vibration during NCT on FeCrAl cladding. The existing fatigue data are for unirradiated FeCrAl and are summarized in Table 6.5. Fatigue data from irradiated cladding must be provided to support safety analysis limits.

Table 6.5. Summary of fatigue data for unirradiated FeCrAl cladding.

Lead	FeCrAl Alloy(s)	Test Description	Results
City University of Hong Kong (Field 2018)	Fe-23.85Cr-3.89Al De-25Cr-2Al	Various strain amplitudes at various temperatures	Three-stage behavior: 1) hardening, 2) saturation, and 3) softening followed by fracture, showing a dependence on temperature and strain amplitude Indicated a potential composition dependency

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 and Langer 1964). Figure 6.7 shows the typical unirradiated Zircaloy fatigue design curve as well as some fatigue data from a particular FeCrAl alloy (Field, et al. 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 2 on stress amplitude or a safety factor of 20 on the number of cycles.

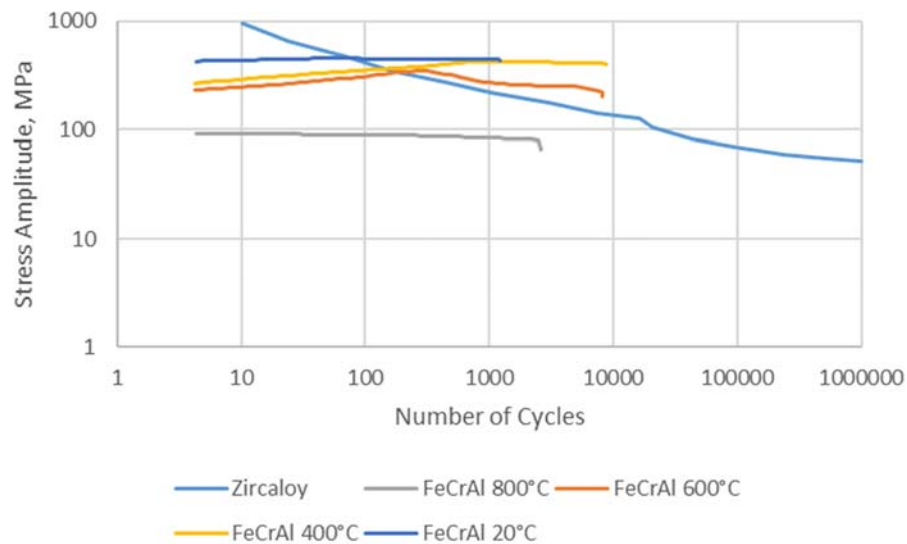


Figure 6.7. Fatigue lifetime curve for unirradiated Zircaloy and fatigue data from FeCrAl (Fe-23.85Cr-3.89Al).

7.0 Conclusions

This report provides an assessment of the NRC regulatory structure for storage and transportation of SNF as it relates to near term ATF concepts, specifically Cr-coated Zr alloy and FeCrAl cladding. An overview of recent ATF development on these concepts is provided both with regard to what various organizations have recently published and general metallurgical background on these concepts including relevant cladding degradation and failure modes.

PNNL concludes that the current regulatory framework for SNF storage and transportation of fuel with these ATF claddings is generally applicable. However, applicants should provide data to justify the use of empirical temperature limits set in interim staff guidance and standard review plans if they wish to proceed with these limits.

Using the established framework, areas where cladding specific data will be necessary for each concept were identified.

For Cr-coated Zr cladding, the following data are needed:

- Cladding mechanical properties (yield stress, ultimate tensile strength, uniform elongation)
- Separate effects tests to identify phenomena that can lead to gross cladding rupture
- Fatigue life
- Justification for peak cladding temperature limits regarding hydride reorientation

For FeCrAl the following data are needed:

- Mechanical properties (yield stress, ultimate tensile strength, uniform elongation)
- Separate effects tests to identify phenomena that can lead to gross cladding rupture
- Thermal properties (thermal conductivity, specific heat, thermal expansion)
- Fatigue life
- Justification for peak cladding temperature limits regarding hydride reorientation
- Cladding cross section and activation

Additionally, for each concept, approved reactor codes should be used to provide limiting rod internal pressure and oxide thickness.

Finally, currently available data that are relevant to SNF storage and transport were identified for each cladding type. For both concepts there is limited representative irradiated data and it is expected that the applicants will provide additional data that is representative of the specific cladding they provide.

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Pacific Northwest National Laboratory

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

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