DRAFT INTERIM STAFF GUIDANCE

SUPPLEMENTAL GUIDANCE REGARDING THE

CHROMIUM-COATED ZIRCONIUM ALLOY FUEL CLADDING

ACCIDENT TOLERANT FUEL CONCEPT

ATF-ISG-01

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC, or Commission) 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 interim staff guidance (ISG) is intended to provide guidance for 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 PIRT report, "Degradation and Failure Phenomena of Accident Tolerant Fuel Concepts: Chromium Coated Zirconium Alloy Cladding" (Reference 1). The suggested cladding properties, specified acceptable fuel design limits (SAFDLs), and new failure mechanisms sections from the PIRT are replicated in Appendix B and C. These appendices supersede sections 5.1 and 5.2 of the PIRT report.

This ISG is not intended as stand-alone review guidance, but instead supplements NUREG-0800, "Standard Review Plan," (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," Section 4.4, "Thermal and Hydraulic Design," and 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 to determine the 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.

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 DOE and industry ATF programs, the staff proactively developed a plan, “Project Plan to Prepare the U.S. Nuclear Regulatory Commission for Efficient and Effective Licensing of Accident Tolerant Fuels” (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:

All holders of 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 who 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 (ESP), combined license (COL), standard design certification (DC), standard design approval (DA), or manufacturing license (ML) referencing a small modular reactor (SMR) design under Title 10 of the Code of Federal Regulations (10 CFR) Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” SMRs 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 (C/Vs) that supply basic components to U.S. Nuclear Regulatory Commission (NRC) licensees under Title 10 of the Code of Federal Regulations (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities” or 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.”

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. Reviewers should ensure that applicants adequately address or disposition each of the criteria cited in the guidance as appropriate for the specific chromium coated cladding technology in reaching a reasonable assurance conclusion.

IMPLEMENTATION

The staff will use the information contained in this ISG to ensure that all known degradation and failure mechanisms for chromium-coated zirconium alloy fuel cladding are considered such that their impact on the acceptance criteria contained in SRP sections 4.2, 4.3, and 4.4 along with chapter 15 can be assessed.

BACKFITTING AND ISSUE FINALITY DISCUSSION

Discussion to be provided in final ISG.

CONGRESSIONAL REVIEW ACT

Discussion to be provided in final ISG.

FINAL RESOLUTION

By 2025, this information will be transitioned into Chapters 4 and 15 of the SRP. Following the transition of this guidance to the SRP, this ISG will be closed.

APPENDICES

A. Supplemental Guidance for SRP Chapters 4 and 15
B. Cladding Material Property Correlations
C. Specified Acceptable Fuel Design Limits (SAFDLs)
D. (Placeholder) Resolution of Public Comments

REFERENCES

1. Chromium-Coated Cladding Final PIRT Report, “Degradation and Failure Phenomena of Accident Tolerant Fuel Concepts: Chromium Coated Zirconium Alloy Cladding,” June 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19172A154)

2. NRC's Standard Review Plan, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (NUREG-0800)," (ADAMS Accession No. ML070810350)
3. NRC's ATF Project Plan, "Project Plan to Prepare the U.S. Nuclear Regulatory Commission for Efficient and Effective Licensing of Accident Tolerant Fuels," September 2018 (ADAMS Accession No. ML18261A414)

<u>Public Meetings</u>: August 6, 2019; December 4, 2019

APPENDIX A

Supplemental Guidance for SRP 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 SRP 4.2 covers additional requirements for review of complete fuel systems, cladding reviews cover these three areas:

- Definition of specified acceptable fuel design limits (SAFDLs) for new cladding,
- Material property correlations to be used in codes to ensure the new cladding satisfies the SAFDLs, and
- 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 ISG does not attempt to set standards for review of any credit or benefit applicants may request by demonstrating these improvements, as strategies for licensing these potential improvements have not yet been submitted to the Nuclear Regulatory Commission. The reviewer of any coated cladding must, therefore, evaluate any proposed property improvements against the data provided by the applicant, the commentary in the PIRT report, and the guidance in this ISG. The reviewer must also evaluate if the data provided supports the full operating domain for the fuel, and place appropriate limitations and conditions when necessary. Finally, if an applicant wishes to take credit for coating behavior up to a certain burnup, or during

certain accident conditions, it is necessary for the adherence of that coating to the substrate to have been justified for the full operating domain.

Definition of SAFDLs for New Cladding

The SAFDLs mentioned in SRP Section 4.2 under “SRP Acceptance Criteria, Design Bases” can be broadly separated into three general categories:

- SAFDLs related to fuel assembly performance that are typically addressed by simple calculation, manufacturing controls, and historical data
- SAFDLs related to fuel rod performance that are typically addressed for normal operation and anticipated operational occurrences (AOOs) using a thermal mechanical code
- 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 4.2 is included in Table 5.2 of the PIRT report and described in further detail in Appendix C of this ISG. These sections detail the expected and potential impact of the coatings on each SAFDL.

The reviewer should ensure that chromium-coated cladding submittals 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 alternate fashion. If a 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 10 CFR 50.46 prescriptive analytical limits, 2200°F (1204°C) peak cladding temperature (PCT) and 17% equivalent cladding reacted (ECR) maximum local oxidation, were established to preserve post-quench ductility (PQD) during a postulated LOCA. These analytical

limits for PQD were based on ring compression tests (RCT) 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 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 ID oxidation is based on zirconium alloy kinetics and cladding OD oxidation would be based on chromium coating kinetics. Hence, the applicability of the 17% ECR analytical limit and, more generally, the use of maximum local oxidation as a surrogate SAFDL for cladding embrittlement is questionable.

If the applicant elects to ignore the potential benefits expected with chromium coatings and continue to use the existing 50.46 analytical limits, then supporting evidence should be provided to demonstrate residual ductility of the coated cladding up to these analytical limits.

If the applicant elects to develop an alternative set of analytical limits, then additional research and testing would likely be necessary. NUREG/CR-7219 identifies new degradation mechanisms which need to be considered. Furthermore, additional degradation mechanisms exist for zirconium alloy cladding above 2200°F 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 1332°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 which 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 GWd/MTU.

<u>Material Property Correlations to Ensure SAFDLs are Met</u>

Appendix B provides a list of cladding material properties that are typically needed to adequately model fuel system response based on development and qualification of 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 details each of the twelve 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 Section 15 of the SRP.

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:

- Fuel temperature
- Fission gas release
- Rod internal pressure and void volume
- Cladding oxide thickness
- 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 code or other approach that will be used to assess performance against functional requirement
- Identification of approach to demonstrate high level of confidence that 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, 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, coolant flow rate, etc. is 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

Section 4.3 of NUREG-0800, (the Standard Review Plan) 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

Section 4.4 of NUREG-0800 covers the thermal hydraulic design for fuel assemblies, including critical heat flux (CHF) or critical power (CP) correlations. As discussed in Appendix C of 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 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 of 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 and/or boiling regime 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 Analyses

USFAR Chapter 15 provides demonstration that the Technical Specification (TS) Limiting Conditions of Operation, TS 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. Chapter 15 of NUREG-0800 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 (See DG-1327), and LOCA PCT and integral time-at-temperature analytical limits (See rulemaking on 10 CFR 50.46c) 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 which 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 TH 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 on the fuel rod's performance during the transient nor influence the overall accident progression. For example, PWR UFSAR Chapter 15.2 safety analyses demonstrates that over-pressure 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 modelled 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 (for example PWR excess steam demand or main steam line break, 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% permanent) or revised RIA PCMI cladding failure thresholds is 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 (for example, loss of A/C 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:

- 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 Chapter 15 AOOs and postulated accidents has been properly assessed. The scope of work needed to complete the 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 ^{235}U 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 Zr-alloy cladding under normal conditions and 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 NRC transient fuel performance code, FRAPTRAN (Geelhood K. , Luscher, Cuta, & Porter, 2016):

- High temperature ballooning behavior
- High temperature (800-1200°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 Cr-coated Zr cladding. The type of data that are typically used to justify each property will be stated. Currently it is not possible to definitively state what data are available to justify these properties, because small differences in applicant specific processes can have a significant impact on the properties. Therefore, the applicant should provide data or other justification from their 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 PIRT report. 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

Zr-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). Thermal conductivity data as a function of temperature from unirradiated samples have typically been used to develop cladding thermal conductivity correlations.

Cr-coated Zr

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 are generally low. This would be similar to the treatment of the ZrO2 that evolves on the surface of the Zr-alloy cladding.

B.2: Thermal Expansion

Zr-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 (Luscher, Geelhood, & Porter, 2015). Thermal expansion data as a function of temperature from unirradiated samples have typically been used to develop cladding thermal expansion correlations.

Cr-coated Zr

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, PVD coatings are usually dense and adherent, and plasma spray coatings can result in anisotropic mechanical properties due to the spray direction, i.e., in plane versus out of plane property differences. The effects of thermal expansion mis-match and their inherent interface strains can be mitigated by processing conditions. For instance, surface treatments that enhance surface area, strain tolerant microstructures, and higher ductility compliant layers can be utilized to reduce interface strains.

B.3: Emissivity

Zr-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 SBLOCA. The emissivity is impacted by the surface conditions including any oxide on the surface of the cladding, and for Zr-alloy typically increases with oxidation until saturation as the oxide becomes opaque.

Cr-coated Zr

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)

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

Zr-alloy Cladding

Cladding enthalpy and specific heat are not expected to change significantly with irradiation based on the currently available data. 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 (Luscher, Geelhood, & Porter, 2015). Enthalpy and/or specific heat data as a function of temperature from unirradiated samples would be useful to develop cladding enthalpy and specific heat correlations.

Cr-coated Zr

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 calculation of stored energy. This would apply equally to metallic and ceramic coatings.

B.5: Elastic Modulus

Zr-alloy Cladding

Cladding elastic modulus has been observed to be a weak function of fast neutron fluence (proportional to fuel burnup) (Geelhood, Beyer, & Luscher, PNNL Stress/Strain Correlation for Zircaloy. PNNL-17700, 2008). 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.

Cr-coated Zr

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) (Kim, et al., 2015) (Shahin, Petrik, Seshadri, Phillips, & Shirvan, 2018). 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

Zr-alloy Cladding

ASME BP&VC methods allow for use of either yield stress or ultimate tensile stress for 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.

Cr-coated Zr

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 (Brachet, et al., 2017) (Kim, et al., 2015) (Shahin, Petrik, Seshadri, Phillips, & Shirvan, 2018). 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

Zr-alloy Cladding

The creep behavior of zirconium alloy tubes has often been characterized by a thermal rate which can be developed based on ex-reactor creep tests, which are a function of stress and temperature, and an irradiation rate which 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. One example of this is the creep rates for recrystallized cladding and stress-relief annealed cladding in FRAPCON. Although both the thermal and irradiation creep rates are greater for the stress-relief annealed cladding than the recrystallized cladding, the two increases are not the same fraction so one increase could not be determined from the other (Geelhood K. , Luscher, Raynaud, & I.E., 2015) (Limback & Andersson, 1996). 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.

Cr-coated Zr
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 (Brachet, et al., 2017). 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 creep behavior of coated cladding differs from that of the uncoated substrate (Framatome, 2019). 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

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

Cr-coated Zr
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

Zr-alloy Cladding
The oxidation rate is important to model in uncoated cladding tubes as the zirconium oxide layer is less conductive than Zr metal. In the zirconium alloy systems, ex-reactor autoclave corrosion data is significantly different from in-reactor corrosion data and should not be used to develop corrosion correlations for coated parts. Additionally, the corrosion behavior of non-fueled 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) (Sabol, Comstock, Weiner, Larouere, & Stanutz, 1993) (Garde, Pati, Krammen, Smith, & Endter, 1993).

Cr-coated Zr
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 chromium 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 if there is aggressive corrosion at cracks or interfaces.

B.10: Hydrogen Pickup

Zr-alloy Cladding
It is important to quantify the hydrogen pickup in uncoated cladding tubes as hydrides in zirconium can lead to brittle behavior of the cladding (Zhao, et al., 2017). 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.

Cr-coated Zr
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 this might not be necessary for Cr-coated Zr cladding tubes. 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

Zr-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 if 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). 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.

Cr-coated Zr
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, so 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) 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.

B:12: High Temperature Steam Oxidation Rate

Zr-alloy Cladding

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

Cr-coated Zr

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 (SAFDLs)

C.1: SAFDLs Related to Assembly Performance

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 Zr-alloy tubes.

C.1.1: Rod Bow

Usually there is a penalty on departure from nucleate boiling ratio (DNBR) or margin to critical power ratio (MCPR) to account for bowing. The limits of what degree of bowing is acceptable will not change with the introduction of 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 non-uniformities 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 Section B.8 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 (PWR) or control blades (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 are not impacted by fuel rod performance, but rather by channel design (BWR) and guide tube design (PWR) and 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: SAFDLs Related to Rod Performance Assessed for Normal Operation and AOOs

Current codes that are informed by the properties in Section 5.1 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 ASME code (American Scociety of Mechanical Engineers, 2017). 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 Section A.6 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 (Geelhood, Beyer, & Luscher, 2008), 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). 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 PIRT report). 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). It is currently unknown if 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) 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). Because of this, fatigue data from irradiated cladding that was produced using a representative process for the applicant

in question is 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 the Standard Review Plan Section 4.2.

C.2.4: Cladding Oxidation, Hydriding, and CRUD

For Zr-alloy cladding, the cladding oxidation limit is designed to preclude oxide spallation that has typically been observed above 100 µm. Oxide spallation can lead to a local cool spot which 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 (>600ppm) 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 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 Cr_2O_3 rather than ZrO_2 and the Cr and/or Cr_2O_3 are expected to be a barrier against hydrogen uptake, and thus limits should be 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 be 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 the Standard Review Plan 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
- A description of any additional failures resulting from departure of nucleate boiling (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 of 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 Section B.7, 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 Section B.7.

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 departure from nucleate boiling, 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 AOO conditions, specified by the critical heat flux (CHF) or critical power (CP) for PWRs and BWRs, respectively. The effects of chromium coating on boiling crisis behavior are dispositioned in Section C.3.1, below.

C.3: SAFDLs Related to Fuel Rod Performance Assessed for Accident Conditions

Current codes that are informed by the properties in Appendix B 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 AOO analysis.

There is currently work underway to change some regulations (10CFR50.46c) and staff guidance (DG1327) for LOCA and RIA analysis. Neither of these is complete yet, so the discussion in this appendix will reflect the current regulations and staff guidance.

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 departure from nucleate boiling, 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 critical heat flux (CHF), and in BWRs the assembly power must be kept below the critical power (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), (Kandlikar, 2001), (O'Hanley, et al., 2013) though some studies have shown a noticeable difference between rough and very smooth surfaces (Weatherford, 1963). Surface porosity and wettability are thought to have a much more significant impact, as demonstrated by several experimental studies (Kandlikar, 2001), (Takata, Hidaka, Masuda, & Ito, 2003), (O'Hanley, et al., 2013). Boiling heat transfer experimental results indicate similar CHF for coated and uncoated cladding (Jo, Yeom, Gutierrez, Sridharam, & Corradini, 2018) (Jo, Gutierrez, Yeom, Sridharan, & Corradini, 2019).

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 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 zirconium 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 chromium-coated zirconium 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 1332°C. This formation should either be considered under this damage mechanism or under generalized cladding melting (Section B.3.7).

<u>C.3.2: Excessive Fuel Enthalpy</u>

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 of 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 post-irradiation 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.

<u>C.3.3: Bursting</u>

Bursting of the fuel rod relates to failure of fuel rods due to high temperature and high gas pressures during a LOCA. This can also be a consideration during 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 (Powers & Meyer, 1980). 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 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.

<u>C.3.4: Mechanical Fracturing</u>

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% of the irradiated

yield stress. This limit should not be exceeded for normal operation and AOOs. 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 Section B.6 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. 10 CFR 50.46 specifies a cladding temperature limit of 2200°F (1204°C) and a peak oxidation of 17% equivalent cladding reacted for Zr-alloy cladding (US Nuclear Regulatory Commission, 2017).

The PIRT ranked this damage mechanism as high. (See Appendix A of 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 if 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 test establishing new limits would be useful to demonstrate embrittlement will not occur. In addition to the tests performed to establish the ballooning (Section B.11) and high temperature oxidation behavior (Section B.12), some prototypic integral LOCA tests (see for example (Flanagan, Askeljung, & Puranen, 2013)) 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

Violent expulsion of fuel relates to the sudden increase in fuel enthalpy from an RIA that can result in 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. Typical limits for violent expulsion of fuel are:

- Peak radial average fuel enthalpy below 230 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 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 1852°C. For Zr alloy tubes the embrittlement limit of 1204°C (Section C.3.5) is more limiting. However, as discussed in Section B.3.5, 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 (1857°C) is virtually identical to that of Zr (1852°C). However, the formation of a low temperature eutectic between Cr and Zr at 1332°C occurs significantly lower than either of the individual melting temperatures. Formation of a low temperature eutectic with a thin coating may not represent 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 (Section C.3.1).

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 the degree of swelling not be underestimated and the balloon 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 (Powers & Meyer, 1980). 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 not be underestimated.

C.3.9: Structural Deformation

Structural deformation refers to externally applied loads during 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% 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 Section A.6 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% to 1% 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% to 0.1% strain) cyclic operation. Finally, cracking could occur during a design basis accident that causes large strain from pellet expansion (RIA) or gas overpressure and ballooning (LOCA).

The PIRT ranked this damage mechanism as high during accident conditions. (See Appendix A of 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 (Section C.2.2) and the cladding fatigue limit (Section C.2.3). 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 (Section B.11) and high temperature oxidation (Section B.12) correlations. If cracking is observed following ballooning, then high temperature oxidation correlations should be developed with consideration of cracked coating. Additionally, cladding embrittlement limits (Section C.3.5) 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 of 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 emergency core coolant system pump in the event of an accident if the debris filters become clogged with debris from delaminated coating. Debris clogging this pump has been identified as Generic Safety Issue 191 (GSI-191) (Shaffer, et al., 2005).

Delamination of the coating should be considered in the development of the cladding strain limit (Section C.2.2) and the cladding fatigue limit (Section C.2.3). 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 (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, the $ZrCr_2$ phase that could form due to interdiffusion could exhibit greater corrosion rate than bare Zr. Additionally, if this is the case, cladding embrittlement limits (Section C.3.5) should be developed with consideration of delaminated cladding. LOCA blowdown loads could also lead to delamination of the coating. To address GSI-191, the potential for delamination should be evaluated and accounted for following burst (Section C.3.3), mechanical fracture (Section C.3.4), ballooning (Section C.3.8), and structural deformation (Section C.3.9).

<u>C.4.3: Cr-Zr Interdiffusion</u>

As discussed in Section 4.2, 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 (300°C-350°C for 2000 days) 0.1 to 0.3 μm thick intermetallic layer
- Loss-of-coolant Conditions (800 to 1200°C for 1 hour) 0.2 to 1.4 μm thick intermetallic layer
- Long term Loss-of Coolant (800 to 1200°C for 1 day) 1 to 7 μm thick intermetallic layer.

Initial data from a number of programs has not observed significant interdiffusion in various coating concepts. It is noted that the numbers above are predictions based on limited data and should not be used without any data from a coating in question.

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 Cr/Zr 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 Cr/Zr 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 (Section C.3.1), 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 tests are performed in localized areas known to have the possibility for interdiffusion.

C.4.4: Radiation Effects on Cr

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 if this isotope will be released to the coolant in significant quantities. For a CrN coating, the nitrogen will lead to the production of some C-14.

A second concern is what the impact of fast neutron irradiation on Cr metal and other Cr containing compounds will be. In zirconium, fast neutron irradiation leads to a dramatic increase in strength and reduction in ductility (Geelhood, Beyer, & Luscher, 2008). Recent ion beam irradiation data indicated that cold spray Cr-coatings are more resistant to radiation defects than bulk Cr. (Maier B. , et al., 2018)

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 we 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 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.

Data may already be available for radiation damage, as chromium-containing alloys and chromium 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). The PIRT ranked this damage mechanism as high during accident conditions. (See Appendix A of the PIRT report). Each process in question should define the allowable defects and justify the presence of these defects based on 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 1332°C. If operation beyond the current design basis temperature limit of 1200°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.