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Spent Fuel Storage and Transportation of Accident Tolerant Fuel Concepts

Cr-Coated Zirconium Alloy Cladding, FeCrAl Cladding, High Burnup and High Enrichment Fuel

October 2024

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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 two ATF cladding concepts being investigated to replace the zirconium-based alloys currently used for fuel cladding and provides current state-of-the-industry information on material properties and fuel performance considerations relevant to the storage and transportation of spent nuclear fuel (SNF). For implementation of the near-term ATF concepts, fuel vendors and power reactor licensees are exploring the possibility of increasing the maximum enrichment of fuel up to 10% and the burnup limit to 75 or 80 GWd/MTU.

Currently, three U.S. nuclear fuel market suppliers are developing ATF designs: Global Nuclear Fuels (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 (Cr_2O_3+Al_2O_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 these ATF cladding concepts, including fuel performance characteristics that may not be addressed within existing regulatory documents. Furthermore, the implications of high burnup (> 62 GWd/MTU) and enhanced enrichment (> 5 w/o ^{235}U) on the storage and transportation of SNF are also discussed. The recommendations made in this report are based on current publicly available data.

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
EDF	Électricité de France
EPRI	Electric Power Research Institute
FFRD	fuel fragmentation relocation and dispersal
GDC	General Design Criteria
GNF	Global Nuclear Fuels
HFIR	High Flux Isotope Reactor
INL	Idaho National Laboratory
ISFSI	Independent Spent Fuel Storage Installation
ISG	Interim Staff Guidance
KAERI	Korea Atomic Energy Research Institute
KEPCO	Korea Electric Power Corporation
LOCA	loss-of-coolant accident
LTA	lead test assembly
LTR	lead test rod
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
w/o	weight percent
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 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 (e.g., robust emergency core cooling system designs, protected or redundant back-up power generation capabilities etc.) 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 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 zirconium alloy 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. The implications of higher burnup and enhanced enrichment must also be considered when evaluating operational expectations of these ATF concepts relative to current fuel technologies.

As most of the NRC’s regulatory framework was developed for the zirconium alloy-cladding, 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 on two ATF cladding concepts. The first cladding concept is a Cr-coated zirconium alloy, generally referred to as “Cr-coated Zr-alloy cladding” and is closest to a license request. This concept includes both metallic and ceramic coatings of chromium on zirconium cladding and is considered a “near-term ATF concept”. The second cladding concept is an iron-chromium-aluminum alloy, generally referred to as “FeCrAl cladding”. The term FeCrAl is applied to a variety of alloys with a broad range of compositions, which can significantly impact the material properties of the cladding. Even though C26M has been the focus of most studies in this category, there is no formal consensus on the composition of FeCrAl cladding. To support the agency’s efforts, this report will identify and discuss degradation and failure modes of Cr-coated and FeCrAl claddings, including fuel performance characteristics that may not be addressed within existing regulatory documents (e.g., regulations, regulatory guidance, NUREG-2215), with regard to spent nuclear fuel (SNF) storage and transportation.

This report also provides an overview of ATF cladding development and an assessment of the applicability of current regulations and guidance for SNF storage and transportation to these ATF concepts. The recommendations made in this report are based on current publicly available data. Additional considerations are provided for the implications of high burnup (e.g., > 62 GWd/MTU) and increased enrichment (i.e., > 5%) as well. Based on this, an assessment of the critical data needs for spent fuel storage and transportation for each ATF cladding concept is made. A recent literature review was performed and captures the current status of the required data. This revision provides an update to the literature review from 2020 to 2023, addresses changes recommended during the expert panel review and documents the phenomena identification and ranking table (PIRT) exercise that was performed.

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 (e.g., Ziry-2 & Ziry-4). 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 (BDBA); 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 for current Zr-based alloy claddings containing UO_2 fuel.

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.

The fuel rods are not actively monitored in the spent fuel pool, but degradation of the fuel, including damaged fuel, is not expected under wet storage conditions (IAEA-TECDOC-1012).

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

[1] ZIRLO® and Optimized ZIRLO™ are registered trademarks of Westinghouse Electric Company LLC.
[2] M5® is a registered trademark of Framatome.

fuel rods to heat up due to the lack of convective cooling 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 experience less 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) (U.S. Nuclear Regulatory Commission 2020a; U.S. Nuclear Regulatory Commission 2020b).

Following the vacuum drying, fuel assemblies are placed in DSS that are passively cooled and backfilled 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 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 (U.S. Nuclear Regulatory Commission 2020a; U.S. 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

Some DSS currently in use are dual-use storage and transportation casks typically referred to as a bolted casks. Most utilities place spent fuel in a sealed canister that is initially placed in ventilated concrete storage cask systems intended to be later transloaded with the use of a transfer cask and shipped inside of a steel transport cask. 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 (U.S. Nuclear Regulatory Commission 2020b).

1.2 Previous Reviews

Four 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) has 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 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 μm)
- Similar mechanical behavior as uncoated cladding if coating is sufficiently thin (<20 μm)
- 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 reduced 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 agrees with these conclusions, and notes that adherence of coatings applied by different processes should also be monitored. 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.

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
- 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 Cr-coated zirconium and FeCrAl cladding concepts. 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 and 2.8 summarize concepts being developed and tested at DOE laboratories. Although the concepts being developed outside the U.S. may not have a planned 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 simultaneously developing both near and long term ATF designs. In the near-term, Westinghouse is working toward commercializing chromium-coated zirconium-alloy cladding with chromium oxide-aluminum oxide (Cr_2O_3+Al_2O_3)- doped UO_2 (ADOPT™) fuel as well as chromium-coated zirconium-alloy cladding with uranium silicide (U_3Si_2) fuel (Karoutas 2019). 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). The above concepts are collectively 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[3] in April 2019 (Westinghouse Nuclear 2019; Avali and Lahoda 2018) with a plan for licensing regional 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 were planned for irradiation in 2022 (Avali and Lahoda 2018).

Following the insertion of the 16 Cr-Coated LTRs in Byron in 2019 (16 LTRs between 2 assemblies), 32 LTRs consisting of Cr-coated Optimized ZIRLO clad UO_2 were inserted among 4 assemblies in Doel[4] in June 2020. Poolside and non-destructive examinations of the Doel LTRs in October 2021 revealed trends consistent with those observed for the Byron rods in 2020. No coating delamination or degradation was observed on these LTRs and they were virtually free of CRUD deposits after one cycle of irradiation.

After the first irradiation cycle at Byron, 7 LTRs were extracted and shipped to ORNL for PIE in 2021. These rods consisted of 3 ATF rods (Cr-coated ZIRLO with ADOPT Fuel) and 4 High Burnup rods (AXIOM). Planned hot cell examinations included both non-destructive (i.e., visual

[1] EnCore® is a registered trademark of Westinghouse Electric Company LLC.
[2] ADOPT™ is a registered trademark of Westinghouse Electric Company LLC.
[3] Illinois PWR
[4] Belgium PWR

inspection, gamma scanning, profilometry) and destructive (i.e., fission gas measurement, metallography on one rod, and hydrogen content) examinations. Although some initial results are available, as of October 2022 all hot cell results were considered preliminary, under review, and subject to change. Following hot cell examinations, the tentative plan was to ship portions of these LTRs to INL for power ramp testing in ATR and TREAT.

Based on results from visual inspection, the 4 assemblies containing these 32 LTRs were approved for reload in a 2nd cycle at Doel, which was followed by another inspection campaign in spring 2023 (Fallot 2022). The LTRs remaining at Byron were returned for a second irradiation cycle, which was completed in May 2022. Following a second cycle in Doel and Byron, the ATF rods remained in pristine condition with excellent coating adherence and little indication of CRUD.

As of October 2022, a License Amendment Request (LAR) was planned for submittal to pursue a third cycle at Byron in 2023 to achieve 75 MWd/kgU (Olson 2022). This request was approved, and rods were inserted in fall of 2023.

Future plans include inserting LTRs in Vogtle Unit 2[1] and obtaining data from high enrichment, high burnup ATF fuel concepts. The Vogtle LTRs were approved for insertion in 2023 and consist of 6% ^{235}U ADOPT™ pellets in Cr-coated cladding. Insertion of these rods is planned for spring of 2025. Results from these LTRs are intended to support high enrichment, high burnup (HEHB) topical reports. Expansion of ADOPT™ fuel in Cr-coated cladding burnup to 75 MWd/kgU also awaits results from the SCIP IV program at Studsvik, which was planned to begin in 2023. This program will perform PIEs and FFRD testing on LTRs irradiated at KKL[2], which were expected to arrive in Sweden in 2023 (Karoutas 2022).

Westinghouse is not currently 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.

In a recent summary of their irradiation experiments, Framatome described how the initial coated cladding irradiation in OSIRIS (2015) provided justification for cladding irradiation in Gosgen[3] as part of the IMAGO program (2016) and, eventually, for fueled rodlets to be irradiated in ATR (2018). This paved the way for LTRs in Vogtle, Gosgen (GOCHROM), and ANO[4] in 2019. As of 2022, an LTA irradiation was underway at Calvert Cliffs[5] (starting April 2021) and experimental irradiations at the High Fux Isotope Reactor (HFIR) were being conducted to project clad behavior at high fluence (Vioujard 2022, Nimishakavi 2022).

The ATF-2 experiment at ATR subjected Cr-coated M5 fuel pins (also referred to as rodlets) to irradiation in a pressurized water loop. Irradiation began in 2018 and 4 pins were discharged in

[1] Georgia PWR
[2] Switzerland BWR
[3] Switzerland PWR
[4] Arkansas PWR
[5] Maryland, PWR

2020 at 15 GWd/MTU for PIE at INL's hot fuel examination facility (HFEF). Examinations revealed excellent coating adherence with smooth transitions to the underlying substrate. No significant change in coating thickness, hydrogen pickup, or mechanical behavior was observed. Three additional pins were discharged from ATF-2 in 2021. Visual inspection of these pins revealed some tarnish but this was attributed to the stainless steel test train that held the pins. The PIE campaign for these pins is ongoing and the remaining rod will be irradiated to end of life burnup (Nimishakavi 2022).

Sixteen lead test rods with Cr-coated M5® cladding and Cr_2O_3-doped UO_2 fuel were inserted at Vogtle Unit 2 in spring 2019. Visual inspections performed in 2020 and 2022 revealed no signs of delamination. Assemblies containing these lead test rods were returned for a third cycle, which ended in 2023 and was followed by pool-side inspections and limited PIE (Nimishakavi 2022). Results from this third cycle are pending.

Two host assemblies containing 20 Cr-coated LTRs began irradiation in 2019 in Gosgen under the GOCHROM program. This is an extension of the IMAGO program and is intended to provide additional data on Cr-coated fuel rods. After two years of irradiation, no signs of delamination or degradation were observed. One LTR was discharged after the 1st cycle and sent to the Paul Scherrer Institute (PSI) for examination. Most of the remaining rods will complete their irradiation after 5 cycles but some will remain for a 6th cycle to achieve a burnup of 70 MWd/kgU) (Nimishakavi 2022).

Thirty-two Cr-coated lead test rods were inserted in fall 2019 at ANO Unit 1. Fueled rods were placed near the core center while some Cr-coated inert rods were placed in baffle slots. The primary objective of this irradiation was to build the database for Cr-coated cladding performance, along with a second objective to evaluate the resistance of the Cr-coated cladding to baffle wear. After one cycle of irradiation, the fueled rods were visually inspected and revealed a lustrous-gold appearance indicating a significant reduction in corrosion relative to uncoated rods. Assemblies containing the inert rods were discharged in 2021 and poolside examinations were performed in January 2022. The coating appeared tightly adherent despite the complex interactions with the baffle wall, which resulted in wear scars on the cladding that indicate the coating did not enhance baffle wear resistance. Subsequent irradiation of the fueled rods was planned for two more cycles (to 2024) with poolside inspection to follow each cycle (Nimishakavi 2022).

Two full assemblies with Cr-coated M5® cladding and Cr_2O_3-doped UO_2 fuel were inserted at Calvert Cliffs in spring 2021 (Reed and Boman 2019). Framatome is on track for a 2025 batch reload (Reed 2019). Framatome is not currently developing any FeCrAl cladding concepts.

2.3 Global Nuclear Fuels

Together with General Electric, Global Nuclear Fuels (GNF) is working toward commercializing two ATF designs for use in BWRs: Abrasion Resistant, More Oxidation Resistant (ARMOR) cladding, a coated zirconium alloy cladding with UO_2 fuel, and a FeCrAl cladding called IronClad (Fawcett 2019).

Both ARMOR and IronClad have been tested in ATR. LTAs containing unfueled, IronClad-segmented rods and ARMOR-segmented rods were irradiated at Plant Hatch[1] and discharged in February 2020 (though one rod will go through two additional cycles). Post-irradiation

[1] Georgia BWR

examination (PIE) results were expected by January 2021. The PIE of the unfueled C26M rods discharged from Hatch was ongoing at ORNL as of 2023 (Kane 2023). LTAs with both ARMOR and three varieties of IronClad clad rods were installed at Clinton[1] (GNF 2020). At the time of this report, no updates for these PIE campaigns were available.

2.4 Japan

The Japanese ATF R&D program is developing a number of ATF concepts including: FeCrAl cladding strengthened by the dispersion of fine oxide particles (FeCrAl-ODS), SiC/SiC composite cladding, and doped-UO_2 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).

In 2022, results were presented from a study conducted on a proposed FeCrAl-ODS BWR cladding alloy to gather data to support normal operating, accident, and storage conditions. Data gathered for normal conditions included buckling and cesium-stress corrosion cracking. Accident conditions included both DBA and BDBA scenarios and these data were obtained from simulated LOCA tests performed at ORNL. Data gathered for storage conditions were obtained from tensile specimens at 150°C. These specimens were previously irradiated in HFIR to 3.9 dpa at 300°C. Results indicated that cesium-stress corrosion cracking and buckling should not be a problem for FeCrAl-ODS fuel claddings in BWRs. The FeCrAl-ODS cladding was also resistant to burst under a wide range of internal pressures and temperatures during LOCA testing. Tensile data collected to estimate cladding performance during storage did not reveal any significant loss of ductility and results obtained at 150°C were intermediate between those obtained at room temperature and 300°C (Sakamoto 2022).

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 UO_2) (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 were indicated at the time of this report.

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-alloy cladding [Cr alloy-coated and oxide dispersion strengthened (ODS)], SiC cladding, Fe-based alloy cladding, doped UO_2 fuel, microcell-, microplate-UO_2 (high

[1] Illinois BWR

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 (Szőke, 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 and discharged in February 2020. Additional rods havel undergone a second cycle of irradiation. PIE, refabrication, and out-of-pile testing is planned at ORNL.

As of 2021, unfueled C26M rods were being irradiated at Hatch and unfueled APMT-2 rods were being irradiated in Clinton. Fueled C26M rods were also being irradiated in Clinton and fueled rodlets of C26M were being irradiated in ATR as well (Dolley 2021). The PIE of the unfueled C26M rods discharged from Hatch was ongoing at ORNL as of 2023 (Kane 2023).

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).

All testing at ATR was halted April 23rd, 2021, when the reactor was shut down to support the core internals changeout (CIC). This is an activity scheduled every 7-10 year to refurbish the core. Irradiations resumed when the ATR was restarted on April 25, 2023.

3.0 Overview of Cr-Coated Zr-alloy Cladding

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

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 Zry-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.

Investigations into the use of protective coatings for in-reactor environments have been performed previously and chromium-based coatings were explored by CEA in the 1960s. At that time, the main objective was to enhance corrosion resistance of zirconium-based alloys exposed to a CO_2 environment for graphite-gas reactors (Brachet 2019). As previously indicated, Cr-coatings on currently licensed Zr-based claddings are now a promising response to current ATF needs. Recent reviews of ex-reactor coating studies, by Kashkarov and Yang, provide insight on performance concerns of the proposed Cr-coatings.

Kashkarov (2021) provided a comprehensive review of a wide array of protective coatings for Zr-based accident tolerant fuel claddings. These coatings are intended to improve corrosion resistance, high temperature oxidation resistance, wear resistance, and reduce hydrogen absorption. Improving these properties helps ensure that the thermomechanical properties and overall integrity of the claddings are retained and that hydrogen embrittlement is mitigated both in service as well as during storage and transportation of spent fuel. This would be particularly useful for fuels with high burnup or increased enrichment due to their increased residence time in the reactor. Kashkarov noted that coatings, such as Cr, that produce a layer of chrome oxide (i.e., chromia, Cr_2O_3) were significantly more stable than either alumina or silica in simulated autoclave tests in the presence of PWR media. Thickness, density (i.e., absence of porosity), and microstructure were also identified as key attributes of the coating. A minimum Cr plating thickness of 10 μm was identified as providing adequate protection while a maximum thickness of 30 μm was proposed to avoid a significant neutronic penalty. This upper bound is consistent with the previous OECD report (OECD-NEA 2018), i.e. < 20 μm, and includes thicknesses explored in other studies (e.g. 15 μm thick coatings described in Bischoff (2018)). As for the density and microstructure of the coating, Kashkarov indicated that the coating should be free of porosity and columnar grains to mitigate diffusion of oxygen and hydrogen between the surface and the substrate in order to protect the substrate from oxidation and hydriding.

Yang (2022) provided a review specifically focused on chromium coated zirconium alloys for accident tolerant fuel claddings. Results presented in this review revealed that pre-existing cracks, which can result from the coating process, provide short diffusion paths for oxygen transport. Following oxidation in a steam environment, these pre-existing cracks can result in the formation of ZrO_2 nodules surrounded by oxygen stabilized α-Zr(O). Subsequent modeling suggests these localized zones may provide crack initiation sites during ballooning that can lead to failure despite the ability of the outer chromia layer to self-heal at high temperature (1000°C). Yang identified other potential failure mechanisms of the coated cladding including the Cr-Zr eutectic reaction, ballooning, the formation of bubbles/blister/and voids, and local oxidation of the weld zone.

The formation of the Cr-Zr eutectic can threaten the coating when temperatures exceed the melting point of the eutectic (~1330°C). This may not be a concern for DBA if the maximal temperature of 1200°C is not exceeded. However, in BDBA conditions, cladding temperatures exceed 1330°C. A potential solution for this concern is a Mo interlayer between the Cr coating and the Zr substrate, which forms a higher temperature eutectic with zirconium (1500°C) (Yang 2022).

Under ballooning conditions, the chromium coating on the outer surface of the cladding will no longer be protective due to extensive crack formation near the ballooned region. Upon rupture of the ballooned region, the inner surface of the cladding tube will be susceptible to high temperature oxidation. A potential solution for this concern is to coat the inner surface of the tube in addition to the outer surface. Although the coating would not be protective near the ballooned region due to cracking, it would protect the remainder of the inner surface from high temperature oxidation (Yang 2022).

Another concern is the formation of bubbles/blisters/voids at the coating-cladding interface. These have been observed after simulated high temperature excursions and pose a threat to the Cr-coating if cracked or ruptured. The formation of these features is attributed to mismatches in volume expansion in the surface oxide, coating, and coating-substrate interface. The differences observed between test data are attributed to the test environment (inert gas, steam, air). Nitrogen has a high affinity for Cr and Zr and can affect the oxidation process of Cr coatings. Additional testing in representative environments is expected to reveal more information about the features, their formation, and potential impacts on the protectiveness of the Cr coating (Yang 2022).

Finally, local oxidation near the weld zone was also considered. Results from an air oxidation experiment on fully Cr-coated Zr alloy tubes were used to evaluate the effect of resistance upset welding. Partial oxidation enhancement was observed in the weld burr region where the Cr coating was absent; allowing oxygen and nitrogen to diffuse into the Zr substrate. This creates a potential weak spot in the cladding tube but additional process refinement could be used to minimize these effects. The review concludes with mention of an OECD-NEA Joint Undertaking project called “QUENCH-ATF”, which is aimed at investigating the chemical, mechanical, and thermal-hydraulic behavior of ATF claddings in DBA and BDBA scenarios (Yang 2022).

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.

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-alloy cladding near-term ATF candidates. The technique used by GNF is proprietary.

Each coating technique is unique. Specific coating technique processes used by the manufacturers may be partially or fully proprietary. The following techniques 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 relatively new technology and has attracted serious attention since unique coating properties can be obtained by this process that are not achievable by conventional thermal spraying. This uniqueness is because coating deposition can take place without exposing the spray or substrate material to high temperature and without melting the sprayed particles. Consequently, oxidation and other undesired reactions such as the interdiffusion between the substrate and coating can be significantly limited or avoided altogether. Spray particles adhere to the substrate only because of their high kinetic energy on impact. Successful bonding is achieved when spray particles exceed a critical impact velocity, which is dependent on the properties of the spray material (Gärtner et al. 2006; Maier et al. 2018, 2019; Yeom et al. 2019).

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 but the sputtering process is frequently cited when discussing coated claddings. These three processes are briefly 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 µm/min) but complex shapes are difficult to accommodate, and the coatings often exhibit poor adhesion (Pierson 1999).

3.1.2.2 Sputtering

Sputtering is a technique used to create thin films and 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 (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).

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). As with evaporation, the method is limited to those metals that can be vacuum evaporated (Grainger 1998).

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, 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) 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 a CrN or Cr_2O_3, 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.

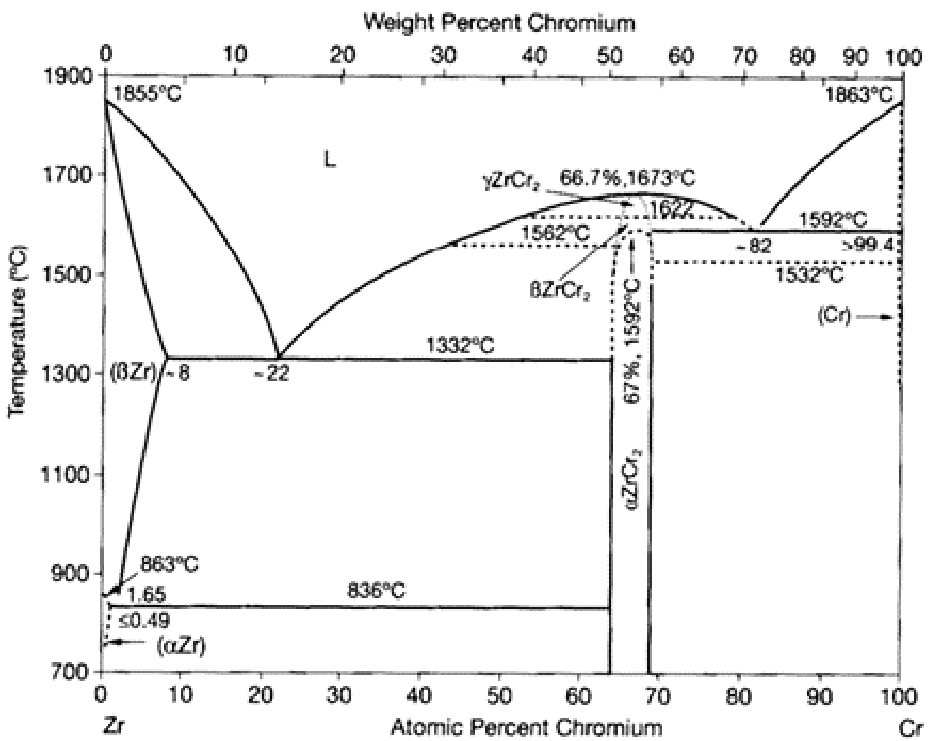

Figure 3.1. Zr-Cr phase diagram (Arias and Abriata 1986).

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 β-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 temperature.

While it is important to understand that eutectics exist for Cr-coated Zr-alloy cladding, conditions associated with spent fuel storage (400°C for normal conditions and 570°C for accident conditions) do not approach the eutectic temperatures identified. Furthermore, there is no intention to coat the inner surface of the cladding, 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-phase 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 temperatures required to attain stable equilibrium. Consequently, the details of the $\beta + \gamma$ phase region of the Cr-Zr phase diagram are 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 (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-alloy Cladding Degradation and Failure Modes

Unlike in-reactor safety analysis, there are no specified acceptable design limits put on the fuel cladding for spent fuel storage and transportation analysis. However, to certify that a 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. These analyses will be discussed in greater detail in Section 5.0. 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-alloy cladding as well. These include:

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

Preliminary data indicates that hydrogen embrittlement and delayed hydride cracking should be of lower concern for Cr-coated Zr-alloy cladding based on a reduction of hydrogen pickup.

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-alloy 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. Japanese efforts intend to greatly improve strength by pursuing ODS FeCrAl alloys (Terrani 2018).

GNF remains the only fuel vendor in the U.S. with FeCrAl cladding planned for the development. 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, the 8 fueled IronClad rods inserted in Clinton were C26M, and the 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 in the following subsections.

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 Y_2O_3

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 additions reduce the driving force of Cr-rich α’ phase formation even in alloys with relatively high Cr contents (Field et al. 2018; Wukusick 1966).

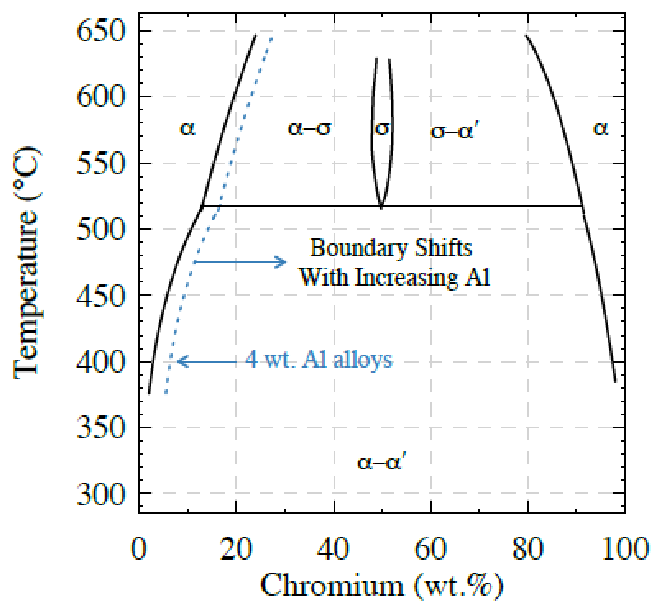

Figure 4.1. Fe-Cr binary alloy phase diagram showing phase boundaries of α-Fe, α’-Cr, and σ-FeCr. The effect of a 4 w/o Al addition on the alpha-α’ phase boundary is also shown as example (Field 2018; Wukusick 1966).

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 alloys[1], intended to increase alloy strength. Mo addition increases alloy strength through solid-solution hardening; Nb addition increases alloy strength through precipitate strengthening by the formation of Fe_2Nb-type Laves phase particles (Raiman et al. 2020).

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

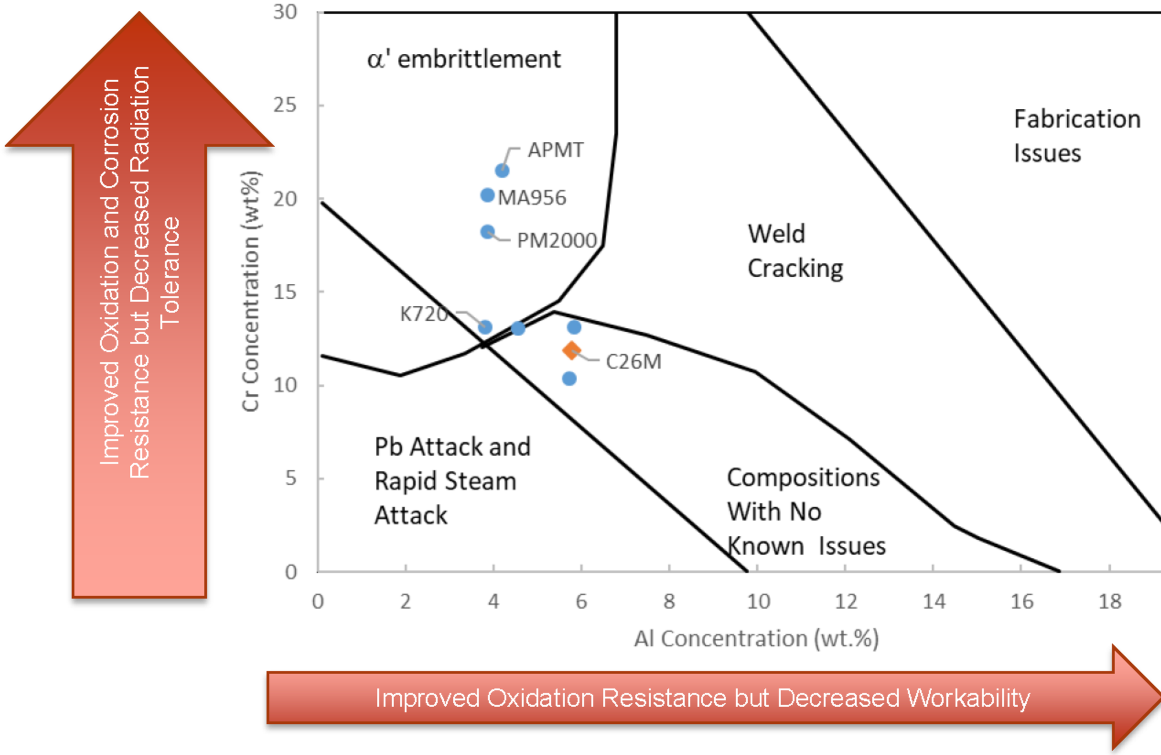

Figure 4.2. Impact of chromium and aluminum concentration in FeCrAl alloys (Yamamoto, Field, et al. 2020).

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 was 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 were 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)
- 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 B_4C 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/B_4C combinations.

The FeCrAl/B_4C 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/B_4C 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 B_4C, which can be confirmed by future 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 failed.

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.

4.6 FeCrAl Cladding Degradation and Failure Modes

Unlike in-reactor safety analysis, there are no specified acceptable design 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. These analyses will be discussed in greater detail in Section 5.0. 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. 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 for in-reactor service. 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[1]. These limits may need to be revised for ATF concepts. 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 zirconium-based 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 a secondary motivation in the development of ATF is to improve 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. Future revisions can be expected as fuels with increased enrichment to achieve even higher burnups are beginning to be explored.

In addition to burnup, safety analyses for the storage and transportation of spent nuclear fuel is also influenced by fuel enrichment (Zipperer et. al. 2020). Increased fuel enrichment results in increased reactivity during subsequent storage and transportation operations. The addition of absorbers, such as soluble boron to the spent fuel pool and the installation of permanent neutron absorbers in dry cask baskets, is used to suppress additional positive reactivity. To better account for the added reactivity, credits are taken for items such as the decay of fissile fission products, build-in of neutron absorbers, use of integral absorbers, and increased fuel burnup. Fuel enrichments above 5 w/o ^{235}U can achieve very high burnup levels (80 GWd/MTU) but, while burnup credits at these levels may benefit spent fuel requirements, the corresponding increase in neutronic de-coupling and lack of data on spent fuels with enrichments above 5 w/o add additional uncertainties to these analyses. Guidance for the storage and transportation of spent nuclear fuels that employ ATF with enhanced enrichments should be revised accordingly (Zipperer et al 2020).

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 and 10 CFR 71, respectively) do not specifically prohibit the use of either ATF concept 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) (U.S. Nuclear Regulatory Commission 2020a; U.S. Nuclear Regulatory Commission 2020b) mention Zr-alloy cladding and stainless steel. There is no mention of either ATF concept in these documents; however, due to

[1] For low-burnup fuel (≤45 GWd/MTU), the peak cladding temperature is allowed to exceed 400°C as long as the cladding hoop stress is ≤90 MPa. There is no cladding stress limit placed on high-burnup fuel.

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 is the empirically derived temperature and stress limits in the SRP, which include:

- 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 (≤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 and the cladding temperature difference limited to less than 65°C (149°F) per cycle.
- For off-normal and accident conditions the maximum cladding temperature should not exceed 570°C (1,058°F).
- For FeCrAl, if the empirical temperature limits of 570°C (1058°F) for off-normal and accident conditions and 400°C (752°F) for normal conditions will be used, data should be provided to demonstrate that the particular alloy will retain sufficient strength at those temperatures.
- 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 (U.S. 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 ATF cladding concepts. 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-alloy cladding. Embrittlement can be caused by interdiffusion between Cr and Zr, but this is not expected to be significant (see Section 3.4). Iron is not embrittled by hydrogen, as zirconium is, thus hydrogen embrittlement is not an issue for FeCrAl (Hales and Gamble 2015). As license applications for ATF concepts are submitted, the NRC should be made aware of any other environmental factors that could embrittle ATF cladding.

The following subsections provide a critical evaluation of the documents that provide the regulatory framework for the safety analyses regarding wet storage (Section 5.1), dry storage (Section 5.2), and transportation (Section 5.3) of SNF as it relates to Cr-coated and FeCrAl ATF concepts. Data recommendations for safety evaluations are provided in 5.4.

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 Cr-coated and FeCrAl 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” (U.S. 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:

- 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

In the context of high burnup fuel, Regulatory Guide 1.13 (US Nuclear Regulatory Commission 2007) identifies that the mechanical properties of the fuel and cladding may change with higher burnup. For instance, high-burnup fuel may become more brittle (i.e., possess lower ductility and fracture toughness) and, therefore, be more vulnerable to failure. In order to protect high-burnup fuel from mechanical damage, this potential vulnerability should be considered in the design of spent fuel handling and storage facilities. Additionally, the decay heat of high burnup SNF is likely greater and needs to be considered. Although there is no expectation that the fidelity of models to calculate decay heat would decrease at high burnup, it’s possible that longer wet storage times may be necessary prior to transferring high burnup fuel assemblies to dry storage.

With regard to fuels with enhanced enrichment, safety analyses within the spent fuel pool must account for the added reactivity. The addition of absorbers, such as soluble boron to the spent fuel pool and the installation of permanent neutron absorbers in dry cask baskets, is used to suppress additional positive reactivity during wet storage and cask loading operations in the spent fuel pool. To better account for the added reactivity, credits are taken for items such as the decay of fissile fission products, build-in of neutron absorbers, use of integral absorbers, and increased fuel burnup. Fuel enrichments above 5 w/o ^{235}U can achieve very high burnup levels (80 GWd/MTU) but, while burnup credits at these levels may benefit spent fuel requirements, the corresponding increase in neutronic de-coupling and lack of data on spent fuels with enrichments above 5 w/o add additional uncertainties to these analyses. Experimental benchmarks should be developed to justify uncertainties in the depletion calculations that form the basis for reactivity credits (Zipperer et. al. 2020).

5.1.2 Application to Cr-Coated Zr-alloy Cladding

For Cr-coated Zr-alloy 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-alloy cladding, then it would be reasonable to use

[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)

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. However, it has recently been found that Cr coatings under BWR normal water chemistry conditions at 300°C can completely dissolve (Umretiya et. al. 2023). Similar tests have not been performed on coatings under spent fuel pool water chemistry and temperatures. Period inspections of Cr-coated cladding in the spent fuel pool is recommended to confirm the coating stability.

Use of Cr-coated Zr-alloy cladding on high burnup (i.e., > 62 GWd/MTU) fuel rods is not expected to cause issues during wet storage of spent fuel. The enhanced corrosion and wear resistance provided by Cr-coated Zr-alloy cladding is expected to mitigate the oxidation and hydride embrittlement associated with Zr-alloy clad fuel rods irradiated to high burnup. However, additional data should be gathered as burnup levels are increased to verify that the overall integrity of the coating and cladding remains intact. Also, an underlying assumption is that the fuel form (e.g., UO_2) does not degrade the cladding at high burnup through pellet-cladding interaction or generate excessive pressure through fission gas release at high burnup. Data should be gathered to verify this assumption, particularly when considering alternative fuel forms to pair with Cr-coated Zr-alloy cladding. Additionally, depletion and decay heat models with appropriate uncertainties will need to be developed since current guidance (RG 1.240) only extends to 60 GWd/MTU burnup.

The impact of using Cr-coated Zr-alloy cladding with fuels with enhanced enrichment (> 5w/o ^{235}U) is not expected to impact wet storage requirements. Spent fuel pool analyses should be performed to ensure subcriticality is maintained and that existing reactivity control measures are compatible with the fuel cladding. Burnup credits can be used to account for reactivity but experimentally determined benchmarks should be developed to justify the uncertainties in the corresponding depletion codes. Criticality, depletion, and decay heat models with appropriate uncertainties will need to be developed since current guidance (RG 1.240) only extends to 5 wt% ^{235}U.

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 that the Co-60 activation from Fe that could lead to increased worker dose. Thus, it should be considered in the spent fuel analysis. To estimate the impact FeCrAl cladding could have to worker dose, a PWR assembly irradiated to 55 GWd/MTU rod was modeled using an ORIGEN-ARP library and the dose after various cooling times was calculated using MAVRIC. The results for a Zircaloy cladding and a FeCrAl cladding are shown in Table 5.1 after various cooling times. These calculations indicate that the dose for an assembly with either cladding is essentially the same.

Table 5.1. Comparison of dose from high burnup fuel assembly with Zircaloy and FeCrAl cladding.

Case	Dose from Unshielded Fuel Assembly at 3 ft (rem/hr) Zircaloy Cladding	Dose from Unshielded Fuel Assembly at 3 ft (rem/hr) FeCrAl Cladding
PWR, 55 GWd/MTU, 5 year decay	1762	1784
PWR, 55 GWd/MTU, 20 year decay	538	540
PWR, 55 GWd/MTU, 40 year decay	324	321

Use of FeCrAl cladding on high burnup (i.e., > 62 GWd/MTU) fuel rods is not expected to cause issues during wet storage of spent fuel. The enhanced corrosion and wear resistance provided by FeCrAl cladding is expected to mitigate the oxidation concerns associated with fuel rods irradiated to high burnup. However, additional data should be gathered as burnup levels are increased to monitor for any signs of cladding degradation. Also, an underlying assumption is that the fuel form (e.g., UO_2) does not degrade the cladding at high burnup through pellet-cladding interaction or generate excessive pressure through fission gas release at high burnup. Data should be gathered to verify this assumption, particularly when considering alternative fuel forms to pair with FeCrAl cladding. Additionally, depletion and decay heat models with appropriate uncertainties will need to be developed since current guidance (RG 1.240) only extends to 60 GWd/MTU burnup.

The impact of using FeCrAl cladding with fuels with enhanced enrichment (> 5w/o ^{235}U) is not expected to impact wet storage requirements. Spent fuel pool analyses should be performed to ensure subcriticality is maintained and that existing reactivity control measures are compatible with the fuel cladding. Burnup credits can be used to account for reactivity but experimentally determined benchmarks should be developed to justify the uncertainties in the corresponding depletion codes. Criticality, depletion, and decay heat models with appropriate uncertainties will need to be developed since current guidance (RG 1.240) only extends to 5 wt% ^{235}U.

5.2 Dry Storage of Spent Nuclear Fuel

This section discusses the dry storage of SNF, including the current regulatory framework application to Cr-coated and FeCrAl ATF concepts, and application to fuels with high burnup and enhanced enrichment.

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" (U.S. Nuclear Regulatory Commission 2015), include a number of fuel-specific and dry storage system (DSS)-specific requirements that may be dependent on the design basis condition of the fuel cladding. 10 CFR 72.44I 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 and 10 CFR 72.122(h)(1) states 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, 10 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” (U.S. Nuclear Regulatory Commission 2010), and NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities” (U.S. Nuclear Regulatory Commission 2000). These guidelines have recently been replaced by NUREG-2215 (U.S. 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” was defined as an ability to retrieve a fuel assembly intact using its normal lifting hardware and has since been expanded to also include retrieval of the canister.

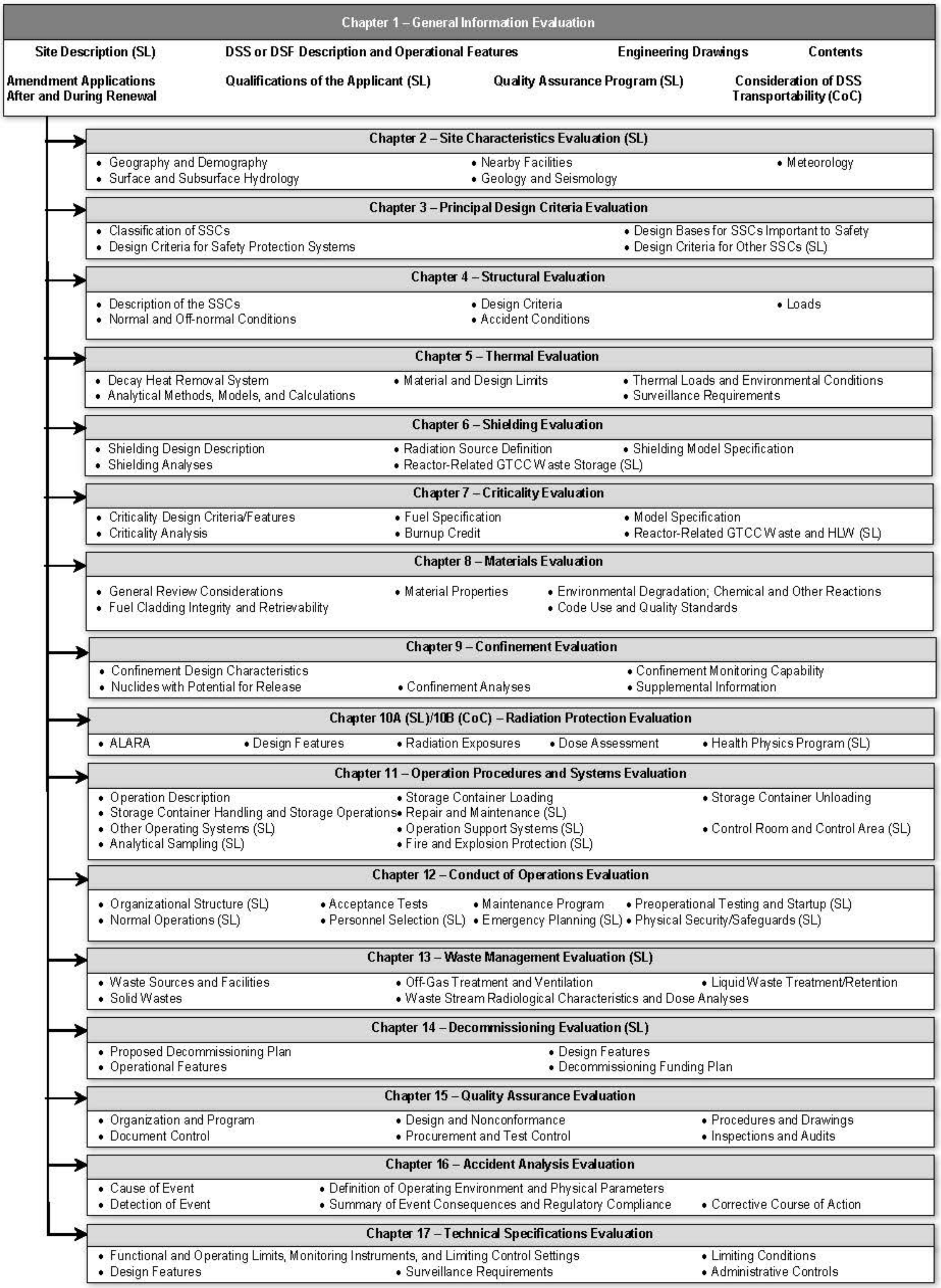

Figure 5.1. Overview of safety evaluation of a DSS (Taken from NUREG-2215).

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 (U.S. Nuclear Regulatory Commission 2010) was revised in 2010 to reflect the latest guidance in ISG-11, Rev. 3. NUREG-1567 (U.S. Nuclear Regulatory Commission 2000) was replaced by NUREG-2215 (U.S. 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 experience less than 10 thermal cycles each <65°C.

NRC staff have recently published NUREG-2224 on dry storage and transportation of high burnup SNF (Ahn, et al. 2018). This document considers fuel with burnup up to 65 GWd/MTU. Additionally, NRC staff recently presented their position 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 (U.S. Nuclear Regulatory Commission 2019) that provides the technical basis for these issues.

5.2.2 Application to Cr-Coated Zr-alloy 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-alloy cladding are:

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

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) taken at relevant temperatures from tensile tests and burst tests.

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 experience less than ten thermal cycles each <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
- Shielding Evaluation: Component Materials, Dimensions
- Criticality Evaluation: Component Materials, Dimensions
- Materials Evaluation: Cladding Integrity.
- Confinement Evaluation: Component Materials, Dimensions

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 were developed for Zr-alloy cladding and are articulated in ISG-11 Rev. 3 and have historically been used for storage of SNF, or propose new limits based on performance data for FeCrAl cladding as current limits were empirically developed for Zr-alloy cladding and are likely not relevance to FeCrAl cladding:

- 400°C peak cladding temperature limit for NCS
- Maximum cladding temperature limit of 570°C for off-normal and accident conditions
- The cladding should experience less than ten thermal cycles each <65°C.

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

5.2.4 Application to Fuels with High Burnup and Enhanced Enrichment

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 fuels with burnup extending from 62 GWd/MTU to 85 GWd/MTU and/or enhanced enrichment (> 5w/o ^{235}U) are:

- Principal Design Criteria Evaluation: Spent Fuel Design Basis
- Structural Evaluation: Component Materials
- Thermal Evaluation: Spent Fuel Cladding, Component Materials, Decay Heat
- Shielding Evaluation: Component Materials
- Criticality Evaluation: Fissile Content Materials, Component Materials
- Materials Evaluation: Cladding Integrity.
- Confinement Evaluation: Component Materials

Based on these identified needs and discussions in NUREG-2224, the following information on fuels with high burnup and/or enhanced enrichment is needed to support the safety analysis of a DSS containing high burnup fuel:

- New cladding material properties (yield stress, ultimate tensile strength, uniform elongation)
- Decay heat from high burnup fuel
- Fissile content.

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

- Rod internal pressure (likely will go up relative to bounding pressures discussed in NUREG-2224 because of enhanced fission gas release observed in high burnup fuel)
- Oxide thickness (should be specific for the alloy in question)
- Hydrogen content (should be specific for the alloy in question).

The radioactive source term with appropriate uncertainties should be re-evaluated for fuels with high burnup and/or enhanced enrichment. NUREG-2224 provides a table of fractions of fuel rods assumed to fail and radioactive fractions available for release for high burnup fuel in non-leak tight dry storage system designs, per ANSI N14.5 (American National Standards Institute 2022). This table is reproduced as Table 5.2. This table is likely not applicable to burnup between 62 GWd/MTU and 85 GWd/MTU and should be reassessed based on the formation of the high burnup fuel rim that may lead to enhanced pellet fragmentation and the dramatic increase in fission gas release Fuels with enhanced enrichments will need reassessment to ensure subcriticality is maintained for SNF under NCS. Criticality models with appropriate uncertainties will need to be developed since current guidance (RG 1.240) only extends to 60 GWd/MTU and 5 wt% ^{235}U.

Table 5.2. Fractions of radioactive materials available for release from high burnup (up to 62 GWd/MTU) SNF under conditions of dry storage (for both PWR and BWR Fuels) (Ahn, et al. 2018)

Variable	Normal Conditions	Off-Normal Conditions	Accident-Fire Conditions	Accident-Impact Conditions
Fraction of Fuel Rods Assumed to Fail	0.01	0.1	1.0	1.0
Fraction of Fission Gases Released Due to a Cladding Breach	0.15	0.15	0.15	0.35
Fraction of Volatiles Released Due to a Cladding Breach	$3x10^{-5}$	$3x10^{-5}$	$3x10^{-5}$	$3x10^{-5}$
Mass Fraction of Fuel Released as Fines Due to a Cladding Breach	$3x10^{-5}$	$3x10^{-5}$	$3x10^{-3}$	$3x10^{-5}$
Fraction of CRUD Spalling Off Cladding	0.15	0.15	1.0	1.0

To address the issues of cladding creep and hydride reorientation, 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 experience less than ten thermal cycles each <65°C.

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

5.3 Transportation of Spent Nuclear Fuel

This section discusses the transportation of SNF, including the current regulatory framework, its application to Cr-coated and FeCrAl ATF concepts and application to fuels with high burnup and enhanced enrichment.

5.3.1 Current Regulatory Framework

The regulations in 10 CFR Part 71, “Packaging and Transportation of Radioactive Material” (U.S. Nuclear Regulatory Commission 2015), include a number of fuel-specific and package-specific requirements that may be dependent on 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” (U.S. Nuclear Regulatory Commission 2000). These guidelines have recently been replaced by NUREG-2216 (U.S. 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.

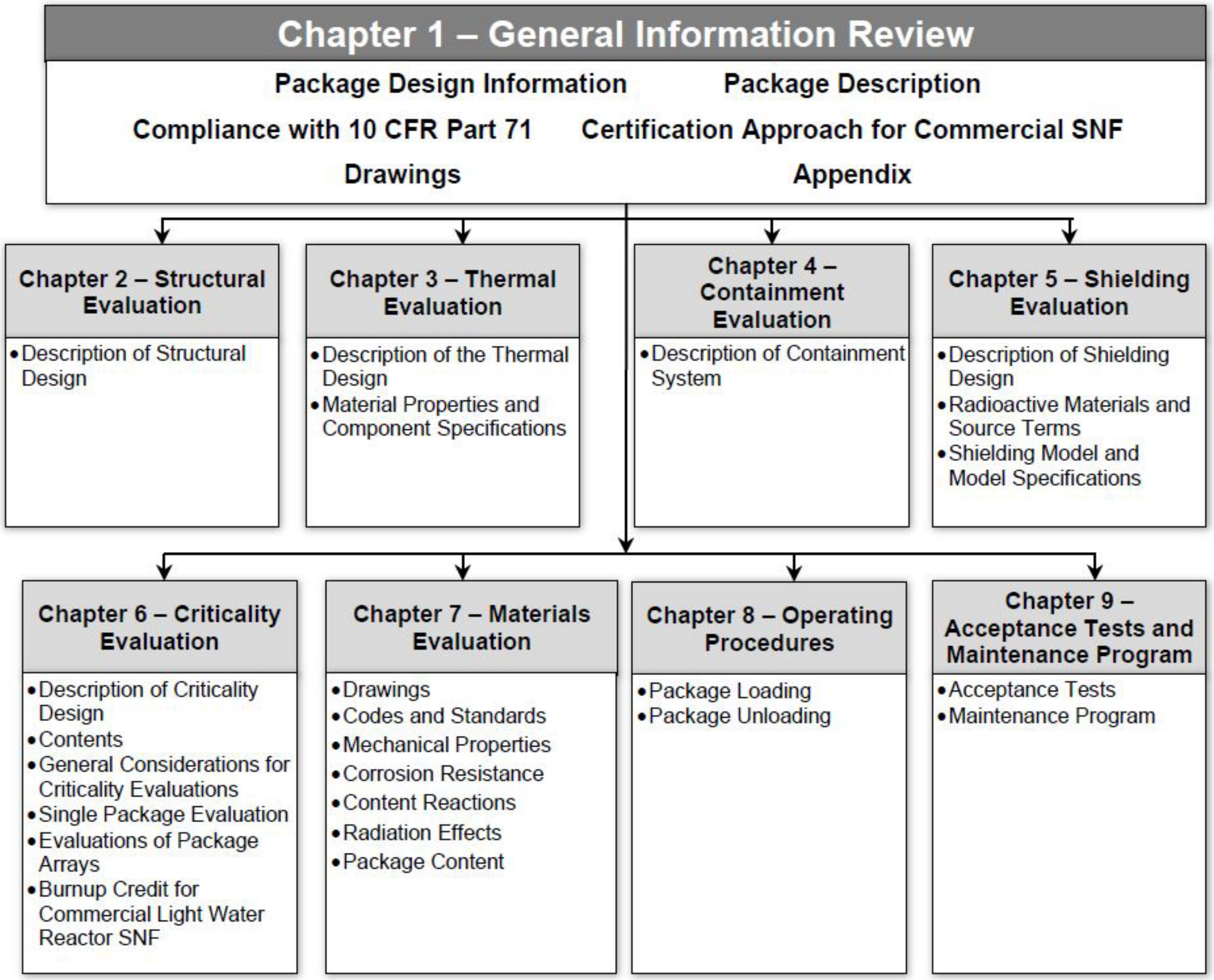

Figure 5.2. Overview of safety evaluation of SNF transportation (taken from NUREG-2216)

As burnups have increased and new data became 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 recently published NUREG-2224 (Ahn et al. 2018) regarding dry storage and transportation of high burnup SNF. This document considers fuel with burnup up to 65 GWd/MTU. Additionally, NRC staff recently presented their position 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-alloy 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-alloy cladding are:

- Structural Evaluation: Component Materials
- Thermal Evaluation: Component Materials
- Containment Evaluation: Component Materials
- Shielding Evaluation: Component Materials
- Criticality Evaluation: Component Materials.
- Materials Evaluation: Corrosion Resistance

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 and transport 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 experience less than ten thermal cycles each <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
- Containment 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 were developed for Zr-alloy cladding and are articulated in ISG-11, Rev. 3 and have historically been used for storage of SNF, or propose new limits based on performance data for FeCrAl cladding:

- 400°C peak cladding temperature limit for NCT
- Maximum cladding temperature limit of 570°C for off-normal and accident conditions
- The cladding should experience less than ten thermal cycles each <65°C

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

5.3.4 Application to Fuels with High Burnup and Enhanced Enrichment

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 fuels with burnup extending from 62 GWd/MTU to 85 GWd/MTU and/or enhanced enrichment (> 5 w/o ^{235}U) are:

- Structural Evaluation: Component Materials
- Thermal Evaluation: Dimensions, Component Materials, Decay Heat
- Confinement Evaluation: Component Materials
- Shielding Evaluation: Component Materials
- Criticality Evaluation: Fissile Content Materials, Component Materials.

Based on these needs identified and discussions in NUREG-2224, the following information on fuel with enhanced enrichment and/or high burnup and cladding is needed to support the safety analysis of a transportation package for SNF containing high burnup fuel:

- New cladding material properties (yield stress, ultimate tensile strength, uniform elongation)
- Cladding fatigue lifetime
- Decay heat from high burnup fuel
- Fissile Content.

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

- Rod internal pressure (likely will go up relative to bounding pressures discussed in NUREG-2224 because of enhanced fission gas release observed in high burnup fuel)
- Oxide thickness (should be specific for the alloy in question)
- Hydrogen content (should be specific for the alloy in question).

NUREG-2224 concludes that the use of best-estimate cladding mechanical properties, not accounting for the presence of the fuel pellet, continue to be adequate for assessing the structural performance of high burnup SNF during the hypothetical 9 m and 0.3 m drops. Additionally, the hydride orientation is not a critical consideration when evaluating these cladding mechanical properties. PNNL assesses that this will continue to be the case for SNF with enhanced enrichment (> 5 w/o ^{235}U) and/or high burnup between 62 and 85 GWd/MTU.

The radioactive source term with appropriate uncertainties should be re-evaluated for fuel with enhanced enrichment and/or high burnup. NUREG-2224 provides a table of fractions of fuel rods assumed to fail and radioactive fractions available for release for high burnup fuel in non-leak tight transportation packages, per ANSI N14.5. This table is reproduced as Table 5.3. This table is likely not applicable to burnup between 62 GWd/MTU and 85 GWd/MTU and should be reassessed based on the formation of the high burnup fuel rim that may lead to enhanced pellet fragmentation and the dramatic increase in fission gas release. Fuels with enhanced enrichments will need reassessment to ensure subcriticality is maintained for SNF under NCT. Criticality models with appropriate uncertainties will need to be developed since current guidance (RG 1.240) only extends to 60 GWd/MTU and 5 wt% ^{235}U.

Table 5.3. Fractions of radioactive materials available for release from high burnup (up to 62 GWd/MTU) SNF under conditions of transport (for both PWR and BWR fuels) (Ahn, et al. 2018)

Variable	NCT	HAC-Fire Conditions	HAC-Impact Conditions
Fraction of Fuel Rods Assumed to Fail	0.03	1.0	1.0
Fraction of Fission Gases Released Due to a Cladding Breach	0.15	0.15	0.35
Fraction of Volatiles Released Due to a Cladding Breach	$3x10^{-5}$	$3x10^{-5}$	$3x10^{-5}$
Mass Fraction of Fuel Released as Fines Due to a Cladding Breach	$3x10^{-5}$	$3x10^{-3}$	$3x10^{-5}$
Fraction of CRUD Spalling Off Cladding	0.15	1.0	1.0

To address the issues of cladding creep and hydride reorientation, the applicant should also justify the following limits that are articulated in ISG-11 Rev. 3 and have historically been used for transportation 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 experience less than ten thermal cycles each <65°C.

The source of information that could be used to assess these needs will be summarized in Section 5.4.

5.4 Data Recommendation for Safety Evaluations

Based on the assessments in the previous sections, the information in Table 5.4 is needed to support safety analyses of a DSS and an SNF transportation package containing fuel with Cr-Coated Zr-alloy 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 5.4. 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-alloy cladding and FeCrAl cladding.

Supporting Information	Recommended Source
Cr-coated Zr-alloy cladding	
Mechanical properties (yield stress, ultimate tensile strength, uniform elongation) Separate effects tests to identify phenomena that can lead to gross cladding rupture	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 temperatures for storage and transportation. 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 after being irradiated to target burnup.
Justification for peak cladding temperature limits regarding hydride reorientation: • 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) Separate effects tests to identify phenomena that can lead to gross cladding rupture	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 temperatures for storage and transportation. Failure limits at high burnup should be confirmed for creep strain capacity.
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 temperatures for storage and transportation.
Fatigue life	Fatigue tests performed on cladding segments that contain fuel or have been de-fueled after being irradiated to target burnup.
Justification for peak cladding temperature limits regarding strength requirements: • 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-alloy cladding and FeCrAl cladding 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, as well as any other fabrication processes, can impact the performance of the cladding. Rather, these data are compiled here to give NRC staff the expected cladding performance, 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-alloy Cladding

The key data needed for analysis of SNF with Cr-coated Zr-alloy cladding during storage and transportation are cladding mechanical properties and cladding fatigue. The current available data for Cr-coated Zr-alloy 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-alloy 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 significant cracking 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-alloy cladding

Organization	Cladding	Test Description	Results
Framatome (Brachet, et al. 2017)	Cr-coated M5® by PVD	Tensile tests at room temperature and 400°C Thermal creep at 400°C for 240 hours	Elastic modulus, yield stress, ultimate tensile strength and uniform elongation similar for coated and uncoated cladding. 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-alloy 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 unirradiated and 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)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
NC State (Joshi et al. 2020)	C26M	Burst Testing	Creep exponents and activation energies measured for C26M and compared with FeCAl alloys evaluated in other studies (e.g., C35M, T35AY, Kanthal® APMT)
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
ORNL (Bell et al. 2021)	C26M T35Y2	Tensile tests on axial tube specimens as a function of temperature to 800°C Burst testing	Yield stress, ultimate strength, and burst characteristics (e.g., stress, strain, burst opening)

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).

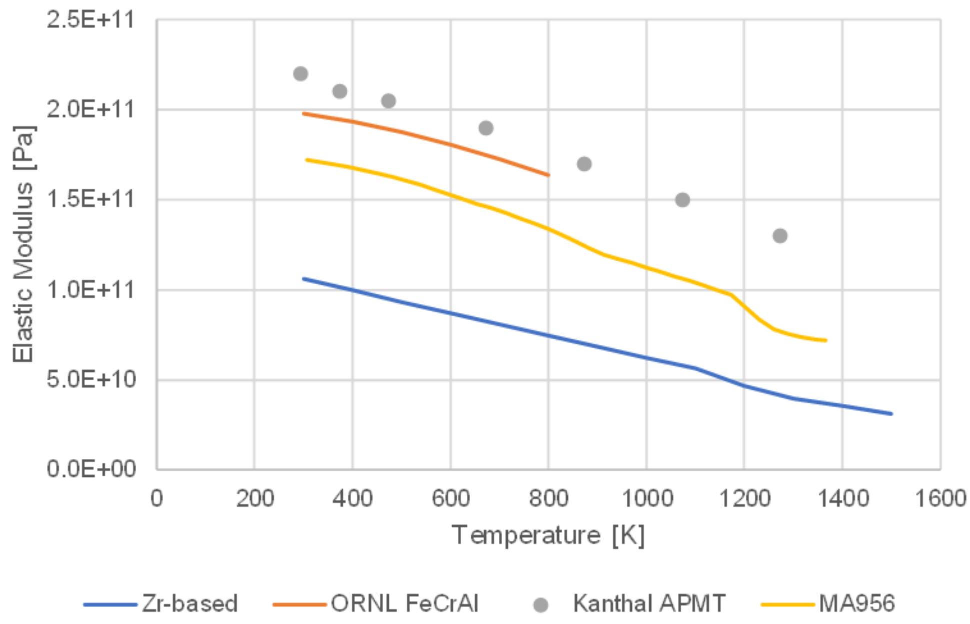

Figure 6.1. Elastic modulus of Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; Kanthal 2019; Special Metals 2004).

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: C35M and Kanthal® APMT (Field 2018), MA956 (Special Metals 2004), and C26M (Yamamoto et al. 2019).

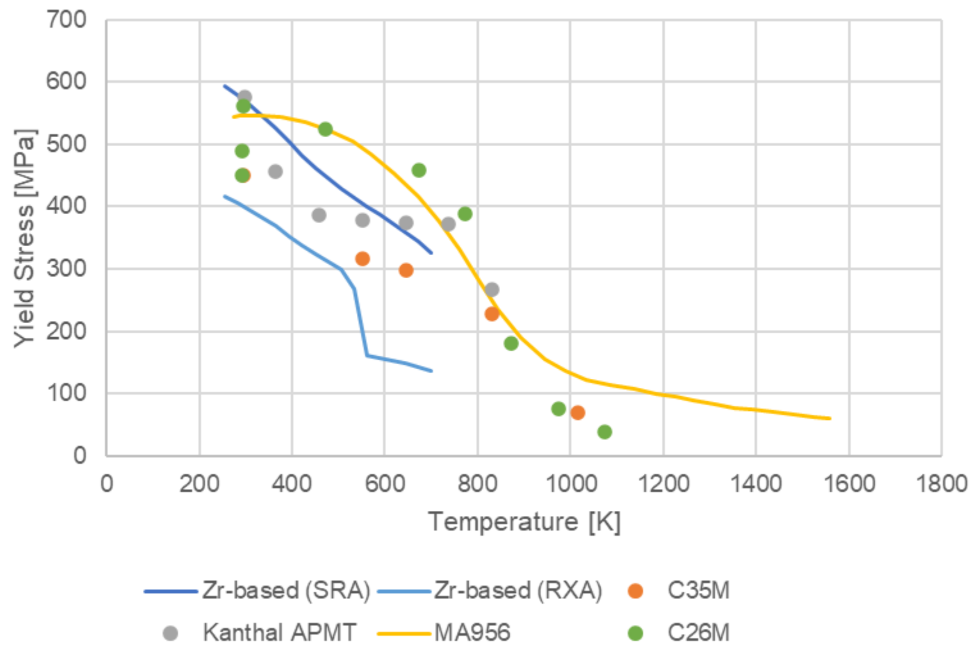

Figure 6.2. Unirradiated yield stress for Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; (Special Metals Corporation 2004); (Yamamoto, Kane, et al. 2019); Bell 2021).

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).

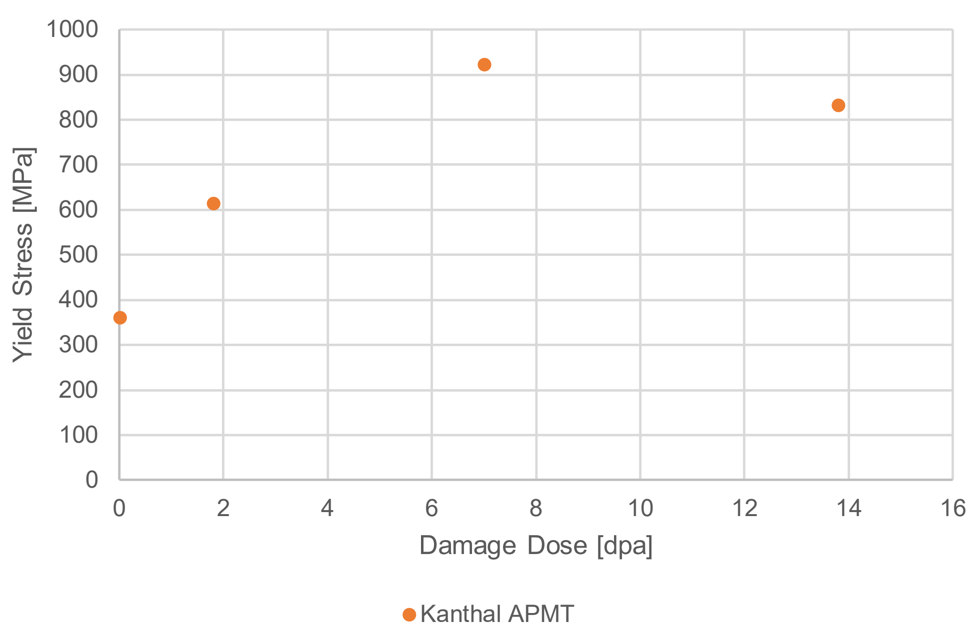

Figure 6.3. Irradiated yield stress for Kanthal® APMT at 320 °C (Field et al. 2017).

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 C06M C35M C36M	Differential scanning calorimetry, laser flash testing, dilatometry	Specific heat capacity, thermal diffusivity, thermal expansion
ORNL (Yamamoto et al. 2019)	C26M	Dilatometry, differential scanning calorimetry, and laser flash testing	Thermal expansion, heat capacity, and thermal diffusivity
China (Qiu 2020)	C35M Varying Compositions	Summary of available correlations in literature	Thermal conductivity, heat capacity, and thermal expansion

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.

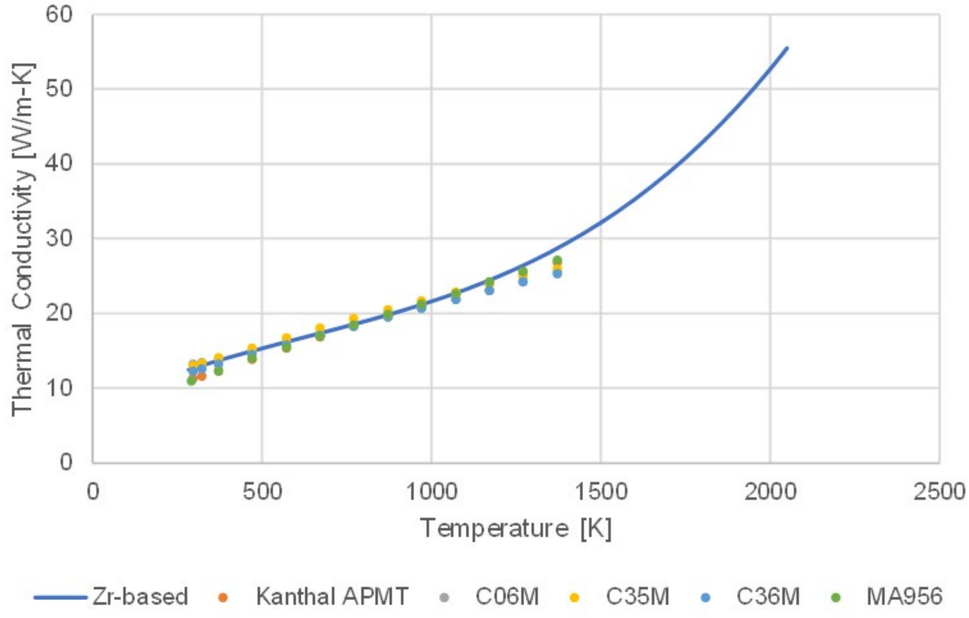

Figure 6.4. Thermal conductivity of Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; Special Metals 2004).

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.

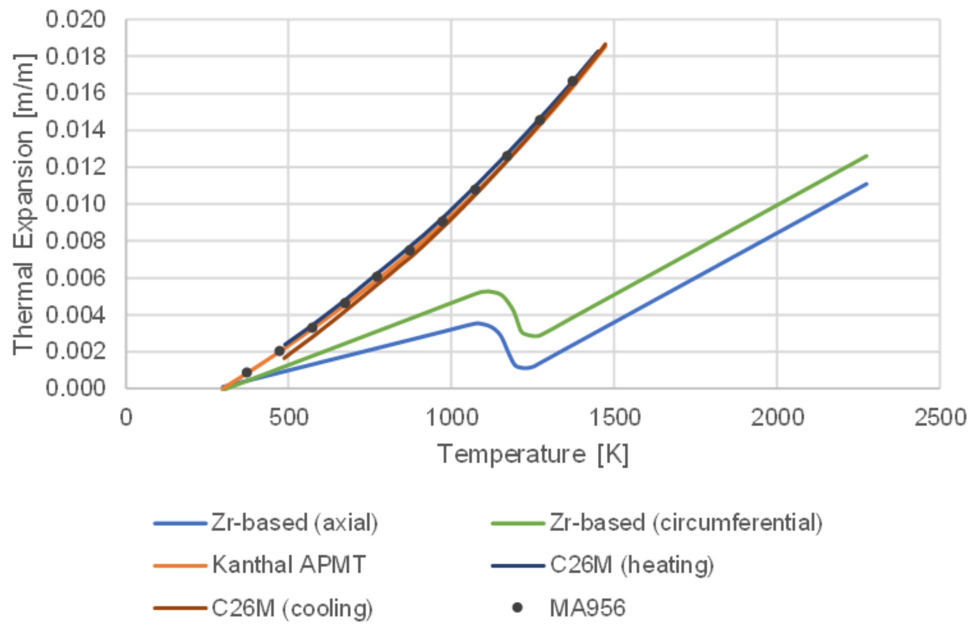

Figure 6.5. Thermal expansion of Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; Yamamoto et al. 2019; Special Metals).

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 expansion.

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.

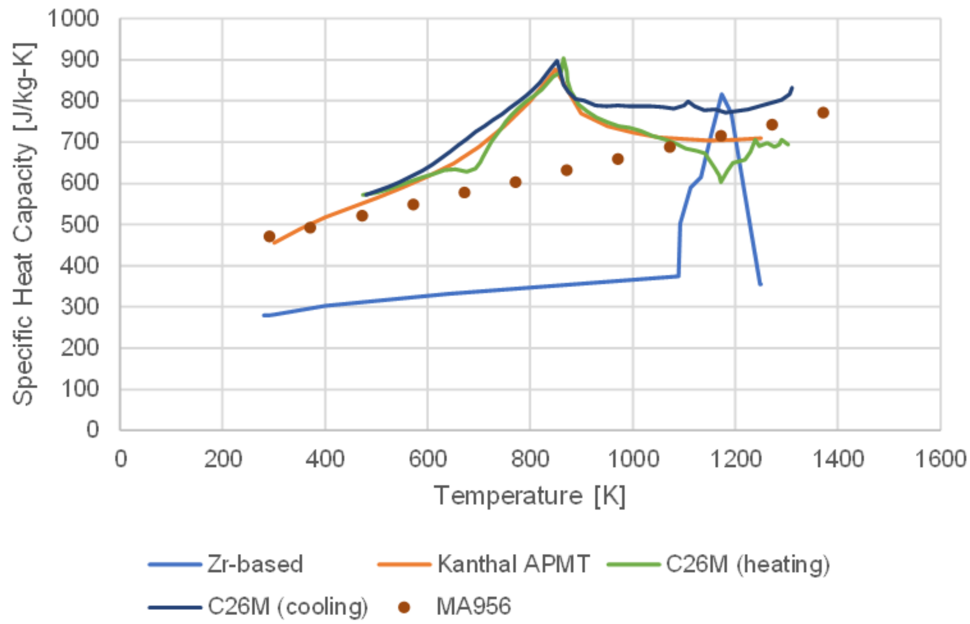

Figure 6.6. Specific heat of Zr-based alloys (Geelhood et al. 2020) and various FeCrAl alloys (Field 2018; Yamamoto et al. 2019; Special Metals).

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 Alloys	Test Description	Results
City University of Hong Kong (Field 2018)	Fe-23.85Cr-3.89Al Fe-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.

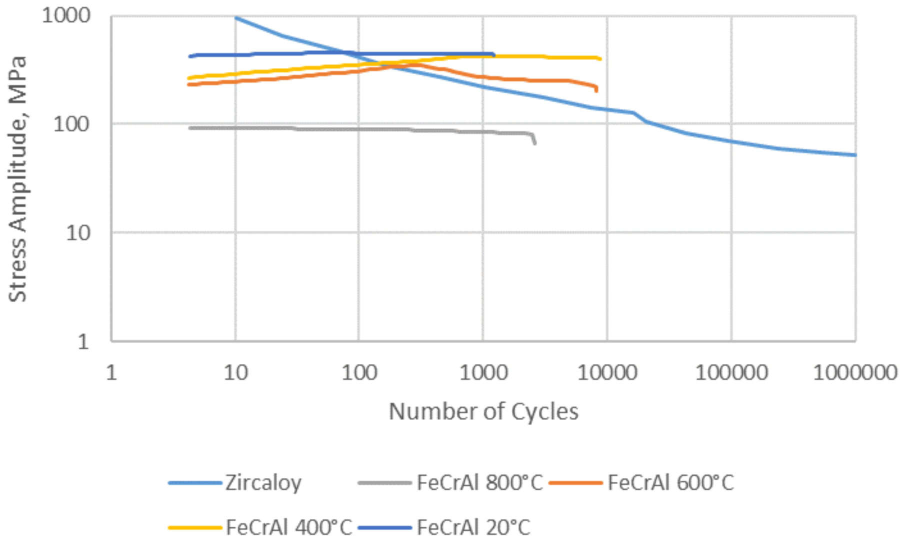

Figure 6.7. Fatigue lifetime curve for unirradiated Zircaloy and fatigue data from FeCrAl (Fe-23.85Cr-3.89Al).

7.0 Conclusions

This report provides an assessment of the NRC regulatory structure for storage and transportation of SNF as it relates to 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-alloy 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 strength requirements
- Cladding cross section and activation.

Sections 5.2.4 and 5.3.4 discuss changes to safety analysis codes, methods, and design limits for dry storage and transportation of high burnup SNF. Key items that have been identified as having greater uncertainty at high burnup and should be supported by test data are:

- Fatigue
- Cladding failure limits (creep strain and others)
- Fraction of rods assumed to fail and radioactive source term from failed rods (See Table 5.2 and Table 5.3)
- Empirically derived limits (temperature and cycling limits)
  - 400°C peak cladding temperature limit for NCS and NCT
  - Maximum cladding temperature limit of 570°C for off-normal and accident conditions

- Limitation to less than ten thermal cycles each <65°C.

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 was 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 their specific cladding license application.

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Appendix A

Phenomena Identification and Ranking Table (PIRT) for Fuel and Cladding Property Changes Relevant to Spent Fuel Storage and Transportation of Accident Tolerant Fuel Concepts

The U.S. Nuclear Regulatory Commission (NRC) requested the Pacific Northwest National Laboratory (PNNL) to assemble an expert elicitation panel to review the PNNL report, *Spent Fuel Storage and Transportation of Accident Tolerant Fuel Concepts: Cr-Coated Zirconium Alloy Cladding, FeCrAl Cladding, High Burnup and High Enrichment Fuel, PNNL-30451 Rev. 1*. This panel of experts shall:

1. Review the report cited above.
2. Prepare summary presentations that provide their expert perspectives on the material property changes necessary to certify spent fuel storage and transportation of degradation and failure modes of Cr-coated zirconium alloy cladding, FeCrAl cladding, high burnup fuel and high enrichment fuel.
3. Travel to and participate in an in-person panel where expert perspective on these issues will be discussed and the significance of each will be evaluated.
4. Review and comment on a final report prepared by the DOE laboratory that documents the expert elicitation.

The expert panel was selected by PNNL and is listed below.

Area of Expertise	Expert
Tubing manufacturing and certification	**Tim Brewer** Formerly of Pacific Northwest National Laboratory and Sandvik
FeCrAl alloy development and testing	**Raul Rebak** GE Vernova
Cr-coating development and testing	**Ben Maier** Westinghouse
Material properties measurement and testing	**Raul Rebak** GE Vernova **Anna-Maria Alvarez**[1] Studsvik

[1] Did not participated in PIRT exercise but did provide feedback on the report.

Area of Expertise	Expert
Spent fuel design basis conditions: drying, transfer, storage, and transportation	**Harold Adkins** Pacific Northwest National Laboratory
Spent nuclear fuel cask and canister requirements	**George Carver** NAC International
Fuel and cladding degradation modes under long term storage	**Mike Billone** Argonne National Laboratory **Hatice Akkurt** Electric Power Research Institute

The NRC requested that the discussion on the significance of each cladding and fuel property change be formalized as a Phenomena Identification and Ranking Table (PIRT) that is included in this appendix to the revision of the PNNL report, *Spent Fuel Storage and Transportation of Accident Tolerant Fuel Concepts: Cr-Coated Zirconium Alloy Cladding, FeCrAl Cladding, High Burnup and High Enrichment Fuel, PNNL-30451*. In addition to those listed in the table above, Framatome also sent a representative (Bryan Flannagan) to participate in the PIRT exercise.

The PIRT process that will be used to document this discussion is described below.

The PIRT Process

The NRC has adopted a nine-step process for implementing a standard PIRT.

1. Define the issue that is driving the need for a PIRT
2. Define the specific objectives for the PIRT
3. Define the hardware and the scenario for the PIRT
4. Define the evaluation criterion
5. Identify, compile, and review the current knowledge base
6. Identify phenomena
7. Develop importance ranking for phenomena
8. Assess knowledge level for phenomena
9. Document PIRT results

Each of these steps is described below for this specific PIRT.

Step 1: Define the issue that is driving the need for a PIRT

The introduction of new accident tolerant fuel concepts into commercial power reactors will result in fuels with new design features in spent fuel pools and dry storage containers. Prior to these new fuel designs reaching spent fuel storage locations, it should be assessed if these fuel types may be stored according to existing safety practices or if new mitigating features must be applied. Unlike in-reactor safety analyses, there are no specified acceptable design limits put on the fuel cladding for spent fuel storage and transportation analyses. However, to certify that a dry storage system 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.

Step 2: Define the specific objectives for the PIRT

The outcome of the PIRT is to define the issues of safety significance for spent fuel storage of identified accident tolerant fuel concepts. Specifically, the PIRT will evaluate the impact of the relevant cladding or fuel property changes on the identified safety analyses (see Step 4) that must be performed for spent fuel pool storage, a dry storage system, or a transportation package.

Potentially, additional new damage and/or property changes will be identified by PIRT panel members and, if so, the same evaluation will be performed on these differences.

Step 3: Define the hardware and the scenario for the PIRT

The hardware and scenarios are described below.

Hardware

ATF concepts that were evaluated under this PIRT include:

- PWR and BWR fuel rods coated with a thin layer of chromium metal or chromium-containing ceramic
- PWR and BWR fuel rods with FeCrAl cladding
- PWR and BWR fuel rods with or without ATF cladding that have been irradiated to a rod-average burnup of 80 GWd/MTU
- PWR and BWR fuel rods with or without ATF cladding with an initial ^{235}U enrichment of up to 10%

Scenarios

<u>Wet Storage Conditions</u>

- Liquid water at atmospheric pressure with temperature between 30 and 60°C

<u>Short Term Loading to Dry Storage Conditions or Bare Fuel Transport Cask</u>
Normal and Off-Normal Conditions

- Vacuum and Inert gas between 0.1 and 0.7 MPa cask pressure
- Cladding temperature less than 400°C
- Less than 10 thermal cycles of less than 65°C variance
- Possibility of 1 thermal cycle greater than 65°C variance

<u>Dry Storage Conditions</u>
Normal Conditions and Off-Normal Conditions

- Vacuum and inert gas between 0.1 and 0.7 MPa cask pressure

- Cladding temperature less than 400°C

Accident conditions

- Cladding temperature less than 570°C

Transportation of Spent Nuclear Fuel stored less than 20 years[1]

Normal Conditions

- Air or inert gas between 0.1 and 0.7 MPa cask pressure
- Cladding temperature less than 400°C

Accident conditions

- Cladding temperature less than 570°C

Step 4: Define the evaluation criteria

To certify that a dry storage system 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. The following analyses are those identified in PNNL-30451 Rev.1 as those impacted by changes to the fuel design.

Analyses for Wet Storage:
(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
(5) prevention of significant reduction in fuel storage inventory under accident conditions
(6) criticality analysis

Analyses for Dry Storage and Transportation:
(1) structural evaluation
(2) thermal evaluation
(3) shielding evaluation
(4) criticality evaluation
(5) materials evaluation (including aging of materials)
(6) containment evaluation

The knowledge level for each relevant property change in the identified ATF designs should be sufficient to adequately perform these safety analyses. Property changes that have a high significance for safety calculations should have the greatest knowledge level.

Step 5: Identify, compile, and review the current knowledge base

PNNL recently produced the report, *Spent Fuel Storage and Transportation of Accident Tolerant Fuel Concepts: Cr-Coated Zirconium Alloy Cladding, FeCrAl Cladding, High Burnup and High Enrichment Fuel, PNNL-30451 Rev. 1*. This report gives a comprehensive overview on what has been done to date regarding these accident tolerant fuel concepts as they relate to spent fuel storage and transportation. All members of the PIRT panel have performed and documented an extensive review of this document. Additional information on high burnup and high enrichment fuel can be found in *Fuel Performance Considerations and Data Needs for Burnup above 62*

[1]Fuel stored more than 20 years may require additional analyses based on NUREG-2224 for high burnup fuel.

GWd/MTU: In-Reactor Performance, Storage, and Transportation of Spent Nuclear Fuel, PNNL-29368 and *Criticality Safety and Fuel Performance Considerations for Enrichment Above 5 Weight Percent in the Uranium Dioxide Fuel Cycle, PNNL-30088*.

Additionally, each PIRT panel member brings key expertise in one or more of the following seven areas:

- Tubing manufacturing and certification
- FeCrAl alloy development and testing
- Cr-coating development and testing
- Material properties measurement and testing
- Spent fuel design basis conditions: drying, transfer, storage, and transportation
- SNF cask and canister requirements
- Fuel and cladding degradation modes under long term storage

Step 6: Identify Phenomena

The report, *Spent Fuel Storage and Transportation of Accident Tolerant Fuel Concepts: Cr-Coated Zirconium Alloy Cladding, FeCrAl Cladding, High Burnup and High Enrichment Fuel, PNNL-30451 Rev. 1,* lists all the fuel and cladding property changes that are expected to impact spent fuel storage and transportation analyses for each ATF concept. These are listed in Tables A-1 through A-4.

If additional new fuel and cladding property changes are identified by PIRT panel members they could be added to this report and included in Tables A-1 through A-4, however, the panel members did not identify any new fuel or cladding property changes.

Step 7: Develop Importance Ranking for Phenomena

PIRT panel members met at PNNL from May 29-30, 2024 to present their review of the report, *Spent Fuel Storage and Transportation of Accident Tolerant Fuel Concepts: Cr-Coated Zirconium Alloy Cladding, FeCrAl Cladding, High Burnup and High Enrichment Fuel, PNNL-30451 Rev. 1* and to participate in the PIRT ranking activity. Following a discussion of each panel member’s comments on the report, panel members ranked each of the fuel and cladding property changes in Tables A-1 through A-4 as to the impact each has to adequately perform safety analyses for storage and transportation of spent fuel.

Each existing phenomenon was assigned an importance rank of "High," "Medium," or "Low," accompanied by a discussion and rationale for the assignment. The NRC definitions associated with each of these importance ranks follow:

Importance ranks and definitions

Importance Rank	Definition
Low (L)	Small influence on current design criteria
Medium (M)	Moderate influence on current design criteria
High (H)	Controlling influence on current design criteria

For new phenomena identified with any of the ATF concepts, each new phenomenon was assigned an importance rank of "High," "Medium," or "Low," based on the likelihood of this fuel or cladding property change to manifest itself during the scenarios identified accompanied by a discussion and rationale for the assignment. The definitions associated with each of these importance ranks follow:

Importance ranks and definitions

Importance Rank	**Definition**
Low (L)	Unlikely to manifest during scenarios
Medium (M)	Possible to manifest during scenarios
High (H)	Likely to manifest during scenarios

Step 8: Assess Knowledge Level for Phenomena

Panel members assessed and ranked the current knowledge level for applicable phenomenon in each PIRT table. High, medium, and low designations were assigned to reflect knowledge levels and adequacy of data and analytical tools used to characterize the phenomena, using the NRC-supplied definitions shown as follows.

Knowledge levels and definitions

Knowledge level	**Definition**
Low (L)	Unknown: 0-30% of complete knowledge and understanding
Medium (M)	Partially known: 30-70% of complete knowledge and understanding
High (H)	Known: Approximately 70-100% of complete knowledge and understanding

Step 9: Documentation of the PIRT

The lists and tables generated at the PIRT panel meeting document the discussions of phenomena identification plus the importance and knowledge level rankings with accompanying rationales. These lists and tables were used to generate charts to document both the collective and individual member assessments. In cases where the "collective assessment" or averaged result differed significantly from that of an individual panel member, the "minority view" was noted in the "rationale" column of the table. Further descriptions of the individual assessments and rationales were collected in the panel members' individual charts, which were generated prior to the discussion by the panel.

PNNL has produced a revision to the report, *Spent Fuel Storage and Transportation of Accident Tolerant Fuel Concepts: Cr-Coated Zirconium Alloy Cladding, FeCrAl Cladding, High Burnup and High Enrichment Fuel, PNNL-30451 Rev. 1*. This revision addresses comments provided by panel members and includes this appendix documenting the results of the PIRT process discussed above.

Tables A-1 to A-4 document the average of the rankings given by the panelists.

Table A.1: Cr-Coated Zr-alloy cladding property changes that could impact spent fuel storage and transportation analyses.

ID#	Cladding Property Change	Comments	Importance	Rationale	Knowledge Level	Rationale
Changes following irradiation relative to unirradiated conditions						
1	Change in yield stress	Secondary to confirming ductility	M	Need for SNF License	M	LTA Tests
2	Change in ductility		M	Need for SNF License	M	LTA Tests
3	Change in fatigue life	Secondary to confirming ductility	M	Need for SNF License	M	LTA Tests
4	Coating cracking or delamination		M		M	Not observed to date
5	Cr-Zr interdiffusion		M	Temperature too low	M	
6	Radiation effects on Cr		M	Small amount of Cr	M	
7	Galvanic corrosion		M	Small reaction zone, Need for SNF License	L	
Aging-related damage mechanisms						
8	Embrittlement		M		M	
9	Delayed hydride cracking		L		M	Depends on underlying cladding
10	Thermal and athermal creep		M		M	Depends on underlying cladding

Table A.2: FeCrAl cladding property changes that could impact spent fuel storage and transportation analyses.

ID#	Cladding Property Change	Comments	Importance	Rationale	Knowledge Level	Rationale
Changes following irradiation relative to unirradiated conditions						
11	Change in yield stress		H	Need for SNF License	L	Emerging data
12	Change in ductility		H	Need for SNF License	L	Emerging data
13	Change in fatigue life		M	Need for SNF License	L	Emerging data
14	Radiation effects on FeCrAl		H		L	Emerging data
15	Galvanic corrosion		M		M	Emerging data
Aging-related damage mechanisms						
16	Embrittlement		H		L	Emerging data
17	Delayed hydride cracking		M	Not a hydride former	M	Not a hydride former
18	Thermal and athermal creep		H	Resistant to creep	L	Emerging data

Table A.3: High enrichment fuel property changes that could impact spent fuel storage and transportation analyses.

ID#	Property Change	Comments	Importance	Rationale	Knowledge Level	Rationale
19	Criticality concerns		M	Significant impact on SNFpool loading	M	Current models likely can be extrapolated, some benchmarks exist
20	Source Term		M		H	Current models likely can be extrapolated, some benchmarks exist

Table A.4: High burnup fuel and cladding property changes that could impact spent fuel storage and transportation analyses.

ID#	Property Change	Comments	Importance	Rationale	Knowledge Level	Rationale
Changes following continued irradiation beyond 62 GWd/MTU						
21	Change in yield stress		M		M	Effect appears to saturate
22	Change in ductility		M		M	Effect appears to saturate
23	Change in fatigue life		M		M	Effect appears to saturate
24	Source term		M	Fraction available for release higher at high burnup	M	Can calculate
25	Decay heat		M		M	Simulation tools need uncertainty quantification
26	Fissile content		H		M	Can calculate
Aging-related damage mechanisms						
27	Embrittlement		H		M	
28	Delayed hydride cracking		M		M	
29	Thermal and athermal creep		H		M	

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