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Supplemental Guidance Regarding the Chromium-Coated Zirconium Alloy Fuel Cladding Accident Tolerant Fuel Concept

Interim Staff Guidance

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INTERIM STAFF GUIDANCE

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PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) staff is providing this interim staff guidance (ISG) to facilitate the staff's understanding of the in-reactor phenomena important to safety for the chromium-coated zirconium alloy fuel cladding concept being pursued by several U.S. fuel vendors as part of the U.S. Department of Energy's accident tolerant fuel (ATF) program.

BACKGROUND

This ISG is intended to provide guidance for the NRC staff reviewing applications involving fuel products with chromium-coated zirconium alloy cladding. For coated claddings of this type, a phenomena identification and ranking table (PIRT) was generated for the NRC by Pacific Northwest National Laboratory; the guidance provided in this ISG extensively references the resulting PIRT report, "Degradation and Failure Phenomena of Accident Tolerant Fuel Concepts: Chromium Coated Zirconium Alloy Cladding," issued June 2019 (Reference 1). The suggested cladding properties specified acceptable fuel design limits and new failure mechanisms sections from the PIRT report are replicated in Appendices B and C. These appendices supersede Sections 5.1 and 5.2 of the PIRT report.

This ISG is not intended as standalone review guidance but instead supplements NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) (Reference 2), Section 4.2, "Fuel System Design," and discusses the potential impact of coated claddings on reviews performed under SRP Section 4.3, "Nuclear Design," SRP Section 4.4, "Thermal and Hydraulic Design," and SRP Chapter 15, "Transient and Accident Analysis." In addition to the guidance provided in this ISG, reviewers of coated cladding applications should familiarize themselves with the PIRT report and with the relevant sections of the SRP.

The PIRT report and this ISG focus primarily on metallic chromium coatings applied to a zirconium alloy base metal, with some additional discussion that is applicable to chromium-based ceramic coatings. Reviewers of submittals on ceramic chromium-coated zirconium alloy claddings should carefully read the PIRT report to determine its applicability to the review.
This ISG does not apply to reviews of fuel products other than metallic or ceramic chromium-based coatings on a zirconium alloy substrate. The use of the term "coated cladding" in this document refers specifically to chromium-based coatings.

RATIONALE

The current review guidance in the SRP assumes the use of uranium dioxide fuel pellets contained within zirconium alloy-based fuel cladding and is targeted to specific degradation and failure modes associated with that material. Based on this fact, along with the aggressive development timelines of U.S. Department of Energy and industry ATF programs, the NRC staff proactively developed a plan, "Project Plan to Prepare the U.S. Nuclear Regulatory Commission for Efficient and Effective Licensing of Accident Tolerant Fuels," issued September 2018 (ATF Project Plan) (Reference 3), to outline a preparation strategy for ensuring staff readiness to perform timely licensing reviews. This ISG will serve as the concept-specific licensing roadmap for chromium-coated zirconium alloy cladding that is detailed as part of the strategy included in the ATF Project Plan.

APPLICABILITY

This guidance applies to the following entities:

- All holders of and applicants for an operating license or construction permit for a nuclear power reactor under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," except those that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

- All holders of and applicants for a power reactor early site permit, combined license, standard design approval, or manufacturing license under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." All applicants for a standard design certification, including such applicants after initial issuance of a design certification rule.

- All holders of and applicants for a power reactor early site permit, combined license, standard design certification, standard design approval, or manufacturing license referencing a small modular reactor design under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." Small modular reactors are defined using the International Atomic Energy Agency definition of small and medium-sized reactors with an electrical output of less than 700 megawatts.

- All contractors and vendors that supply basic components to NRC licensees under 10 CFR Part 50 or 10 CFR Part 52.

GUIDANCE

The information contained in Appendix A to this ISG provides supplemental guidance to Chapters 4 and 15 of the SRP for NRC reviewers. The foundation for this additional guidance is the chromium-coated cladding PIRT report.

IMPLEMENTATION

The NRC staff will use the information contained in this ISG to review degradation and failure mechanisms for chromium-coated zirconium alloy fuel cladding such that the staff can assess their impact on the acceptance criteria contained in SRP Sections 4.2, 4.3, and 4.4 and SRP Chapter 15, and ultimately, the applicant or licensees compliance with applicable regulatory requirements.

BACKFITTING AND ISSUE FINALITY DISCUSSION

Issuance of this ISG does not constitute a backfit as defined in 10 CFR 50.109(a)(1) and is not otherwise inconsistent with the issue finality provisions in 10 CFR Part 52. Thus, the NRC staff did not prepare a backfit analysis for the issuance of this ISG.

The NRC's position is based upon the following considerations:

1. The ISG positions do not constitute backfitting, inasmuch as the ISG is guidance directed to the NRC staff with respect to its regulatory responsibilities. The ISG provides interim guidance to the staff on how to review certain requests. Changes in guidance intended for use by only the staff are not matters that constitute backfitting as that term is defined in 10 CFR 50.109 or involve the issue finality provisions of 10 CFR Part 52.

2. Backfitting and issue finality—with certain exceptions discussed in this section—do not apply to current or future applicants. Applicants and potential applicants are not, with certain exceptions, the subject of either the Backfit Rule or any issue finality provisions under 10 CFR Part 52. This is because neither the Backfit Rule nor the issue finality provisions of 10 CFR Part 52 were intended to apply to every NRC action that substantially changes the expectations of current and future applicants. The exceptions to the general principle are applicable whenever a 10 CFR Part 50 operating license applicant references a construction permit or a 10 CFR Part 52 combined license applicant references a license (e.g., an early site permit) and/or an NRC regulatory approval (e.g., a design certification rule) for which specified issue finality provisions apply. The NRC staff does not currently intend to impose the positions represented in this ISG in a manner that constitutes backfitting or is inconsistent with any issue finality provision of 10 CFR Part 52. If in the future the NRC staff seeks to impose positions stated in this ISG in a manner that would constitute backfitting or be inconsistent with these issue finality provisions, the NRC staff must make the requisite showing as set forth in the Backfit Rule or address the regulatory criteria set forth in the applicable issue finality provision, as applicable, that would allow the staff to impose the position.

3. The NRC staff has no intention to impose the ISG positions on existing nuclear power plant licensees either now or in the future (absent a voluntary request for a change from the licensee). The staff does not intend to impose or apply the positions described in the ISG to existing (already issued) licenses (e.g., operating licenses and combined licenses). Hence, the issuance of this ISG—even if considered guidance subject to the

Backfit Rule or the issue finality provisions in 10 CFR Part 52— would not need to be evaluated as if it were a backfit or as being inconsistent with issue finality provisions. If, in the future, the NRC staff seeks to impose a position in the ISG on holders of already issued licenses in a manner that would constitute backfitting or does not provide issue finality as described in the applicable issue finality provision, then the staff must make a showing as set forth in the Backfit Rule or address the criteria set forth in the applicable issue finality provision, as applicable, that would allow the staff to impose the position.

CONGRESSIONAL REVIEW ACT

This ISG is a rule as defined in the Congressional Review Act (5 U.S.C. 801-808). However, the Office of Management and Budget has not found it to be a major rule as defined in the Congressional Review Act.

FINAL RESOLUTION

By 2025, this information is expected to be incorporated into SRP Chapter 4, “Reactor,” and SRP Chapter 15. Following the transition of this guidance to the SRP, this ISG will be closed.

APPENDICES

A. Supplemental Guidance for Standard Review Plan Chapters 4 and 15
B. Cladding Material Property Correlations
C. Specified Acceptable Fuel Design Limits
D. References
E. Analysis of Public Comment Resolution

APPENDIX A

Supplemental Guidance for Standard Review Plan Chapters 4 and 15

NUREG-0800—Chapter 4, Section 4.2, "Fuel System Design"
For reviews of new fuel products where the only change from an existing approved fuel design that utilizes zirconium alloy cladding is the adoption of chromium-coated cladding, the licensing of a new cladding alloy can be used as a model. While NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Chapter 4, "Reactor," Section 4.2, "Fuel System Design," covers additional requirements for the review of complete fuel systems, cladding reviews cover these three areas:

(1) definition of specified acceptable fuel design limits (SAFDLs) for new cladding
(2) material property correlations to be used in codes to ensure the new cladding satisfies the SAFDLs
(3) any changes that must be made to existing methodologies to accommodate the new cladding

These topics will be discussed in more detail in the following sections.

While chromium coatings may only be a fraction of the thickness of the base cladding, they are designed to provide the following benefits over uncoated cladding:

- harder surface
  - improves cladding fretting performance and wear resistance
- negligible oxidation during normal operation
  - protects zirconium cladding from oxidation
  - protects zirconium cladding from hydrogen uptake
- improved high-temperature steam oxidation kinetics
  - reduced rate of corrosion and heat of oxidation
  - protects zirconium cladding from oxidation
  - reduced hydrogen liberation
- improved high-temperature strength

This interim staff guidance (ISG) does not attempt to set standards for the review of any credit or benefit applicants may request by demonstrating these improvements, as strategies for licensing these coated cladding concepts have not yet been submitted to the U.S. Nuclear Regulatory Commission (NRC). The reviewer of any coated cladding must, therefore, evaluate any proposed property improvements against the data provided by the applicant; the commentary in the phenomena identification and ranking table (PIRT) report generated for the NRC by Pacific Northwest National Laboratory, "Degradation and Failure Phenomena of Accident Tolerant Fuel Concepts: Chromium Coated Zirconium Alloy Cladding," issued June 2019 (Reference 1); and the guidance in this ISG. The reviewer must also evaluate whether the data provided support the full operating domain for the fuel and place appropriate limitations and conditions when necessary. Appropriately conservative assumptions based on the current understanding of the phenomena could be defined if explicit data does not exist. Finally, the NRC staff should ensure that, if an applicant wishes to take credit for coating behavior up to a certain burnup or during certain accident conditions, adherence of that coating to the substrate has been justified for the full operating domain.

Definition of Specified Acceptable Fuel Design Limits for New Cladding
The SAFDLs mentioned in SRP Chapter 4, Section 4.2, under "SRP Acceptance Criteria, Design Bases," can be broadly separated into three general categories:

(1) SAFDLs related to fuel assembly performance that are typically addressed by simple calculation, manufacturing controls, and historical data
(2) SAFDLs related to fuel rod performance that are typically addressed for normal operation and anticipated operational occurrences (AOOs) using a thermal mechanical code
(3) SAFDLs related to fuel rod performance that are typically addressed for accident conditions using a system analysis code with initial conditions provided by a thermal mechanical code

Each SAFDL listed in SRP Section 4.2 is included in Table 5.2 of the PIRT report and described in further detail in Appendix C to this ISG. These sections detail the expected and potential impact of the coatings on each SAFDL.

The reviewer should ensure that submittals for chromium-coated cladding address each of the SAFDLs where the PIRT report notes that additional concerns may exist. Table 5.3 of the PIRT report contains a summary of tests that could be performed to justify SAFDLs; however, the NRC does not require any specific testing to be performed, and applicants may be able to sufficiently address a SAFDL in an alternative fashion. If a topical report or other generic submittal is under review, some of the SAFDLs may be left to address in application-specific reviews, as plants apply for license amendments to load batch quantities of fuel with coated cladding. If this is the case, these should be noted in the safety evaluation for the application for the coated cladding product, typically as a condition or limitation.

Potential new damage mechanisms have been identified in Appendix C, Section C.4, of this ISG. The reviewer should ensure that these mechanisms have been ruled out sufficiently by the applicant for the domain approved by the NRC, that existing SAFDLs already protect against the mechanisms, or that new SAFDLs have been developed to protect against them.

Based upon an investigation of available performance testing and known data gaps, Section 6.4.2 of the PIRT report identified several performance concerns for chromium-coated zirconium alloys. The reviewer should ensure that these performance concerns have been ruled out sufficiently by the applicant for the domain approved by the NRC, that existing SAFDLs already protect against the damage mechanisms, or that new SAFDLs have been developed to protect against them.

The combination of prescriptive analytical limits in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," namely, 2,200 degrees Fahrenheit (F) (1,204 degrees Celsius (C)) peak cladding temperature (PCT) and 17-percent equivalent cladding reacted maximum local oxidation, were established to preserve postquench ductility during a postulated loss-of-coolant accident (LOCA). These analytical limits for postquench ductility were based on ring compression tests conducted on zirconium cladding segments exposed to various levels of high-temperature steam oxidation. The point of nil-ductility was predicted by integrating time-at-temperature using the Baker-Just weight gain correlation, even though it was understood that cladding embrittlement is governed by oxygen diffusion into the base metal and is not directly related to the growth of the oxide layer.

Differences in oxidation kinetics between zirconium-based cladding and chromium-coated cladding change the relationship between oxygen diffusion and oxide growth. This issue is further complicated within the burst region where cladding inner diameter oxidation is based on zirconium alloy kinetics and cladding outer diameter oxidation would be based on chromium coating kinetics. Hence, the applicability of the 17-percent equivalent cladding reacted analytical limit and, more generally, the use of maximum local oxidation as a surrogate SAFDL for cladding embrittlement is questionable.

The NRC staff should ensure that, if the applicant elects to ignore the potential benefits expected with chromium coatings and continues to use the existing 10 CFR 50.46 analytical limits, then supporting evidence has been provided to demonstrate residual ductility of the coated cladding up to these analytical limits.

The NRC staff should ensure that, if the applicant elects to develop an alternative set of analytical limits, sufficient research and testing has been performed to justify those limits. NUREG/CR-7219, “Cladding Behavior during Postulated Loss-of-Coolant Accidents,” issued July 2016 (Reference 4), identifies new degradation mechanisms that need to be considered. Furthermore, additional degradation mechanisms exist for zirconium alloy cladding above 2,200 degrees F (1,204 degrees C) and must be considered. Finally, burst node survival would need to be considered as the relationship between cladding embrittlement and burst node fracture toughness would change.

Section 4 of the PIRT report describes the zirconium-chromium phase diagram. The formation of a liquid phase at the eutectic point shown at 2,430 degrees F (1,332 degrees C), which is well below the melting point of either the chromium coating or the zirconium alloy substrate, is another concern with respect to establishing a PCT SAFDL. The reviewer should ensure that the applicant provides a sufficient empirical database to define performance metrics and analytical limits that preserve acceptable fuel rod behavior under LOCA conditions.

As described in Section 6.2.2 of the PIRT report, chromium coating may also impact the fuel rod ballooning characteristics under accident conditions. While no regulatory limits are currently defined to limit the extent of ballooning or the size of the rupture opening, concerns related to fuel fragmentation, relocation, and dispersal may warrant future SAFDLs for fuel rod burnup extensions beyond rod-average values of 62 gigawatt days per metric ton unit.

<u>Material Property Correlations to Ensure Specified Acceptable Fuel Design Limits are Met</u>
Appendix B to this ISG provides a list of cladding material properties that are typically needed to adequately model fuel system response based on development and qualification of the NRC’s independent fuel performance code, FRAPCON, and previously approved thermal-mechanical codes. These property correlations are then used by the thermal-mechanical codes to demonstrate compliance with the SAFDLs. This approval may come at the topical report review stage, if an applicant demonstrates that the SAFDL is satisfied for the entire design and operating domain, or a methodology may be approved to be used for each licensee that wishes to load the fuel.

The PIRT report also suggests two paths that an applicant may take to analyze each property: treating the cladding and coating as separate layers and treating the cladding and coating together as a composite material. A subset of the composite material strategy may be to demonstrate that the coating will have a negligible impact on a property and to use the property of the underlying substrate. Any of these paths may be appropriate provided sufficient justification from the applicant, and a variety of these strategies may be used to disposition the various properties.

Appendix B to this ISG details each of the 12 properties identified in the report. Applicants intending to use chromium-coated zirconium alloy cladding should address all these properties. If the applicant assumes that the coated cladding will behave the same as the underlying substrate without supporting evidence that the property is unchanged, this assumption should be demonstrated to be conservative for normal operation, AOOs, and accidents described in SRP Chapter 15, “Transient and Accident Analysis.”

Changes to Existing Codes and Methodologies

New cladding properties need to be properly modeled using computer codes to assess the performance of the coated cladding. If, for a given property, the coated cladding is treated as a composite material, changes to the codes and methods may not be needed beyond updates to the property correlations; however, if the cladding is treated as a separate layer, codes may need to be modified to account for the additional layer as well as interface effects.

Regardless of the changes made to address the coating, the codes and methods must be validated. Section 5.3.1 of the PIRT report identifies five areas where validation is critical:

(1) fuel temperature
(2) fission gas release
(3) rod internal pressure and void volume
(4) cladding oxide thickness
(5) cladding permanent hoop strain following a power ramp

Sections 5.3.1.1 through 5.3.1.5 of the PIRT report go into each of these in more detail. Table 5.4 of the PIRT report provides a list of test data that could be used in code assessment.

The methodology for performing the fuel system safety analysis consists of the following pieces:

- identification of functional requirements for the fuel and assembly
- identification of limits for each functional requirement
- identification of the code or other approach that will be used to assess performance against a functional requirement
- identification of the approach to demonstrate a high level of confidence that the design will not exceed functional requirements:
  - selection of power histories to be considered
  - identification of uncertainties in operational parameters
  - identification of fabrication uncertainties
  - identification of modeling uncertainties
  - approach to quantify an upper tolerance level based on identified uncertainties

The identification of functional requirements for the fuel and assembly and the limits for each is satisfied by the selection of appropriate SAFDLs. There have been new damage mechanisms identified in Appendix C, Section C.4, to this ISG that should be implicitly handled via existing SAFDLs and considered in the development of those SAFDL limits. Alternatively, the methodology may be modified to explicitly address these mechanisms through new functional requirements and limits.

The material property updates and the code assessment have been discussed. No further methodology change is anticipated as far as the use of codes is considered. The identification of operational parameters, such as rod power and coolant flow rate, are not expected to be impacted by the implementation of chromium-coated zirconium alloy cladding. Any further changes to the code or operational parameters should be evaluated during the review of the application.

The identification of fabrication uncertainties (including uncertainties in coating parameters) will be taken from uncertainty specifications on the drawings or from manufacturing data. Although specific values may change, the general approach for obtaining these values is not expected to change. Any changes to this general approach should be dispositioned sufficiently in the application.

Modeling uncertainties should be identified during the implementation and assessment of new material properties in codes. Comparing property data to correlations and code predictions to measurements should allow for the appropriate development of acceptable modeling uncertainties. The application should identify modeling uncertainties and explain how the uncertainties were determined.

Existing approaches to calculate upper tolerance levels are robust and should be acceptable to perform these calculations for chromium-coated zirconium alloy cladding assuming that the activities discussed above are rigorously performed. Any changes to these approaches should be dispositioned in the application.

NUREG-0800—Chapter 4, Section 4.3, "Nuclear Design"

SRP Section 4.3, "Nuclear Design," covers the review of the nuclear design of fuel assemblies, control systems, and the reactor core. The reviewer of coated cladding in this area should ensure that the cross-sections generated for the fuel include the effect of the coating.

NUREG-0800—Chapter 4, Section 4.4, "Thermal and Hydraulic Design"

SRP Section 4.4, "Thermal and Hydraulic Design," covers the thermal-hydraulic design for fuel assemblies, including critical heat flux (CHF) or critical power (CP) correlations. As discussed in Appendix C to this ISG, the impacts of coating on fuel thermal-hydraulics are expected to be minimal and constrained mostly to the effect of the coating on cladding surface conditions, which could impact boiling crisis behavior. Existing CHF or CP correlations are expected to continue to be applicable for chromium-coated zirconium alloy cladding, provided surface conditions are similar to current zirconium cladding surface conditions. The reviewer of a coated cladding submittal in this area should ensure that applicants appropriately disposition the following three areas, with justification:

(1) whether changes to hydraulic diameter due to the coating thickness affect the applicability of the CHF or CP correlation
(2) whether the addition of a chromium coating, including consideration of the effects of surface roughness, changes the fuel rod boiling crisis behavior
(3) for boiling-water reactor applications, whether the addition of a chromium coating affects the rewet temperature following dryout (i.e., T_{min})

Coating degradation mechanisms, as discussed in Appendix C to this ISG, may affect the cladding thermal-hydraulic characteristics. This is particularly true for coating cracking and delamination, which have the potential to change the flow or boiling regime, or both, near the cladding surface. Coating cracking and delamination may also result in nucleation sites that have the potential to cause hot spots and localized corrosion. The reviewer should ensure that these effects are appropriately accounted for or that coating degradation is otherwise prevented.

NUREG-0800—Chapter 15, "Transient and Accident Analysis"

Updated Final Safety Analysis Report (UFSAR) Chapter 15 provides demonstration that the technical specification limiting conditions of operation, technical specification limiting safety system setting, and reactor protection system and engineered safety features actuation system are capable of performing their safety functions, ensuring fuel does not exceed SAFDLs during normal operation and AOOs, and mitigating the consequences of postulated accidents. SRP Chapter 15 provides guidance for the review of these safety analyses.

As described above for SRP Section 4.2, chromium coatings may impact the cladding's material properties and mechanical and thermal behavior. These changes should be incorporated, where necessary, in the fuel rod thermal-mechanical models, which provide important fuel parameters and initial conditions to the reactor core neutronic (SRP Section 4.3) and thermal-hydraulic (SRP Section 4.4) models and nuclear steam supply system codes used in the Chapter 15 demonstration.

Chromium coatings may have an impact on the cladding initial condition and mechanical properties at the onset of AOOs and postulated accidents. Depending on the oxidation characteristics of the chromium-coated cladding, the load-bearing zirconium cladding may experience little to no corrosion-related wall thinning and potentially less hydrogen uptake. This reduces cladding stress and preserves beneficial ductility prior to a transient event. AOO overpower cladding strain analytical limits, reactivity-initiated accident pellet-cladding mechanical interaction (RIA PCMI) cladding failure thresholds, and LOCA PCT and integral time-at-temperature analytical limits (see the rulemaking on 10 CFR 50.46c, "Requirements for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors") are all influenced by initial cladding hydrogen content. Hence, any reduction in hydrogen uptake provided by the chromium coating would have a beneficial impact for these transient events.

As described above for SRP Section 4.2, the addition of a chromium coating may necessitate changes to existing SAFDLs or require new SAFDLs. These impacts would need to be incorporated into the Chapter 15 demonstration.

Any inherent impacts of the chromium coating that potentially impact the fuel rod initial conditions (e.g., gap conductivity, stored energy) should be captured in the fuel rod performance models (SRP Section 4.2). Similarly, potential impacts on core reactivity should be captured in the reactor physics models (SRP Section 4.3). Finally, potential impacts on the rod-to-coolant heat transfer, CHF correlation, and safety limits should be captured in core thermal-hydraulic models (SRP Section 4.4). To capture benefits in one of these areas, the coating should be explicitly considered; however, as discussed above, applicants may be able to demonstrate that any negative impacts of the chromium coating in these areas are negligible, and the coating could then be ignored or lumped into the modeling of the base zirconium alloy cladding.

For many AOOs and postulated accidents, the presence of a thin chromium coating is not expected to play a significant role in the fuel rod's performance during the transient nor influence the overall accident progression. For example, safety analyses in the Pressurized-Water Reactor (PWR) UFSAR, Section 15.2, demonstrate that overpressure

protection systems (e.g., main steam safety valves, pressurizer safety valves) protect the integrity of the reactor pressure boundary during decrease in secondary heat removal AOOs and postulated accidents. For this demonstration, the fuel rods are not modeled in specific detail, and the presence of a thin chromium coating will have no impact.
For AOOs and postulated accidents involving an increase in global or local core power (e.g., PWR excess steam demand or main steamline break, boiling-water reactor (BWR) loss of feedwater heater or turbine trip, PWR inadvertent bank withdrawal or control rod ejection, and BWR rod withdrawal error or blade drop), the presence of a brittle chromium coating may act as a nucleation site for crack propagation into the base zirconium cladding. Alternatively, a thin ductile chromium coating would likely not initiate crack propagation. A review of coated cladding products under SRP Section 4.2 should evaluate the potential impact of the chromium coating on the cladding's strain loading capability and whether a revised AOO overpower cladding strain failure threshold (e.g., 1.0 percent permanent) or revised RIA PCMI cladding failure thresholds are needed. Nevertheless, the presence of the chromium coating will not change the systems' response to the initiating event.

For AOOs and postulated accidents involving a decrease in reactor coolant flow (e.g., loss of alternating current power and PWR reactor coolant pump locked rotor), the presence of the chromium coating will not change the systems' response to the initiating event.

During a postulated LOCA, the design features of the chromium coating are expected to have an impact on the fuel rod's performance during the transient. During the LOCA, multiple phenomena may be affected, such as the following:

- heat of oxidation
- oxygen ingress to the cladding outside diameter
- hydrogen-enhanced beta-layer embrittlement
- plastic strains
-

As a result of these improvements, chromium-coated fuel rod structural integrity and coolable geometry may be more readily maintained than with a typical, uncoated zirconium-alloy-based cladding.

While it is not expected that the chromium coating will improve fuel rod cladding-to-coolant heat transfer, LOCA core temperatures may be reduced due to the reduction in heat addition from cladding oxidation. These lower temperatures, combined with improved oxidation kinetics, may reduce core-wide inventories of liberated hydrogen.

The reviewer should ensure that the impact of chromium coating on each of the SRP Chapter 15 AOOs and postulated accidents has been properly assessed. The scope of work needed to complete the SRP Chapter 15 demonstration may increase if the chromium coating negatively impacts fuel temperature, fuel rod cladding-to-coolant heat transfer, or CHF correlation or if the application is accompanied with an increase in fuel rod peaking factors, cycle length, allowable fuel rod burnup, or increased uranium-235 enrichment.

APPENDIX B

Cladding Material Property Correlations

The following cladding material properties are typically needed to perform fuel thermal-mechanical analysis of nuclear fuel with zirconium (Zr) alloy cladding under normal conditions and anticipated operational occurrences (AOOs):

- thermal conductivity
- thermal expansion
- emissivity
- enthalpy and specific heat
- elastic modulus
- yield stress
- thermal and irradiation creep rate (function of stress, temperature, and fast neutron flux)
- axial irradiation growth
- oxidation rate
- hydrogen pickup

The following additional material properties are typically needed to perform fuel-mechanical analysis of nuclear fuel under accident conditions based on the development and qualification of the U.S. Nuclear Regulatory Commission (NRC) transient fuel performance code, FRAPTRAN (Geelhood et al., 2016 (Reference 6)):

- high-temperature ballooning behavior
- high-temperature 1,472-2,192 degrees Fahrenheit (F) (800-1,200 degrees Celsius (C)) steam oxidation rate

If the first approach discussed above (to independently model the coating and the cladding) is taken, then each of the above properties and the impact of irradiation on these should be determined as well as the interface behavior. If the second approach discussed above (to model the cladding and the coating as a composite material) is taken, then the impact of the coating on the base metal should be determined. The following discussion provides information on the potential impact of a metallic or ceramic coating on the base metal.

Each of these properties are discussed in the following sections as they relate to chromium (Cr)-coated Zr cladding. The type of data that are typically used to justify each material property model will be stated. Differences in applicant-specific processes could impact the material properties of the coated cladding. Therefore, the applicant should provide sufficient data or other technical justification based on its specific cladding product to justify material property models. There is a growing body of generic data from various Cr-coated Zr samples, as discussed in Section 6.0 of the phenomena identification and ranking table (PIRT) report generated for the NRC by Pacific Northwest National Laboratory (PNNL), “Degradation and Failure Phenomena of Accident Tolerant Fuel Concepts: Chromium Coated Zirconium Alloy Cladding,” issued June 2019 (Reference 1). These data are important because they provide the NRC staff a baseline of what to expect when reviewing an application and claims of large deviations from the generic database may indicate an area for a more detailed review. In the following discussion, it should be noted that the coatings under consideration are 5 to 30 microns thick on cladding that is 500 to 700 microns thick. Table 5.1 in the PIRT report provides a summary of the tests that could be performed to quantify the material properties discussed below. Use of this table should be further informed by the remainder of this appendix.

B.1: Thermal Conductivity

Zirconium Alloy Cladding

Cladding thermal conductivity is not expected to change significantly with irradiation, based on the currently available data. Typically, heat transfer in a metal is due to electronic heat transfer, which is not significantly impacted by lattice damage done by fast neutron irradiation. No change in thermal conductivity with irradiation is used in FRAPCON (Luscher, Geelhood, & Porter, 2015, (Reference 7)). Thermal conductivity data as a function of temperature from unirradiated samples have typically been used to develop cladding thermal conductivity correlations.

Chromium-Coated Zirconium

Either an effective thermal conductivity for the coated cladding could be developed, or a method for combining the thermal conductivity from the base metal and the coating could be described. The thermal conductivity of Cr metal is not expected to be strongly impacted by irradiation. The thermal conductivity of a Cr-based ceramic may be impacted by irradiation. It is possible that the overall cladding thermal conductivity may not be strongly impacted by this as the coating is expected to be relatively thin. However, a ceramic coating will have a greater impact as the thermal conductivity of ceramics is generally low. This would be similar to the treatment of the zirconium dioxide that evolves on the surface of the Zr alloy cladding.

B.2: Thermal Expansion

Zirconium Alloy Cladding

Cladding thermal expansion is not expected to change significantly with irradiation, based on the currently available data. Thermal expansion is caused by crystal lattice expansion and does not change much with the introduction of dislocations from fast neutron irradiation. No change in thermal expansion with irradiation is used in FRAPCON (Reference 7). Thermal expansion data as a function of temperature from unirradiated samples have typically been used to develop cladding thermal expansion correlations.

Chromium-Coated Zirconium

Typically, the thermal expansion of a coated part will be the same as that of an uncoated part if the coating is relatively thin. However, thermal expansion data from representative cladding tubes would be useful to justify the correlation and to demonstrate that there has not been a change in behavior with the coating due to thermal expansion mismatch between the substrate and the coating. Thermal expansion mismatch between a coating and substrate typically results in plastic strain in the thin coating, which is weaker than the substrate because of its thickness. This is particularly true for the Zr-Cr system since the textured hexagonal crystal structure leads to different thermal expansion in different directions, while the cubic Cr or Cr-ceramic coatings will have similar thermal expansion in all directions.

Many ceramics have a limited strain capability. A ceramic coating with a significant thermal expansion mismatch strain may exhibit cracking upon heating and cooling due to the inability of that coating to tolerate plastic strain.

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

B.3: Emissivity

Zirconium Alloy Cladding

Emissivity on the outside of the cladding is important to calculate radiative heat transfer, which can be dominant in very-high-temperature transients as well as steam-only heat transfer. Some design-basis accidents may be influenced by emissivity, such as a small-break loss-of-coolant accident (LOCA). The emissivity is impacted by the surface conditions, including any oxide on the surface of the cladding. For a Zr alloy, it typically increases with oxidation until saturation as the oxide becomes opaque.

Chromium-Coated Zirconium

Some system codes and accident analysis codes account for cladding surface emissivity and radiation heat transfer from fuel rods to other reactor core components, as well as radiation heat transfer to steam. In general, shinier surfaces have lower emissivity and therefore lower radiative heat transfer. As chromium coatings resist oxidation and retain their surface appearance, it is likely that the coating will negatively impact cladding temperature for transients where radiation to steam is the dominant mode of heat transfer. Therefore, it is likely necessary to revise the outer surface emissivity for accident analyses. This would apply equally to metallic and ceramic coatings (Seshadri, Philips, & Shirvan, 2018, (Reference 8))

Because the current coatings are on the outer surface, it would be acceptable to retain the emissivity used for an uncoated Zr alloy tube for the inner tube surface during thermal-mechanical analysis.

B.4: Enthalpy and Specific Heat

Zirconium-Alloy Cladding

Cladding enthalpy and specific heat are not expected to change significantly with irradiation, based on the currently available data. The specific heat of a material is dependent on the composition and the crystal structure and does not change much with the introduction of dislocations from fast neutron irradiation. No change in enthalpy or specific heat with irradiation is used in FRAPCON (Reference 7). Enthalpy and specific heat data as a function of temperature from unirradiated samples would be useful to develop cladding enthalpy and specific heat correlations.

Chromium-Coated Zirconium

Either an effective enthalpy and specific heat for the coated cladding could be developed, or a method for combining the enthalpy and specific heat from the base metal and the coating could be described. Cladding enthalpy and specific heat are only needed for transient fuel performance analysis and for the calculation of stored energy. This would apply equally to metallic and ceramic coatings.

B.5: Elastic Modulus

Zirconium-Alloy Cladding

Cladding elastic modulus has been observed to be a weak function of fast neutron fluence (proportional to fuel burnup) (Geelhood et al., 2008, (Reference 9)). Not all applicants include a fluence dependence, but if one is included, then temperature-dependent data from irradiated and unirradiated coated tubes would be useful to justify the correlation used.

Chromium-Coated Zirconium

Recent data on unirradiated Cr-coated Zr indicate the elastic modulus of a coated part will be the same as that of an uncoated part (Brachet et al., 2017, (Reference 10); Kim et al., 2015, (Reference 11); Shahin et al., 2018, (Reference 12)). Typically, ceramic materials are stiffer (greater elastic modulus) than metallic materials. However, for thin coatings, the enhanced stiffness of the coating is not expected to strongly impact the overall stiffness of the substrate. Nano-indentation could be used to evaluate the elastic modulus of the coating.

B.6: Yield Stress and Ultimate Tensile Stress

Zirconium-Alloy Cladding

Methods in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code allow for the use of either yield stress or ultimate tensile stress for the evaluation of cladding. Cladding yield stress and ultimate tensile stress have been observed to be a strong function of fast neutron fluence (proportional to fuel burnup) early in life and saturate at moderate fluence levels. Temperature-dependent data from irradiated and unirradiated coated tubes should be provided to justify the correlation used.

Chromium-Coated Zirconium

Recent data on unirradiated Cr-coated Zr indicate the yield stress of a coated part will be the same as that of an uncoated part (References 10, 11, 12). In tension, ceramic materials display a wide variation in strength. However, for thin coatings, the variable strength of the coating is not expected to strongly impact the overall strength of the substrate. Nano-indentation could be used to evaluate the yield stress of the coating. Although the yield stress of the tube may not change, if the thickness of the substrate tube is reduced to accommodate a coating that offers no strength, then the maximum load capability of that tube will be reduced.

B.7: Thermal and Irradiation Creep Rate

Zirconium-Alloy Cladding

The creep behavior of Zr alloy tubes has often been characterized by a thermal rate that can be developed based on ex-reactor creep tests, which are a function of stress and temperature, and an irradiation rate that can be developed based on the additional creep observed at the same stress and temperature during an in-reactor creep test. This creep rate can change significantly with small changes to alloy composition or microstructure. The increase or decrease in the thermal creep rate does not directly correlate to an increase or decrease in the irradiation creep rate. The creep rates for recrystallized cladding and stress-relief annealed cladding in FRAPCON are examples of this. Although both the thermal and irradiation creep rates are greater for the stress-relief annealed cladding than for the recrystallized cladding, the two increases are not the same fraction, so one increase could not be determined from the other (Geelhood, Luscher, & Raynaud, 2015 (Reference 13); Limback & Andersson, 1996 (Reference 14)). Both in-reactor and ex-reactor creep tests are recommended to justify the cladding creep correlation used, as these processes are potentially controlled by different mechanisms.

Chromium-Coated Zirconium

Recent data on unirradiated Cr-coated Zr indicate the thermal creep behavior of a coated part will be the same as that of an uncoated part (Reference 10). A thin metallic or ceramic coating on the cladding is unlikely to impact the thermal or irradiation creep behavior of the substrate. However, as mentioned above, small changes in composition and microstructure can have a significant impact on creep behavior, such that the application of the metallic or ceramic coating may impact the creep behavior. Additionally, one applicant has stated that the creep behavior of coated cladding differs from that of the uncoated substrate (Framatome, 2019, (Reference 15)). For this reason, both in-reactor and ex-reactor creep tests are recommended to justify the cladding creep correlation used for Cr-coated Zr cladding. The coating will put the substrate under compression (depending on methodology), which may improve the creep properties.

B.8: Axial Irradiation Growth

Zirconium-Alloy Cladding

Zr alloy tubes have been observed to grow axially with increased fast neutron fluence (Luscher, Geelhood, & Porter, 2015, (Reference 16)). This growth rate can change significantly with small changes to alloy composition, texture, or microstructure (e.g., Zircaloy-2, Zircaloy-4, M5®, ZIRLO). In-reactor data would be useful to justify the axial growth correlation used.

Chromium-Coated Zirconium

There is no current experience with the axial irradiation growth of coated parts relative to uncoated parts. Like thermal expansion mismatch strain, a difference in growth rates between the coating and substrate could lead to plastic deformation in the coating. This could be especially exacerbated for ceramic coatings, as ceramics typically have low plastic strain capability. Large differences in growth rate between the cladding and coating could lead to cracking or adhesion issues.

B.9: Oxidation Rate

Zirconium-Alloy Cladding

The oxidation rate is important to model in uncoated cladding tubes as the Zr oxide layer is less conductive than Zr metal. In the Zr alloy systems, ex-reactor autoclave corrosion data are significantly different from in-reactor corrosion data and should not be used to develop corrosion correlations for coated parts. Additionally, the corrosion behavior of nonfueled cladding segments may also not be representative of fueled cladding corrosion, as the surface heat flux in the fueled cladding seems to strongly impact oxidation rate (Cox, 2005, (Reference 17); Sabol, Comstock, Weiner, Larouere, & Stanutz, 1993, (Reference 18); Garde, Pati, Krammen, Smith, & Endter, 1993, (Reference 19)).

Chromium-Coated Zirconium

The Cr coatings under consideration will most likely result in very low oxidation rates under normal conditions and AOOs. Both the metallic and ceramic Cr coatings tend to produce a protective Cr oxide layer that exhibits excellent corrosion resistance, but this is a function of the coating application method. In-reactor data from fueled rods under prototypical coolant conditions could be used to demonstrate the oxidation rate or lack of one. Appropriate consideration should be given to unfueled corrosion data. It is also recommended that in-reactor data from rods with cracked coatings be evaluated to assess whether there is aggressive corrosion at cracks or interfaces.

B.10: Hydrogen Pickup

Zirconium-Alloy Cladding

It is important to quantify the hydrogen pickup in uncoated cladding tubes, as hydrides in Zr can lead to brittle behavior of the cladding (Zhao et al., 2017, (Reference 20)). Hydrogen from the outer surface is of primary concern, as hydrogen from the inner surface is controlled by the fuel fabricators by controls on pellet moisture.

Chromium-Coated Zirconium

In the case of Cr-coated Zr, if it is demonstrated that the metallic or ceramic Cr coating leads to negligible oxidation and is a barrier to hydrogen pickup, then hydrogen pickup might not be a concern for Cr-coated Zr cladding tubes during normal operation. Cracks and defects in the coating may also lead to higher localized hydrogen pickup and lead to cladding damage. Depending on the coating application method, there is potential for hydrogen pickup during coating fabrication. This is expected to be mitigated by process controls.

B.11: High-Temperature Ballooning Behavior

Zirconium Alloy Cladding

The burst stress as a function of temperature is important to know for LOCA analysis as this will determine when to start two-sided oxidation. The ballooning strain is important to determine flow blockage and establish whether a coolable geometry has been maintained. Ex-reactor burst tests at temperatures of interest for LOCA on representative cladding segments have been used in the past to establish the high-temperature ballooning behavior of Zr alloy tubes (Powers & Meyer, 1980, (Reference 21)). A significant difference in ballooning behavior between irradiated and unirradiated tubes has not been observed. This is likely due to annealing of radiation defects at burst temperatures.

Chromium-Coated Zirconium

Burst stress and ballooning strain are especially important for Cr-coated cladding, as the Cr coating is expected to provide a barrier to high-temperature oxidation, but it has not been proposed to coat the inner surface of the tube. Once ballooning and burst has occurred, there will be at least some bare Zr available for reaction with high-temperature steam. The existing data (see Section 6.2.2 of the PIRT report) on coated cladding indicate there may be smaller balloon sizes and rupture openings in coated cladding. This may limit high-temperature steam on the inner surface. Ex-reactor burst tests at temperatures of interest for LOCA on representative cladding segments would be useful on metallic or ceramic Cr-coated Zr alloy tubes to quantify the ballooning and burst behavior.

<u>B.12: High-Temperature Steam Oxidation Rate</u>

Zirconium Alloy Cladding

The steam oxidation rate is important for LOCA analysis because this determines whether the cladding has been overly thinned. This also determines the extra heat generation from the corrosion reaction and impacts the diffusion of oxygen into the beta-substrate, which leads to clad embrittlement.

Chromium-Coated Zirconium

Ex-reactor oxidation tests at temperatures of interest for LOCA on representative cladding segments have been used to establish the high-temperature steam oxidation rate of Zr alloy tubes. Such data would be useful on either metallic or ceramic Cr-coated Zr alloy tubes to quantify the oxidation rate.

APPENDIX C

Specified Acceptable Fuel Design Limits

C.1: Specified Acceptable Fuel Design Limits Related to Assembly Performance

Specified acceptable fuel design limits (SAFDLs) related to assembly performance are typically evaluated with simple hand calculations or by citing manufacturing controls or historic data. These limits may need revision relative to those typically used for zirconium (Zr) alloy tubes.

C.1.1: Rod Bow

Usually there is a penalty on departure from nucleate boiling (DNB) ratio or margin to critical power ratio to account for bowing. The limits of what degree of bowing is acceptable will not change with the introduction of chromium (Cr)-coated Zr, as this is controlled by the physical dimensions of the fuel assembly. However, bowing methods rely on correlations that are very empirical. Some testing or assessment would be useful to assess the applicability of the rod bow correlation used for Cr-coated cladding. The coating application should result in a uniform thickness as coating nonuniformities could lead to rod bow.

C.1.2: Irradiation Growth

The assembly design allows for a given amount of growth and will define the limit. The axial growth from Appendix B, Section B.8, of this guidance will be used to assess maximum growth. Change in the irradiation-induced growth for fuel rods may impact assembly growth through changes in slip loads through the spacer grids. This may affect some assembly designs differently, depending on the load chain.

C.1.3: Hydraulic Lift Loads

The limits for hydraulic lift loads are such that the upward hydraulic forces do not exceed the weight of the assembly and the downward force of the holddown springs. None of these parameters are expected to change with the introduction of Cr-coated Zr cladding. Existing limits and methods are expected to be adequate.

C.1.4: Fuel Assembly Lateral Deflections

The limits for fuel assembly lateral deflections are such that the control rod (pressurized-water reactor (PWR)) or control blades (boiling-water reactor (BWR)) can still be inserted as needed. Current assembly and channel bow methods are used to assess performance relative to these limits. Assembly and channel bow is not impacted by fuel rod performance but rather by channel design (BWR) and guide tube design (PWR). Therefore, these limits and methods are not expected to change with the introduction of Cr-coated Zr cladding tubes.

C.1.5: Fretting Wear

Current design limits state that fuel rod failures will not occur due to fretting. Fretting has historically been controlled though debris filters that reduce the possibility for debris fretting and through spacer design to reduce fretting between fuel rods and grid features. Ex-reactor fretting tests on unirradiated Cr-coated Zr cladding tubes would be useful to ensure that fretting behavior will not be an issue with the coating. A concern for Cr-coated Zr is that grid features are not damaged by the hard coating on the fuel rod. Ex-reactor fretting tests could be used to demonstrate that grids are not damaged by the hard coating on the fuel rod.

C.2: Specified Acceptable Fuel Design Limits Related to Rod Performance Assessed for Normal Operation and Anticipated Operating Occurrences

Current codes that are informed by the properties in Appendix B of this ISG can perform the following analyses. However, the limits may need revision relative to those typically used for Zr alloy tubes. Several of these SAFDLs also have application in accident analysis.

C.2.1: Cladding Stress

Cladding stress limits are typically set using a method described in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. Typically, these limits are based on unirradiated yield stress to represent the lowest yield stress. For Cr-coated Zr, the use of the unirradiated yield stress determined in Appendix A, Section A.6, of this guidance should be acceptable to determine a stress limit.

C.2.2: Cladding Strain

There are two cladding strain limits that are typically employed. The first steady-state limit is the maximum positive and negative deviation from the unirradiated conditions that the cladding may deform throughout life. The second transient strain limit is the maximum strain increment caused by a transient. This transient cladding strain may also be applicable to accident analysis. These cladding strain limits are typically justified based on mechanical tests (axial tension tests and tube burst tests) performed on irradiated cladding tubes. Ductility tends to decrease with irradiation (Reference 9) and saturates at some amount of radiation damage, so these tests are most relevant when performed until the effect saturates.

The uniform elongation or strain away from the rupture has been typically used as the strain capability for Zr-based alloys (Geelhood, Beyer, & Cunningham, 2004, (Reference 22)). This would be a good metric for Cr-coated Zr cladding to protect against cladding mechanical failure. For Cr-coated cladding, there is the additional concern that large strains in the cladding may lead to cracking of the coating (see Section 6.3.1 of the phenomena identification and ranking table (PIRT) report generated for the U.S. Nuclear Regulatory Commission (NRC) by Pacific Northwest National Laboratory, “Degradation and Failure Phenomena of Accident Tolerant Fuel Concepts: Chromium Coated Zirconium Alloy Cladding,” issued June 2019, (Reference 1)). Cracking of the coating can lead to a loss of corrosion protection for the substrate along with delamination.

C.2.3: Cladding Fatigue

The cladding fatigue limit is typically based on the sum of the damage fractions from all the expected strain events being less than 1.0. The damage fractions are typically found relative to the O’Donnell and Langer irradiated fatigue design curve (O’Donnell & Langer, 1964, (Reference 23)). It is currently unknown whether the O’Donnell and Langer irradiated fatigue design curve would be applicable to Cr-coated Zr. If unirradiated testing of coated cladding can demonstrate no change to the fatigue design curve, the use of O’Donnell and Langer may be appropriate.

It has been noted (Kvedaras, Vilys, Ciuplys, & Ciuplys, 2006, (Reference 24)) that in steels, Cr coating can improve or significantly worsen the fatigue lifetime due to different microstructures produced in the coating. This was also observed in the case of (cold-spray) Cr-coated Zr, where the fatigue life went down with the application of a coating (Sevecek et al., 2018, (Reference 25)). Because of this, fatigue data from irradiated cladding that was produced using a representative process for the applicant in question are recommended to either confirm the O'Donnell and Langer irradiated fatigue design curve or to develop a new fatigue design curve. 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 as mentioned in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 4.2, "Fuel System Design," (Reference 2).

C.2.4: Cladding Oxidation, Hydriding, and Crud[1]

For Zr-alloy cladding, the cladding oxidation limit is designed to preclude oxide spallation that has typically been observed above 100 micrometers (μm). Oxide spallation can lead to a local cool spot that acts as a sink for hydrides, creating a local, extremely brittle hydride lens. The hydrogen limit is designed to ensure that the strain limit previously identified will be applicable since high levels of hydrogen (greater than 600 parts per million) can cause embrittlement of the cladding. Hydrogen is not the only embrittlement mechanism, and there may be other embrittlement mechanisms that are discussed elsewhere. There is no explicit limit on crud, other than that it be explicitly considered if it is present, and it is typically modeled as an insulating layer around the fuel rod in plants that have crud issues.

None of these limits are particularly relevant to Cr-coated cladding since the outer oxide will be chromium (III) oxide (Cr_2O_3) rather than zirconium dioxide (ZrO_2) and the Cr or Cr_2O_3, or both, are expected to be a barrier against hydrogen uptake, and thus limits are proposed and justified for the coatings to ensure cladding integrity.

If intermetallics form on the surface of the cladding, the oxide could be a mixture of ZrO_2 and Cr_2O_3.

As with Zr-alloy cladding, the crud should be monitored in plants, and the NRC staff should ensure that it is explicitly considered if it is present and modeled as an insulating layer around the fuel rod.

C.2.5: Fuel Rod Internal Pressure

There are several possible limits for rod internal pressure that are discussed in SRP Section 4.2. The first and most straightforward is that the rod internal pressure shall not exceed the coolant system pressure. No outward deformation or hydride reorientation is possible if the stress in the cladding is in the compressive directions. This situation does not change with the application of a Cr coating. Therefore, this limit would still be applicable to Cr-coated Zr cladding.

Greater rod internal pressures may be justified based on the following criteria:

- no cladding liftoff during normal operation
- no reorientation of the hydrides in the radial direction in the cladding

[1] Crud is defined as a colloquial term for corrosion and wear products (rust particles, etc.) that become radioactive (i.e., activated) when exposed to radiation.

- a description of any additional failures resulting from DNB caused by fuel rod overpressure during transients and postulated accidents

It has typically been determined by applicants with Zr-alloy cladding that the first of these criteria, no cladding liftoff during normal operation, is the most limiting. This should be confirmed by the applicant using a Cr-coated Zr cladding to still be the case. If this is found to be the case, the pressure limit where cladding liftoff could occur is typically set as the pressure where the upper bound cladding creep rate will exceed the lower bound fuel pellet swelling rate. For Cr-coated Zr cladding, the fuel pellet swelling rate will not be changed and the cladding creep rate will be determined as discussed in Appendix B, Section B.7, of this guidance, provided that the coating does not significantly change the cladding thermal conductivity.

C.2.6: Internal Hydriding

Internal hydriding is typically addressed through manufacturing controls on the pellet moisture limit. The inner surface for the Cr-coated Zr cladding will be the same, and therefore the typical approach would also apply for Cr-coated Zr cladding. It is not expected that the application of a coating will impact this conclusion.

C.2.7: Cladding Collapse

Cladding collapse in modern nuclear fuel rods has been mitigated by pellet design features such as dishes and chamfers on the ends of the pellet that effectively eliminate axial gaps in the fuel pellet column. Nevertheless, cladding collapse analyses are performed for potential small axial gaps between pellets and in the upper plenum region. The key input into this analysis is the cladding creep rate. For Cr-coated Zr, the cladding creep rate will be determined as discussed in Appendix B, Section B.7, of this guidance.

C.2.8: Overheating of Fuel Pellets

For this analysis, the limit is the melting temperature of the fuel pellets. This will not be impacted by the introduction of Cr-coated Zr cladding, and therefore the limit for this SAFDL may stay the same.

C.2.9: Pellet-to-Cladding Interaction

Typically, there is no explicit limit set on pellet-to-cladding interaction. Various manufacturing designs and inspections and the transient cladding strain limit are expected to cover this SAFDL. The inner surface for the Cr-coated Zr cladding will be the same, and therefore the typical approach would also apply for Cr-coated Zr cladding.

C.2.10: Boiling Crisis

“Boiling crisis” refers to the point at which the boiling regime changes to one that is no longer capable of supporting the heat transfer from the rod surface necessary for adequate cooling, resulting in a cladding temperature excursion. For PWRs, the boiling regime change of concern is usually DNB, while in BWRs, the boiling regime change of concern is typically the onset of transition boiling. Current fuel designs have SAFDLs related to the prevention of boiling crisis for steady-state and anticipated operational occurrence (AOO) conditions, specified by the critical heat flux (CHF) or critical power (CP) for PWRs and BWRs, respectively. The effects of Cr coating on boiling crisis behavior are discussed in Section C.3.1 of this appendix.

C.3: Specified Acceptable Fuel Design Limits Related to Fuel Rod Performance Assessed for Accident Conditions

Current codes that are informed by the properties in Appendix B to this guidance can perform the analyses described below. However, the limits may need revision relative to those typically used for Zr alloy tubes. Several of these SAFDLs also have application in AOO analysis.

C.3.1: Overheating of the Cladding

Overheating of the cladding results when the boiling regime changes to one that is no longer capable of supporting the heat transfer from the fuel rod surface necessary for adequate cooling. For PWRs, the boiling regime change of concern is usually DNB, while in BWRs, the boiling regime change of concern is typically the onset of transition boiling (also known as dryout). Current fuel designs have SAFDLs related to the prevention of boiling crisis for steady-state, AOO, and some accident conditions. In PWRs, the fuel rod heat flux must be kept below the CHF, and in BWRs the assembly power must be kept below the CP. For AOOs, the SAFDLs must be met for the design basis to be satisfied. For design-basis accidents, any fuel rods exceeding the thermal margin criteria are assumed to have failed and are included in fission product release dose calculations.

The boiling transitions are shown graphically in Figure 5.1 of the PIRT report. Typical limits are based on ex-reactor flow tests on electrically heated fuel assembly mockups to determine where DNB or boiling transition occurs. The CHF or CP is primarily influenced on the geometry of the assembly, although surface conditions of the fuel rods may also impact the CHF or CP. Surface conditions include surface roughness, wettability, and porosity (e.g., of a crud layer). Most studies have concluded that roughness has little or no impact on CHF (Collier & Thome, 1994 (Reference 26); Kandlikar, 2001, (Reference 27); O'Hanley et al., 2013, (Reference 28)), though some studies have shown a noticeable difference between rough and very smooth surfaces (Weatherford, 1963, (Reference 29)). Surface porosity and wettability are thought to have a much more significant impact, as demonstrated by several experimental studies (Reference 27; Takata, Hidaka, Masuda, & Ito, 2003, (Reference 30); Reference 28). Boiling heat transfer experimental results indicate similar CHF for coated and uncoated cladding (Jo, Yeom, Gutierrez, Sridharam, & Corradini, 2018, (Reference 31); Jo, Gutierrez, Yeom, Sridharan, & Corradini, 2019, (Reference 32)).

The application of a coating to fuel rods, while keeping the rest of the assembly the same, is not expected to impact CHF or CP correlations if the surface conditions of the coating are similar to that of the reference Zr alloy tubes. It is currently not known what the surface roughness, contact angle, or crud deposition rate for a Cr-coated tube will be relative to an uncoated tube. If the coating results in a significantly different surface condition or cladding outer diameter than the reference Zr alloy tube, then ex-reactor flow tests on electrically heated fuel assembly mockups with a prototypical coated cladding tube could be performed to determine the effect on DNB or boiling transition behavior.

Currently, the majority of CHF/CP tests are performed on electrically heated prototypical fuel assemblies constructed of Inconel instead of Zr alloy. If the coating affects the cladding surface conditions in a manner that influences DNB or boiling transition behavior, the use of plain Inconel tubes may not be appropriate for determining CHF or CP for Cr-coated Zr alloy cladding.

As mentioned in Section 4.1 of the PIRT report, the possibility of formation of a low-temperature eutectic between Cr and Zr exists if temperature exceeds 2,430 degrees Fahrenheit (F) (1,332 degrees Celsius (C)). This formation should either be considered under this damage mechanism or under generalized cladding melting (see Appendix B, Section B.3.7, of this guidance).

C.3.2: Excessive Fuel Enthalpy

Excessive fuel enthalpy relates to the sudden increase in fuel enthalpy from an RIA below the fuel melting limit that can result in cladding failure due to pellet-cladding mechanical interaction. Current fuel enthalpy limits are based on RIA tests that have been performed on irradiated and unirradiated fuel rodlets in various test reactors, and a limit has been determined for what level of fuel enthalpy increase will cause cladding failure.

For Zr alloy cladding, these data have been collected over a very long period, and it may not be practical to collect this amount of data for Cr-coated Zr cladding.

An alternate approach comes from the fact that cladding failure due to excessive fuel enthalpy is driven by pellet-cladding mechanical interaction, which causes the cladding to exceed its ductility limit. Therefore, it is possible to collect uniform elongation (strain at maximum load) data from the irradiated cladding mechanical tests that need to be performed to determine postirradiation strength and ductility. If it can be shown that the Cr coating has a beneficial or negligible impact on the uniform elongation relative to the reference Zr alloy cladding, then it could be reasonably argued that the current RIA failure limits are applicable to Cr-coated Zr cladding.

C.3.3: Bursting

Bursting of the fuel rod relates to the failure of fuel rods due to high temperature and high gas pressures during a LOCA. This can also be a consideration during an RIA. It is important to know the rupture stress as a function of temperature and the amount of ballooning that would occur. There are no specific design limits associated with cladding rupture other than that the degree of swelling not be underestimated and the balloon not block the coolant channel. Additionally, the time of rupture needs to be known so that oxidation on the cladding inner surface and its associated heat is correctly modeled.

An applicant will typically use an empirical correlation for burst stress and ballooning strain such as the one given in NUREG-0630, “Cladding Swelling and Rupture Models for LOCA Analysis,” issued April 1980, (Reference 21). If an applicant uses NUREG-0630 for Cr-coated Zr cladding, it would be useful to collect some data to show that the performance of Cr-coated Zr is bounded by these limits. Alternatively, if the applicant wants to propose new burst stress and ballooning strain limits, a significant body of burst data would be useful to demonstrate that the degree of swelling will not be underestimated. Currently available data suggest that for Cr-coated cladding, the balloon region is smaller and burst temperature increases (see Section 6.2.2 of the PIRT report); however, this should be confirmed for the specific coating in question.

C.3.4: Mechanical Fracturing

Mechanical fracturing refers to a defect in the cladding caused by an externally applied force. Typically, this limit has conservatively been set as applied stresses above 90 percent of the irradiated yield stress. This limit should not be exceeded for normal operation and AOOs. For design-basis accidents, the fuel rods exceeding this limit are assumed to have failed and are included in fission product release dose calculations.

This limit is acceptable for Cr-coated Zr cladding given that the irradiated yield stress obtained as described in Appendix B, Section B.6, of this guidance is used.

C.3.5: Cladding Embrittlement

Cladding embrittlement relates to embrittlement of the fuel cladding, particularly in the ballooned region of the cladding during LOCA. Cladding embrittlement during LOCA should be precluded so the fuel assemblies with ballooned rods are not severely damaged by post-LOCA loads such as reflood and quenching, including blowdown loads. In 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," the NRC specifies a cladding temperature limit of 2,200 degrees F (1,204 degrees C) and a peak oxidation of 17-percent equivalent cladding reacted for Zr alloy cladding.

The PIRT ranked this damage mechanism as high (see Appendix A to the PIRT report). It is not known if these limits will be acceptable for Cr-coated Zr cladding. It appears as if the outer surface will reduce the high-temperature metal-water reactor from that of bare Zr, but it is unknown whether some other mechanism could cause embrittlement of the cladding. One possible mechanism could be Zr-Cr interdiffusion as discussed in Section 4.2 of the PIRT report. The formation of a brittle rim of $ZrCr_2$ could lead to brittle cladding failure similar to how the formation of a dense hydride rim can lead to brittle cladding failure.

Tests showing ductility (see Section 6.2.6 of the PIRT report) at either these existing limits or establishing new limits would be useful to demonstrate that embrittlement will not occur. In addition to the tests performed to establish the ballooning (see Appendix B, Section B.11, of this guidance) and high-temperature oxidation behavior (see Appendix B, Section B.12, of this guidance), some prototypic integral LOCA tests (e.g., Flanagan, Askeljung, & Puranen, 2013, (Reference 33)), where cladding tubes are subject to ballooning and burst in steam under expected time frames and samples are then subjected to mechanical loading such as bend tests after ballooning, burst, and high-temperature oxidation, are very useful to establish cladding embrittlement limits. For these tests, irradiated cladding tubes are preferable.

C.3.6: Violent Expulsion of Fuel

The violent expulsion of fuel relates to the sudden increase in fuel enthalpy from an RIA that can result in the melting, fragmentation, and dispersal of fuel. This could result in a loss of coolable geometry and produce a pressure pulse that could damage the reactor vessel. The following are typical limits for violent expulsion of fuel:

- peak radial average fuel enthalpy below 230 calories per gram (cal/g)
- peak fuel temperature below melting temperature

It is expected that cladding failure will occur well before 230 cal/g for both Zr alloy and Cr-coated Zr cladding. These limits are derived to prevent the violent ejection of fuel from failed cladding. As such, these limits relate more to the fuel than to the cladding and are expected to be appropriate for Cr-coated Zr cladding.

C.3.7: Generalized Cladding Melting

Generalized cladding melting is applicable to design-basis accidents and is set to preclude the loss of coolable geometry. The limit is set as the cladding melting temperature, which for Zr is 3,366 degrees F (1,852 degrees C). For Zr alloy tubes, the embrittlement limit of 2,200 degrees F (1,204 degrees C), (see Section C.3.5 of this appendix) is more limiting. However, as discussed in Appendix B, Section B.3.5, of this guidance, it is unknown what the limit for Cr-coated Zr embrittlement will be, so cladding melting should still be considered for Cr-coated Zr.

The melting temperature of Cr at 3,375 degrees F (1,857 degrees C), is virtually identical to that of Zr at 3,366 degrees F (1,852 degrees C). However, the formation of a low-temperature eutectic between Cr and Zr at 2,430 degrees F (1,332 degrees C), occurs significantly lower than either of the individual melting temperatures. The formation of a low-temperature eutectic with a thin coating may not represent a loss of geometry such as generalized cladding melting, but the formation of the eutectic should either be considered under this damage mechanism or under overheating of the cladding (see Section C.3.1 of this appendix).

C.3.8: Fuel Rod Ballooning

Ballooning of the fuel rod relates to failure of fuel rods due to high temperature and high gas pressures during a LOCA. It is important to know the rupture stress as a function of temperature and the amount of ballooning that would occur. There are no specific design limits associated with cladding rupture, other than that the degree of swelling should not be underestimated and the balloon should not block the coolant channel.

An applicant will typically use an empirical correlation for burst stress and ballooning strain such as the one given in NUREG-0630 (Reference 21). If an applicant uses NUREG-0630 for Cr-coated Zr cladding, it would be useful to collect some data to show that the performance of Cr-coated Zr is bounded by these limits. Alternatively, if the applicant wants to propose new burst stress and ballooning strain limits, a significant body of burst data from either unirradiated or irradiated cladding tubes would be useful to demonstrate that the degree of swelling will not be underestimated.

C.3.9: Structural Deformation

Structural deformation refers to externally applied loads during a LOCA or safe-shutdown earthquake that could deform the fuel assemblies or cause fuel fragmentation such that coolable geometry would be lost. This limit has conservatively been set as applied stresses above 90 percent of the irradiated yield stress. For design-basis accidents, the number of fuel rods exceeding this limit are assumed to have failed and are included in fission product release dose calculations.

This limit is acceptable for Cr-coated Zr cladding given that the irradiated yield stress obtained as described in Appendix A, Section A.6, of this guidance is used.

C.4: New Degradation Mechanisms/Other Considerations

There have been several new damage mechanisms identified for Cr-coated Zr cladding. These may either be addressed by applicants through existing limits or as separate limits. The following sections identify those new damage mechanisms that have been identified for

Cr-coated Zr through a technical review of the recent data and a general understanding of coating behavior. Each section will identify the potential for fuel system damage, fuel rod failure, or impact on fuel coolability. These sections will also identify existing SAFDLs that could be used to account for these damage mechanisms. These damage mechanisms are physical mechanisms and should be addressed even if no credit for coating performance is credited in the fuel system safety review.

C.4.1: Coating Cracking

Cracking of the coating could occur during the relatively large (0.5-percent to 1-percent strain) deformations that are observed occur in the cladding due to cladding thermal expansion, cladding creepdown, deformation of the cladding due to pellet swelling, and axial irradiation growth. Cracking could also occur in the cladding due to repeated small strain (0.01-percent to 0.1-percent strain) cyclic operation. Finally, cracking could occur during a design-basis scenario that causes large strain from pellet expansion, e.g., from an RIA, or gas overpressure and ballooning, e.g., from a LOCA.

The PIRT ranked this damage mechanism as high during accident conditions (see Appendix A to the PIRT report). Excessive cracking of the coating could reduce or eliminate the benefit that the coating provides for normal operation (reduced in-reactor corrosion and hydrogen pickup) as well as during accident conditions (may expose significant amount of Zr to high temperature steam). Cracking of the coating could also create crack tips that extend into the Zr cladding that could provide stress concentrations for further environmentally assisted crack mechanisms and could ultimately lead to cladding failure.

Cracking of the coating should be considered in the development of the cladding strain limit (see Appendix C, Section C.2.2, of this guidance) and the cladding fatigue limit (Appendix C, Section C.2.3, of this guidance). In these cases, it should be considered if failure is defined when cracking of the coating is observed. Cracking of the coating should also be considered in the development of high-temperature ballooning (Appendix B, Section B.11, of this guidance) and high-temperature oxidation (Appendix B, Section B.12, of this guidance) correlations. If cracking is observed following ballooning, then high-temperature oxidation correlations should be developed with consideration of cracked coating. Additionally, cladding embrittlement limits (Appendix C, Section C.3.5, of this guidance) should be developed with consideration of cracked coating.

C.4.2: Coating Delamination

Delamination of the coating could occur due to a variety of reasons, including poor adherence to the substrate and differential thermal expansion between the coating and the substrate.

The PIRT ranked this damage mechanism as high during accident conditions (see Appendix A to the PIRT report). Delamination of the coating could eliminate the benefit that the coating provides for normal operation (reduced in-reactor corrosion and hydrogen pickup) as well as during accident conditions (may expose significant amount of Zr to high-temperature steam), depending on the amount of delamination. Local coating delamination could create a local cool spot on the cladding, which is a sink for hydrogen diffusion. This local cool spot could develop a hydride blister that results in local brittle cladding behavior. Finally, coating delamination can increase the quantity of debris in the reactor coolant system, which could lead to enhanced debris fretting and could impact the performance of the emergency core coolant system pump in the event of an accident if the debris filters become clogged with debris from delaminated coating.

Delamination of the coating should be considered in the development of the cladding strain limit (see Section C.2.2 of this appendix) and the cladding fatigue limit (Section C.2.3 of this appendix). In these cases, it should be considered if failure is defined to be observed delamination of the coating. Delamination of the coating should also be considered in the development of high-temperature ballooning (Appendix B, Section B.11) and high-temperature oxidation (Section B.12) correlations. If delamination is observed following ballooning, then high-temperature oxidation correlations should be developed with consideration of cladding with a delaminated coating. As discussed in Section 4.2 of the PIRT report, the $ZrCr_2$ phase that could form due to interdiffusion could exhibit a greater corrosion rate than bare Zr. Additionally, if this is the case, cladding embrittlement limits (Section C.3.5 of this appendix) should be developed with consideration of delaminated cladding. LOCA blowdown loads could also lead to delamination of the coating. The NRC staff should consider debris effects on long-term core cooling. Specifically, the potential for delamination should be evaluated and accounted for following burst (Section C.3.3 of this appendix), mechanical fracture (Section C.3.4), ballooning (Section C.3.8), and structural deformation (Section C.3.9).

C.4.3: Chromium-Zirconium Interdiffusion

As discussed in Section 4.2 of the PIRT report, if temperatures at the Cr-Zr interface and the time at temperature are great enough, there will be the formation of a CrZr intermetallic that is more brittle than either Cr or Zr separately. If this intermetallic layer is thick enough, it could lead to brittle cladding failure. Thin layers of this intermetallic would likely not reduce the overall cladding ductility. However, the critical thickness for overall brittle behavior is not known. The calculations from Section 4.2 are shown below:

- normal conditions (572 degrees F (300 degrees C) – 662 degrees F (350 degrees C) for 2,000 days): 0.1 to 0.3 μm-thick intermetallic layer
- loss-of-coolant conditions (1,652 degrees F (800 degrees C) – 2,192 degrees F (1200 degrees C) for 1 hour): 0.2 to 1.4 μm-thick intermetallic layer
- long-term loss-of-coolant (1,472 degrees F (800 degrees C) – 2,192 degrees F (1,200 degrees C) for 1 day): 1 to 7 μm-thick intermetallic layer

Initial data from several programs have not shown significant interdiffusion in various coating concepts. It is noted that the numbers above are predictions based on limited data and are provided for context. The NRC staff should evaluate data specific to the technology under review.

Unless otherwise accounted for in specific strain or ballooning limits, the formation of this CrZr intermetallic should be avoided. During normal operations and AOOs, the temperature at the Cr-Zr interface is only expected to allow for the formation of a very thin CrZr intermetallic layer, but during design-basis accidents, the cladding temperature may be large enough to form a significant thickness of this layer (see Section 4.2 of the PIRT report). Other possibilities for the formation of the CrZr intermetallic phase include during application of the coating if the substrate temperature is too great, and during the welding of end caps in the heat-affected zone of the weld.

The CrZr intermetallic is both brittle and exhibits extremely poor high-temperature corrosion behavior (see Section 4.2 of the PIRT report). If a significant thickness of CrZr intermetallic were to form during high-temperature conditions during a design-basis accident or some manufacturing process, the cladding could behave in a brittle manner, the corrosion reaction may worsen, and various design limits on strain and cladding embrittlement may no longer be applicable.

Cr-Zr interdiffusion should be considered either implicitly or explicitly in the development of limits on overheating of the cladding (see Section C.3.1 of this appendix), clad embrittlement (Section C.3.5), and eutectic formation related to generalized clad melting (Section C.3.7). If some Cr-Zr interdiffusion is caused during the manufacturing process, then it should be ensured that limits are developed on prototypic parts from this process and that tests are performed in localized areas known to have the possibility for interdiffusion.

C.4.4: Radiation Effects on Chromium

It has been noted that the irradiation of Cr will result in the formation of the radioisotope Cr-51 with a half-life of 28 days. It is known that this isotope will be formed, but it is not known whether this isotope will be released to the coolant in significant quantities. For a chromium nitride (CrN) coating, the nitrogen will lead to the production of some carbon-14.

A second concern is what the impact of fast neutron irradiation on Cr metal and other Cr-containing compounds will be. In Zr, fast neutron irradiation leads to a dramatic increase in strength and a reduction in ductility (Reference 9). Recent ion beam irradiation data indicate that cold-spray Cr coatings are more resistant to radiation defects than bulk Cr (Maier et al., 2018, (Reference 34).

The release of Cr-51 from the cladding into the coolant could challenge the plant dose release limit or the ability of the chemical and volume control system to eliminate Cr ions before they plate out on the fuel and the other reactor components. The impact of fast neutron irradiation on the strength and ductility of the Cr metal or other Cr-containing compounds could lead to a degradation in coating performance beyond what was expected based on tests on unirradiated material.

The formation and possible release of Cr-51 is an issue that may be monitored through ongoing surveillance at the plant. Plants already have a process in place to evaluate the radioisotopes and the gaseous and liquid effluents and to report this information to the NRC on an annual basis. If Cr-51 in the coolant begins to challenge plant dose release limits, it will be observed to increase as more of the fuel in the core is transitioned to Cr-coated Zr cladding. In this case, systems can be implemented to effectively remove this radioisotope before it becomes a safety problem. Similarly, with the impact of Cr ions on the coolant chemistry, a surveillance plan put in place alongside the implementation of Cr-coated Zr cladding to monitor the coolant chemistry will mitigate any impact of Cr ions. The impact of fast neutron irradiation on Cr mechanical properties will be inherently included in material property correlations and limits that are developed based on irradiated material as described in previous sections of this guidance.

Data may already be available for radiation damage, as Cr-containing alloys and Cr coatings are already present in core components.

C.4.5: Subsurface Damage

As mentioned in Section 3.0 of the PIRT report, many physically bonded coating systems may require mechanical preparation such as grit blasting to obtain a suitable surface for coating bonding. It is currently unknown what the impact of this surface preparation will be on the performance of the coated cladding. The impact will undoubtedly be highly process dependent and should be evaluated for each qualified coating in question.

C.4.6: Residual Stress

When coatings are applied at a different temperature than their operation temperature, it is possible to develop residual stress in the cladding and the coating. This stress could lead to unexpected cladding or coating failure. It is currently unknown what the impact of this residual stress will be on the performance of the coated cladding. The impact will undoubtedly be highly process dependent and should be evaluated for each qualified coating in question.

C.4.7: Galvanic Corrosion

Galvanic corrosion refers to corrosion damage induced when two dissimilar materials are coupled in a corrosive electrolyte. It occurs when two (or more) dissimilar metals are brought into electrical contact under water. Galvanic corrosion can be accelerated under the effects of radiation, as has been observed with the so-called “shadow corrosion” observed between BWR channel boxes and control blades. When a galvanic couple forms, one of the metals in the couple becomes the anode and corrodes faster than it would all by itself, while the other becomes the cathode and corrodes slower than it would alone.

Dissimilar metals in this case include Cr+Zr, Inconel+Cr, and CrN+Zr. No indication of galvanic corrosion, irradiation assisted or otherwise, between these systems has been found in this effort.

C.4.8: Defects

Any coating process will result in some population of defects. Depending on the size and concentration of these defects, they could lead to oxidation under the coating either in normal operating conditions or accident conditions. This could lead to cracking or delamination of the coating, which could eliminate the benefits of the coating and have other safety consequences (see Sections C.4.1 and C.4.2 of this appendix). The PIRT ranked this damage mechanism as high during accident conditions (see Appendix A to the PIRT report). Each process in question should define the allowable defects and justify the presence of these defects based on the testing of cladding with similar defect concentrations.

C.4.9: Eutectic Formation

The formation of eutectics seems to be a concern primarily for beyond-design-basis accident conditions. The lowest temperature eutectic for the Cr-Zr system occurs at 2,430 degrees F (1,332 degrees C). If operation beyond the current design-basis temperature limit of 2,192 degrees F (1,200 degrees C), is requested, then the formation of eutectics and their impact on the coating should be considered. Additionally, in systems other than the Cr-Zr system, such as Cr-Zr-N, the formation of lower temperature eutectics should be considered for both design-basis and beyond-design-basis accident conditions.

APPENDIX D

References

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APPENDIX E

Analysis of Public Comments on
Draft Interim Staff Guidance, "Supplemental Guidance Regarding the Chromium-Coated Zirconium Alloy Fuel Cladding Accident Tolerant Fuel Concept

Comments on the subject draft interim staff guidance are available electronically at the U.S. Nuclear Regulatory Commission's (NRC's) electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can access the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. The following table lists the comments the NRC received on the draft ISG.

Letter Number	ADAMS Accession No.	Commenter Affiliation	Commenter Name
1	ML19344C125	Nuclear Energy Institute	Benjamin Holtzman

This document lists each public comment by letter number, as given in the table above. The original comment as written by the commenter is listed first, followed by the NRC's response.

Letter 1—Comments from the Nuclear Energy Institute

Comment No. 1-1

Industry recognizes the importance of process controls in manufacturing but believes that the NRC should not license specific manufacturing processes. Industry has brought up this concern at the NRC Public Meeting on August 6th and NRC management agreed that the vendor suppliers produce high-quality, defect free fuel rods through the current supply chain qualification process, quality control of suppliers and manufacturing, product inspections and certifications under the provisions of 10CFR50 App B. NUREG-0800 should remain a performance-based standard. Furthermore, there was agreement that the coated cladding concepts addressed in the ISG are not conceptually different from the current fuel products and that the existing manufacturing oversight framework is adequate.

However, the ISG text still references PIRT Section 6.4.2 and states that the applicant should ensure the performance concerns referenced in the PIRT Section are addressed. This creates the potential for a reviewer to regulate on process rather than follow the current performance-based regulatory process as noted as the intent by NRC management in the August 6th Public Meeting. Therefore, the ISG text should be clarified that the specifics of the manufacturing process should not be included in the licensing criteria.

Proposed Change

Please revise the text as noted:
Currently it is not possible to definitively state what data are available to justify these properties, ~~because small d.~~ Differences in applicant-specific processes can ~~have a significant~~ impact on the final mechanical and material properties. Therefore, the applicant should provide data or

other justification from its specific cladding product to justify material property models. (Appendix B page 1 of 7)

...microstructures, and higher ductility-compliant layers can be utilized to reduce interface strains. The important mechanical and material properties to be reviewed are those of the finished applicant-specific fuel product, not those during an interim manufacturing process. (Appendix B.2, page 3 of 7)

NRC Response

The NRC staff agrees in part with this comment. It is the responsibility of the applicant to provide a definition of the product under review. In the case of legacy fuel rod cladding alloys, other industry and ASTM specifications may sufficiently describe the manufacturing and testing of the cladding. In the case of coatings, no such standards exist. As such, each applicant needs to describe the product for which NRC approval is requested, which may or may not include manufacturing process parameters. This description of the product could include material and dimensional specifications, as well as quality control and testing measures to ensure that the coating will perform equivalent to the coating used during testing and data collection. While this process is greatly aided by the use of performance-based metrics (i.e. non-destructive testing of cladding after coating, or destructive testing of a cladding batch sample), there is not yet a standard set of tests due to the novelty of these coatings. Therefore, the NRC cannot categorically reject the possibility of including a manufacturing process parameter within the scope of the topical report and safety evaluation because the content of the topical report submittals, and specifically the product descriptions, are unknown. The NRC staff agrees that existing regulatory structure for oversight of quality control of suppliers and manufacturing is effective and should accommodate these coated claddings.

The NRC staff agrees with the proposed changes to Appendix B, page 1. The final ISG was changed by incorporating the proposed language.

The NRC staff disagrees with the proposed changes to Appendix B.2, page 3. The paragraph for which the changes are proposed is included in the ISG to provide background information to the reviewer on different coating techniques and contains no requirements or recommendations for review. The staff believes that adding a disclaimer that it does not apply to intermediate manufacturing products may cause unnecessary confusion. The final ISG was not changed as a result of this comment.

Comment No. 1-2

The NRC is transforming to become a modern, risk-informed regulator but Fuels continue to lag behind the rest of the NRC in becoming risk-informed. The mission of the NRC is reasonable assurance of adequate protection of public health and safety as noted in the NRC letter ‘Applying the Principles of Good Regulation as a Risk-Informed Regulator.’ Adequate protection (no undue risk) does not mean zero risk. The area of fuels licensing continues to be regulated deterministically on design-basis accident considerations only. Any risk-informed performance-based idea, while accepted in principle, are not implemented in regulations or guidance as there is a fear a failure for any aspect of the fuel product at any time.

This approach is unnecessarily conservative and demonstrates a lack of risk-informed perspective. The fuel pellet matrix and fuel cladding are the first two boundaries protecting the public, but they are not the only boundaries. A failure of the fuel matrix and fuel cladding does not create a situation where the reasonable assurance of adequate protection of public health and safety is no longer maintained – though it is a bad and costly day for both the fuel supplier and utility.

Proposed Change

Please revise the text as noted:

Potential new damage mechanisms have been identified in Appendix C, Section C.4, of this ISG. The reviewer should ensure that these mechanisms have been ruled out sufficiently by the applicant for the domain approved by the NRC such that the entirety of the fuel in concert with the supporting systems maintains the reasonable assurance of adequate public health and safety, that existing SAFDLs already protect against the mechanisms, or that new SAFDLs have been developed to protect against them.

Based upon an investigation of available performance testing and known data gaps, Section 6.4.2 of the PIRT report identified several performance concerns for chromium-coated zirconium alloys. The reviewer should ensure that these performance concerns have been ruled out sufficiently by the applicant for the domain approved by the NRC such that the entirety of the fuel in concert with the supporting systems maintains the reasonable assurance of adequate public health and safety, that existing SAFDLs already protect against the damage mechanisms, or that new SAFDLs have been developed to protect against them. (Appendix A, pages 2 and 3 of 9)

The PIRT report also suggests two paths that an applicant may take to analyze each property: treating the cladding and coating as separate layers and treating the cladding and coating together as a composite material. A subset of the composite material strategy may be to demonstrate that the coating will have a negligible impact on a property and to use the property of the underlying substrate. Any of these paths may be appropriate provided sufficient justification from the applicant such that the entirety of the fuel in concert with the supporting systems maintains the reasonable assurance of adequate public health and safety, and a variety of these strategies may be used to disposition the various properties. (Appendix A, page 4 of 9)

NRC Response

The NRC staff disagrees with this comment. The standard for NRC staff review is reasonable assurance of adequate protection of public health and safety. The ISG is an extension of NUREG-0800, "Standard Review Plan," which provides guidance to the staff on this review standard. The proposed edits are not necessary to reinforce the reasonable assurance standard and are therefore not incorporated.

With regard to the rest of the comment, the NRC considers the safety of the fuel to be of paramount importance because it is the driver of the accident source term and contains two primary fission product barriers. Therefore, General Design Criterion 10, "Reactor Design," requires the fuel to be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation and anticipated operational occurrences (AOOs).

Within these constraints, however, the NRC has continued to risk-inform fuel licensing where appropriate. The design basis acceptance criteria for fuels (the SAFDLs) are performance-based, with few exceptions, and are proposed by the applicant when seeking to license new fuels. Some of the generally-accepted SAFDLs explicitly incorporate uncertainty and risk in their development and evaluation (e.g., boiling transition limits).

For ATF, the overall licensing approach is further evidence of the NRC's commitment to better risk-informing fuel licensing. The development of a phenomena identification and ranking table (PIRT), for example, provides reviewers with information on which phenomena and SAFDLs are most safety-significant. Collection of in-reactor data using LTAs, which is an integral part of the ATF licensing process, is inherently risk-informed in that the size and scope of LTA campaigns is commensurate with the probability and consequences of their failure. The NRC recently issued a guidance letter to NEI on LTAs (ADAMS Accession No. ML18323A169) that clarifies the regulatory requirements and improves regulatory reliability.

As discussed in the PRA Policy Statement promulgated by the Commission on August 10, 1995 (62 FR 42622), the continued use of defense-in-depth is an integral part of the NRC's approach to considering risk in decisionmaking, as are safety margins, performance monitoring, continued compliance with the regulations, and change in risk. The staff is committed to appropriately weighing all elements of risk information when making risk-informed decisions.
No change was made to the final ISG as a result of this comment.