



U.S. NUCLEAR REGULATORY COMMISSION

DRAFT REGULATORY GUIDE DG-1263, Revision 1

Proposed New Regulatory Guide 1.224

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ESTABLISHING ANALYTICAL LIMITS FOR ZIRCONIUM-BASED ALLOY CLADDING

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes methods for establishing analytical limits for the ductile-to-brittle transition and breakaway oxidation susceptibility for zirconium-alloy cladding. This guide also provides a hydrogen generation limit for fuel system designs with zirconium alloy cladding to minimize the risks associated with explosive quantities of combustible gas in containment.

Applicability

This RG applies to applicants for and holders of construction permits and operating licenses for power reactors under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities” (Ref. 1), and applicants for and holders of standard design approvals, combined licenses, and manufacturing licenses, and applicants for standard design certifications (including an applicant after the Commission has adopted a final design certification regulation), under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants” (Ref. 2).

Applicable Regulations

- 10 CFR Part 50, including Appendix A, “General Design Criteria for Nuclear Power Plants,” provides regulations for licensing production and utilization facilities.
 - 10 CFR 50.46a, “Alternative acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” provides a voluntary alternative to 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.” Licensees that adopt 10 CFR 50.46a are required to have NRC-approved limits that address cladding degradation phenomena and that avoid explosive concentrations of combustible gas.
 - 10 CFR Part 50, Appendix A, General Design Criterion 35, “Emergency core cooling,” requires that the emergency core cooling system (ECCS) be designed to remove heat from the

This RG is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not received final staff review or approval and does not represent an NRC final staff position. Public comments are being solicited on this DG and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal rulemaking website, <http://www.regulations.gov>, by searching for draft regulatory guide DG-1263 Revision 1. Alternatively, comments may be submitted to Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Rulemakings and Adjudications Staff. Comments must be submitted by the date indicated in the *Federal Register* notice.

Electronic copies of this DG, previous versions of DGs, and other recently issued guides are available through the NRC’s public website under the Regulatory Guides document collection of the NRC Library at <https://nrc.gov/reading-rm/doc-collections/reg-guides>. The DG is also available through the NRC’s Agencywide Documents Access and Management System (ADAMS) at <http://www.nrc.gov/reading-rm/adams.html>, under Accession No. ML24304A946. The regulatory analysis is associated with a rulemaking and may be found in ADAMS under Accession No. ML26181A099.

reactor core following a LOCA to prevent fuel and cladding damage that could interfere with effective core cooling and to limit clad metal-water reaction to negligible amounts.

- 10 CFR 50.46 includes cladding temperature and oxidation limits meant to ensure post-quench ductility (PQD), a limit on allowable hydrogen generation, and a requirement to maintain a coolable core geometry. While 10 CFR 50.46 does not specifically require periodic breakaway oxidation testing or testing for PQD, the methods in this RG could be used to confirm that new zirconium-based cladding alloys meet the requirements of 10 CFR 50.46.
- 10 CFR Part 52 governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities. The regulations in 10 CFR 52.47, “Contents of applications; technical information”; 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report”; 10 CFR 52.137, “Contents of applications; technical information”; and 10 CFR 52.157, “Contents of applications; technical information in final safety analysis report,” for applications for standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities, respectively, state that the applicable 10 CFR Part 50 regulations cited above are also required for applicants for and holders combined licenses, standard design approvals, and manufacturing licenses and applicants for a standard design certification (including an applicant after the Commission has adopted a final design certification regulation). There are no similar requirements related to ECCS performance for applicants for and holders of early site permits.

Related Guidance

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP) (Ref. 3), provides guidance to the NRC staff for the review of license applications and license amendments for nuclear power plants.
 - SRP Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary,” provides guidance for reviewing LOCAs.
 - SRP Section 4.2, “Fuel System Design,” provides guidance for reviewing reactor fuel designs.
- RG 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance” (Ref. 4), provides guidance for calculating realistic or best-estimate ECCS performance during LOCAs.
- RG 1.203, “Transient and Accident Analysis Methods” (Ref. 5), describes a process that the staff considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.
- Draft Regulatory Guide (DG)-1261, Revision 1 (proposed new RG 1.222), “Conducting Periodic Testing for Breakaway Oxidation Behavior” (Ref. 6), describes a method to address a zirconium-based cladding alloy embrittlement mechanism known as breakaway oxidation.
- DG-1262, Revision 1 (proposed new RG 1.223), “Testing for Post-Quench Ductility” (Ref. 7), provides a method for measuring the ductile-to-brittle transition for a zirconium-based cladding alloy as a function of hydrogen content.

Purpose of Regulatory Guides

The NRC issues RGs to describe methods that are acceptable to the staff for implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific issues or postulated events, and to describe information that the staff needs in its review of applications for permits and licenses. Regulatory guides are not NRC regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs are acceptable if the applicant provides sufficient basis and information for the NRC staff to verify that the alternative methods comply with the applicable NRC regulations.

Paperwork Reduction Act

This RG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), under control numbers 3150-0011 and 3150-0151, respectively. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch, Office of the Chief Information Officer, Mail Stop: T6-A10M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001 or by e-mail to Infocollects.Resource@nrc.gov, and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0011 and 3150-0151), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW, Washington, DC, 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

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B. DISCUSSION

Reason for Issuance

Licensees implementing the alternative ECCS requirements in 10 CFR 50.46a must have NRC-approved limits that address cladding degradation phenomena, maintain fuel coolability, avoid explosive concentration of combustible gas, and demonstrate sufficient decay heat removal. This guide provides acceptable limits and methods to meet the cladding degradation and combustible gas generation requirements for fuel system designs using uranium oxide or mixed uranium-plutonium oxide pellets within zirconium-alloy cladding. More specifically, this guide provides methods to establish analytical limits to limit cladding embrittlement and to ensure cladding post-quench ductility following a postulated LOCA. This RG also provides guidance regarding the development of hydrogen uptake models, which are needed to establish hydrogen-dependent post-quench ductility analytical limits, and guidance on how to consider oxygen diffusion from inside surfaces in an ECCS evaluation. Finally, this guide provides a hydrogen generation limit that precludes the formation of a combustible gas mixture in the reactor coolant system or in containment. This limit is identical to the historical limit included in 10 CFR 50.46(b)(3). The basis for this limit is discussed elsewhere (e.g., in the U.S. Atomic Energy Commission opinion paper (Ref. 8) that established the ECCS requirements in 10 CFR 50.46).

On May 23, 2025, President Donald J. Trump signed Executive Order (E.O.) 14300, “Ordering the Reform of the Nuclear Regulatory Commission” (Ref. 9). Section 5, “Reforming and Modernizing the NRC’s Regulations,” requires the NRC to undertake a review and wholesale revision of its regulations and guidance documents as guided by the policies set forth in section 2 of the E.O. Additionally, as part of the NRC’s overarching review of all of its regulations, the agency identified a number of additional changes to the NRC’s regulations that will improve the efficiency and efficacy of its licensing process that are also included in this RG.

Background

In 1996, the NRC initiated a fuel-cladding research program intended to investigate the behavior of high-burnup fuel and high-exposure fuel cladding under accident conditions. This program included an extensive LOCA research and testing program at Argonne National Laboratory, as well as jointly funded programs at the Kurchatov Institute (see NUREG/IA 0211, “Experimental Study of Embrittlement of Zr 1%Nb VVER Cladding under LOCA-Relevant Conditions” (Ref. 10)) and the Halden Reactor Project (see IFE/KR/E 2008/004, “LOCA Testing of High Burnup PWR Fuel in the HBWR. Additional PIE [post-irradiation examination] on the Cladding of the Segment 650.5” (Ref. 11)), to develop the body of technical information needed to evaluate LOCA regulations for high-burnup fuel. The research findings were summarized in Research Information Letter-0801, “Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46,” dated May 30, 2008 (Ref. 12). Most of the detailed experimental results from the program at Argonne National Laboratory appear in NUREG/CR-6967, “Cladding Embrittlement during Postulated Loss-of-Coolant Accidents,” issued July 2008 (Ref. 13), and NUREG/CR-7219, “Cladding Behavior during Postulated Loss-of-Coolant Accidents,” issued July 2016 (Ref. 14).

The research results revealed that hydrogen, which is absorbed into the cladding during the waterside corrosion process under normal operation that increases with fuel burnup, has a significant influence on embrittlement during a postulated LOCA. When that cladding is exposed to high-temperature LOCA conditions, the elevated hydrogen levels increase the solubility and the rate of diffusion of oxygen in the metal. Thus, for cladding exposed to high-temperature LOCA conditions, embrittlement can occur for increasingly shorter periods of high-temperature steam oxidation for increasing hydrogen pickup during normal operation (which is correlated with increasing fuel burnup).

The research results also revealed that an embrittlement mechanism referred to as “breakaway oxidation” might occur during prolonged exposure to elevated cladding temperature during a LOCA.

The NRC’s LOCA research program identified that, for high-burnup fuel, oxygen can diffuse into the cladding metal during a LOCA from the interior diameter (ID) even when no steam oxidation is occurring on the ID (See IFE/KR/E 2008/004 and Research Information Letter-0801). The ID oxygen diffusion phenomenon was discovered in the United States in 1977, confirmed by tests in Germany in 1979, and is seen in the Halden results (See IFE/KR/E 2008/004). Combined with oxidation on the cladding outside diameter, oxygen ingress from the cladding ID would further limit integral time at temperature to nil ductility.

Existing Embrittlement Database

NUREG/CR-6967 summarizes most of the cladding embrittlement experimental results from the NRC’s LOCA research program. Since the publication of NUREG/CR-6967 in 2008, additional testing was conducted, focusing on cladding materials with hydrogen contents in the range of 200–350 weight parts per million (wppm). Additional oxidation and PQD tests were conducted with cladding samples sectioned from high-burnup ZIRLO^{®1} defueled segments characterized by a 25–30 micrometer corrosion-layer thickness and 300–340 wppm of hydrogen in the cladding metal before oxidation. Also, the ductility data for an oxidation sample with approximately 600 wppm hydrogen were reassessed. In addition, since the publication of NUREG/CR-6967, oxidation and PQD tests were conducted with pre-hydrided cladding samples containing 200–300 wppm of hydrogen. These additional tests are summarized in NUREG/CR-7219.

The more recent data published in NUREG/CR-7219 were combined with the data reported in NUREG/CR-6967 to generate a more robust and informed description of cladding embrittlement as a function of hydrogen content. The resulting behavior description of cladding embrittlement as a function of hydrogen content is shown in figure 1.

For modern as-fabricated cladding (Zircaloy-2, Zircaloy-4, ZIRLO[®], and M5^{®2}), embrittlement thresholds cluster at 19–20 percent equivalent cladding reacted (ECR), as calculated using the Cathcart-Pawel (CP) correlation (CP-ECR), as compared to 16 percent CP-ECR for older Zircaloy-4 cladding. However, this improvement relative to the as-fabricated cladding specimens used to develop the existing acceptance criteria for cladding ductility in 10 CFR 50.46(b) is negated for modern claddings with hydrogen pickup as low as 100 wppm. A bilinear function for CP-ECR versus hydrogen content was used to fit the embrittlement data for pre-hydrided and high-burnup cladding. The embrittlement rate is steep for cladding with ≤ 400 -wppm hydrogen. For higher hydrogen content, the embrittlement rate is more gradual. Embrittlement is highly sensitive to both hydrogen content and peak oxidation temperature, and both of these factors should be considered when determining acceptable limits.

The tests performed in the NRC’s LOCA research program were conducted at peak cladding temperatures (PCTs) less than or equal to approximately 1,200 degrees Celsius (°C) (approximately 2,200 degrees Fahrenheit (°F)). Above this temperature, oxygen solubility and diffusion in the beta zirconium layer become high enough to lead to cladding embrittlement following cladding cooldown and quench (Ref. 14). This is the basis for the limit of 1,204°C (2,200°F) in 10 CFR 50.46(b)(1), as discussed in Refs. 8 and 14.

¹ ZIRLO is a registered trademark of Westinghouse Electric Company LLC.

² M5 is a registered trademark of AREVA.

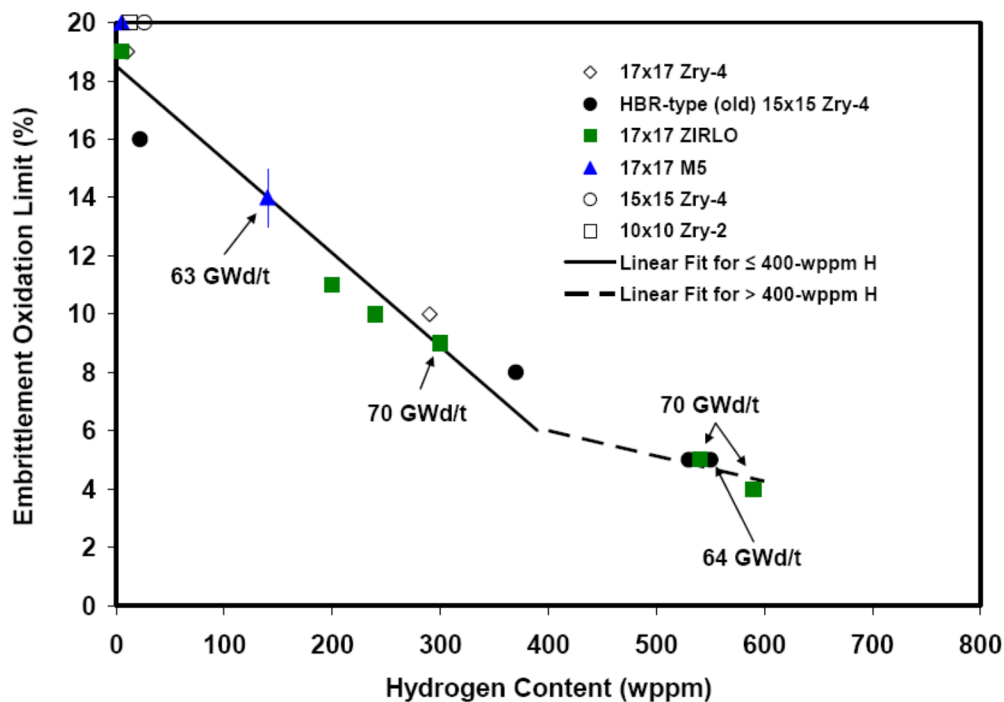


Figure 1. Ductile-to-brittle transition oxidation level (CP-ECR) as a function of pretest hydrogen content in cladding metal for as-fabricated, pre-hydrided, and high-burnup cladding materials. Samples were oxidized at $\leq 1,200\text{ }^{\circ}\text{C} \pm 10\text{ }^{\circ}\text{C}$ and quenched at $800\text{ }^{\circ}\text{C}$. For high-burnup cladding with about 550-wppm hydrogen, embrittlement occurred during the heating ramp at 1,160–1,180 $^{\circ}\text{C}$ peak oxidation temperatures (Ref. 14).

Experimental Techniques

In 10 CFR 50.46a(f)(1)(i), the NRC requires the licensee to establish analytical limits to address cladding degradation phenomena for all fuel system designs. To prevent adverse cladding degradation for zirconium-based alloy cladding, licensees should establish limits on PCT and integral time at temperature, which correspond to the measured ductile-to-brittle transition for the zirconium-alloy cladding material. One NRC-approved experimental technique for measuring the ductile-to-brittle transition for the zirconium-alloy cladding material is provided in DG-1262, but other methods may be submitted for NRC review and approval. Licensees should also establish limits for breakaway oxidation. One NRC-approved experimental technique for measuring breakaway oxidation behavior for zirconium-alloy cladding material is provided in DG-1261, but other methods may be submitted for NRC review and approval. Throughout this guide, when reference is made to “data generated with an NRC-approved experimental technique,” the experimental techniques may be those provided in DG-1261 and DG 1262, or a different NRC-approved experimental technique.

Consideration of International Standards

The International Atomic Energy Agency (IAEA) works with member states and other partners to promote the safe, secure, and peaceful use of nuclear technologies. The IAEA develops Safety

Requirements and Safety Guides for protecting people and the environment from harmful effects of ionizing radiation. This system of safety fundamentals, safety requirements, safety guides, and other relevant reports, reflects an international perspective on what constitutes a high level of safety. To inform its development of this RG, the NRC considered IAEA Safety Requirements and Safety Guides pursuant to the Commission's International Policy Statement (Ref. 15) and Management Directive and Handbook 6.6, "Regulatory Guides" (Ref. 16).

The following IAEA Safety Requirements and Guides were considered in the development of the Regulatory Guide:

- IAEA SSG-52, "Design of the Reactor Core for Nuclear Power Plants," issued in 2019 (Ref. 17), states that limits should be established to prevent cladding embrittlement due to oxidation at high temperature.
- IAEA SSG-56, "Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants," issued in 2020 (Ref. 18), includes high-level information about ECCS requirements for loss-of-coolant accidents.

C. STAFF REGULATORY GUIDANCE

This RG describes an approach to establish analytical limits for PQD (Section C.1) and breakaway oxidation (Section C.2) for zirconium-alloy cladding materials that the NRC considers acceptable to implement the requirement in 10 CFR 50.46a(f)(1)(i). All of these approaches use a PCT no greater than 1,204°C (2,200°F), which would also prevent failure due to cladding degradation mechanisms that occur at higher temperatures (e.g., excessive exothermic metal-water reaction, alloy-specific eutectics, and loss of fuel rod geometry due to plastic deformation). Section C.3 provides an acceptable limit on hydrogen generation. Finally, Section C.4 provides guidance to address the impact of oxygen diffusion from the cladding inside surfaces (if an oxygen source is present on the inside surface of the cladding at the onset of the LOCA) on cladding embrittlement.

C.1 Analytical Limits for Post-Quench Ductility

In 10 CFR 50.46a(f)(1)(i), the NRC requires the licensee to establish analytical limits to address cladding degradation phenomena for all fuel system designs. To prevent adverse cladding degradation for zirconium-based alloy cladding, licensees should establish limits on PCT and integral time at temperature, which correspond to the measured ductile-to-brittle transition for the zirconium-alloy cladding material. There are four methodologies outlined in this section; the use of any one of these options is an acceptable method to meet the requirement to establish an analytical limit for the cladding degradation phenomenon of post-quench ductility. Section C.1.A of this guide provides an acceptable post-quench ductility analytical limit for the zirconium-alloy cladding materials tested in the NRC's LOCA research program. Section C.1.B of this guide describes a method to establish the analytical limit in figure 2 of this guide for cladding alloys not tested in the NRC's LOCA research program. Section C.1.C of this guide describes a method to establish a cladding-specific analytical limit other than the limits provided in figure 2 of this guide. Section C.1.D of this guide describes methods for establishing analytical limits for zirconium-alloy cladding materials at peak oxidation temperatures less than 1,204°C (2,200°F). Section C.1.E of this guide provides guidance related to hydrogen pickup models. Section C.1.F of this guide provides general requirements applicable to all analytical limits established for post-quench ductility.

C.1.A An Acceptable Post-Quench Ductility Analytical Limit for Zircaloy-2, Zircaloy-4, ZIRLO[®], M5[®] and Optimized ZIRLO^{TM3}

The analytical limits defined in figure 2 are acceptable for the zirconium-alloy cladding materials tested in the NRC's LOCA research program, which were Zircaloy-2, Zircaloy-4, ZIRLO[®], and M5.

Westinghouse's approved cladding alloy Optimized ZIRLOTM was not tested as part of NRC's LOCA research program. Westinghouse submitted a comment on DG-1263, Revision 0, requesting that Optimized ZIRLOTM be included as one of the acceptable cladding materials for which figure 2 is applicable. The NRC staff conducted an audit of the supporting PQD experimental procedures and results and concluded that figure 2 is applicable to Optimized ZIRLOTM. The basis of the staff's finding is documented in Reference 19.

³ Optimized ZIRLO is a trademark of Westinghouse Electric Company LLC.

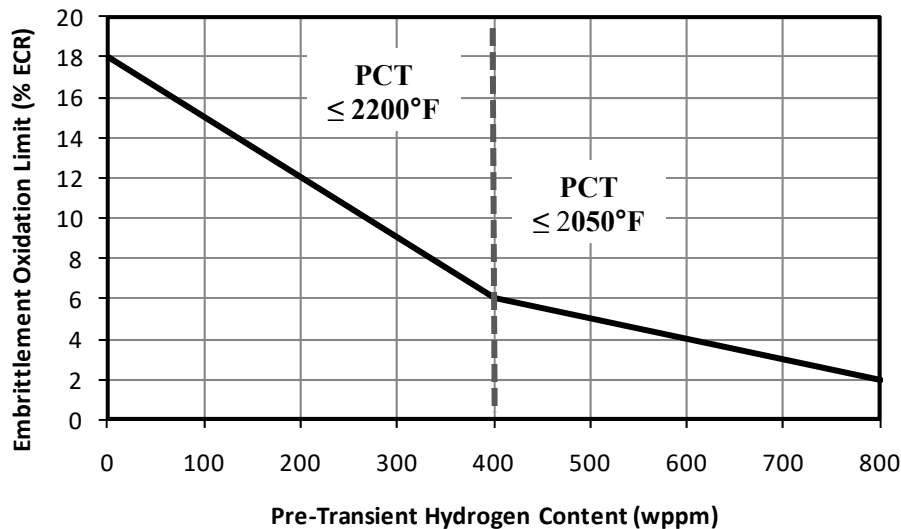


Figure 2. Acceptable analytical limits for PQD and integral time at temperature (as determined in local oxidation calculations using the CP correlation).

Licensees using these alloys may accept this analytical limit without further testing, provided that the test conditions used in the NRC’s LOCA research program are relevant to the calculated ECCS performance and provided that local oxidation is calculated using the CP correlation. Additional discussion is provided below.

The database established in the NRC’s LOCA research program and the resulting analytical limit in figure 2 were intended to provide a best-estimate limit for the ductile-to-brittle transition for zirconium alloys. The curve is a “best estimate” in that the NRC has confidence that the transition from ductile to brittle lies near, but not below, the line. Because PQD tests on materials with greater than 400 wppm hydrogen were conducted at a peak oxidation temperature below 1,204°C (2,200°F), a separate PCT analytical limit must be defined that is consistent with the peak oxidation temperature achieved during the test. The limits on peak cladding temperature of 1,204°C (2,200°F) for materials with less than 400 wppm cladding hydrogen content and 1,121°C (2,050°F) for materials with 400 wppm cladding or greater hydrogen content are acceptable.

The analytical limit defined in figure 2 is applicable for plants equipped with ECCS designs that are bounded by the oxidation conditions of a peak cladding temperature of 1,204°C (2,200°F) and quench temperature of 800°C (1,472°F). In the NRC test program, experiments were conducted at maximum oxidation temperatures less than or equal to 1,200 plus or minus 10°C (2,192 plus or minus 18°F) and quenched at 800°C (1,472°F). These test conditions were selected with the objective of bounding the performance of ECCSs. They are considered relevant and bounding for current light-water reactor ECCSs. However, it may be necessary to evaluate and possibly modify the conditions accordingly for ECCSs of new reactor designs. In addition, post-quench ductility measurements were made at 135°C (275°F). During the development of the original ECCS rule, investigators suggested considering a temperature for post-quench mechanical tests no higher than the saturation temperature during re-flood (i.e., about 135°C or 275°F). This test condition is considered relevant for current light-water reactor ECCSs. However, it may be necessary to evaluate and possibly modify the conditions accordingly for ECCSs of new reactor designs.

The ductile-to-brittle threshold defined in figure 2 is an acceptable analytical limit on integral time at temperature as calculated using local oxidation calculations and the CP correlation. Use of figure 2 as an acceptable analytical limit necessitates that the LOCA method also employ the CP correlation to integrate time-at-temperature. Use of a different correlation to calculate the metal-water reaction heat generation rate is acceptable provided it has been approved for that application. Appendix K LOCA evaluation models should continue to use the Baker-Just correlation for this purpose but needs to use the CP correlation to integrate time-at-temperature for figure 2 to be applicable.

C.1.B Methodology for Demonstrating Consistency with the Existing NRC Database for New Cladding Alloys

The analytical limits defined in figure 2 of this RG can be established for zirconium-alloy cladding materials not tested in the NRC's LOCA research program by demonstrating comparable post-quench cladding performance with the analytical limit defined in figure 2. It is acceptable to demonstrate consistency by showing that data generated with an NRC-approved experimental technique to identify the ductile-to-brittle transition CP-ECR for the cladding material is greater than or equal to the analytical limits provided in figure 2. The ductile-to-brittle transition CP-ECR should be identified for as-received cladding material and for at least two hydrogen content levels: (1) within 100 wppm of the maximum hydrogen content specified at end of life (EOL) and (2) within 100 wppm of half of the maximum hydrogen content specified at EOL material. For new cladding alloys that meet conditions described below, it is acceptable to test only un-irradiated cladding samples that are pre-charged with hydrogen. For new cladding alloys that do not meet the conditions below, testing of irradiated material is also required to demonstrate comparable post-quench cladding performance with the analytical limit defined in figure 2.

The NRC's LOCA research program included irradiated zirconium-alloy cladding samples, as well as un-irradiated zirconium-alloy cladding samples pre-charged with hydrogen. For the materials tested, un-irradiated samples were pre-charged with hydrogen embrittled at the same CP-ECR level as the irradiated samples with the same pre-transient hydrogen content. Therefore, for the cladding alloys tested, Zircaloy-2, Zircaloy-4, ZIRLO[®] and M5[®], it is presumed that un-irradiated pre-charged samples are an adequate surrogate for irradiated cladding. Further, for new cladding alloys that (1) use the Kroll process as the reduction method, (2) operate less than or equal to the maximum fluence, (3) include only the alloying elements present in the materials tested and (4) have similar alloying content of each element to the materials tested in the NRC's LOCA research program, it can also be assumed that un-irradiated pre-charged samples are an adequate surrogate for irradiated cladding. The research results documented in NUREG/CR-6967 and NUREG/CR-7219 revealed that alloy composition has a minor effect on embrittlement associated with high-temperature steam oxidation, in contrast to the finding that alloy composition can have a significant effect on breakaway oxidation behavior. Therefore, when considering high-temperature steam oxidation, alloys with similar alloying content are expected to have similar embrittlement behavior. In this case, "similar alloying content" of each element is defined by less than or equal to 25 percent deviation from the alloying limits defined for the tested alloy. For example, the chemical requirements for Zircaloy-2 in ASTM B351/B351M, "Standard Specification for Hot-Rolled and Cold-Finished Zirconium and Zirconium Alloy Bars, Rod, and Wire for Nuclear Application" (Ref. 20), defines the elemental composition range for tin to be 1.20–1.70 weight percent. A new cladding alloy that falls within all chemical requirements defined in ASTM B351/B351M but has a tin content between 0.90 and 2.13 weight percent would be considered to have a similar alloying content to Zircaloy-2. Other definitions of "similar alloying content" may also be acceptable and can be submitted to the NRC for review and approval in the license amendment request or vendor topical report for a new fuel design.

If the new cladding alloy does not conform to the criteria above, then testing on irradiated material is required and the irradiated testing must confirm the desired ECR limit curve. If the ductile-to-

brittle transition CP-ECR determined from tests on irradiated material is less than the ductile-to-brittle transition CP-ECR determined from as-received and hydrogen pre-charged samples, then the testing should be expanded such that the burnup-embrittlement dependency of the alloy can be determined. The methodology for establishing a zirconium alloy specific limit provided in Section C.1.C below should be followed in this case.

To demonstrate comparable performance with the existing NRC database and adoption of the analytical limits provided in this guide for a new fuel design, the applicant would submit experimental results as part of the documentation supporting its request for NRC approval of the new fuel design (e.g., license amendment request or vendor topical report). The applicant should provide details of the experimental technique (unless the submitted material states that the experiments were conducted in accordance with DG-1262) and the results of experiments conducted with as-fabricated, pre-hydrided, and/or irradiated cladding material.

C.1.C Methodology for Establishing a Zirconium-Alloy-Specific Limit on Post-Quench Ductility

The existing NRC database and the resulting analytical limits described in this regulatory guide are intended to provide a best-estimate limit for the ductile-to-brittle transition for zirconium alloys. In some instances, a zirconium-alloy cladding material may experience the transition from ductile to brittle behavior at a higher or lower level of oxidation than that established by the NRC's research program. Thus, analytical limits other than those defined in figure 2 of this RG can be established for zirconium-alloy cladding materials. The ductile-to-brittle transition CP-ECR should be identified using an acceptable experimental technique for as-received cladding material and for hydrogen pre-charged cladding material with at least four hydrogen content levels within the range defined by the maximum hydrogen content specified at EOL. If testing on irradiated material is required as noted above, the ductile-to-brittle transition CP-ECR should also be identified through testing of irradiated cladding material with at least four hydrogen content levels within the range defined by the maximum hydrogen content specified at EOL. The analytical limit should be defined as a "best estimate" line, where "best estimate" means that the transition from ductile to brittle behavior lies near, but not below, the line.

To establish a zirconium-alloy-specific limit for a new or existing fuel design, the applicant should provide experimental results as part of the documentation supporting the applicant's request for NRC approval of the new or existing fuel design (e.g., license amendment request or vendor topical report). The applicant should provide details of the experimental technique (unless the submitted material states that the experiments were conducted in accordance with DG-1262) and the results of experiments conducted with as-fabricated, pre-hydrided, and irradiated material, as appropriate, as well as a specified analytical limit on peak cladding temperature and integral time at temperature that corresponds to the measured ductile-to-brittle transition for the zirconium-alloy cladding material.

C.1.D Methodology for Establishing Analytical Limits for Post-Quench Ductility at Peak Oxidation Temperatures Less than 1,204°C (2,200°F)

Analytical limits other than those defined in figure 2 of this RG can be established for zirconium-alloy cladding materials at a peak cladding temperature lower than 1,204°C (2,200°F). The ductile-to-brittle transition CP-ECR should be identified using an acceptable experimental technique for as-received cladding material and for hydrogen pre-charged cladding material with at least four hydrogen content levels within the range defined by the maximum hydrogen content specified at EOL. If testing on irradiated material is required as noted above, the ductile-to-brittle transition CP-ECR should also be identified through testing of irradiated cladding material with at least four hydrogen content levels within the range defined by the maximum hydrogen content specified at EOL. The hypothesis that un-irradiated pre-charged samples are an adequate surrogate for irradiated cladding has not been extensively

investigated at temperatures below 1,204°C (2,200°F). The analytical limit should be defined as a “best estimate” line, where “best estimate” means that the transition from ductile to brittle behavior lies near, but not below, the line.

The existing NRC database and the resulting analytical limits described in this regulatory guide are intended to bound ECCS performance of current light-water reactor designs. In the NRC research program, experiments were conducted at maximum oxidation temperatures less than or equal to 1,200 plus or minus 10°C (2,192 plus or minus 18°F) and quenched at 800°C (1,472°F). Some ECCSs may perform such that the maximum oxidation temperature is significantly below 1,204°C (2,200°F). Oxidation at lower temperatures has been shown to increase the allowable calculated oxidation before embrittlement. Therefore, conducting tests at lower peak temperatures may provide additional margin for some zirconium-alloy cladding materials.

To establish analytical limits at peak oxidation temperatures less than 1,204°C (2,200°F), the applicant should provide experimental results as part of the documentation supporting its request for NRC approval of the new fuel design or existing fuel design (e.g., license amendment request or vendor topical report). The applicant should provide details of the experimental technique (unless the submitted material states that the experiments were conducted in accordance with DG-1262) and the results of experiments conducted with as-fabricated, prehydrided, and irradiated material, as appropriate, as well as a specified analytical limit on peak cladding temperature and integral time at temperature that corresponds to the measured ductile-to-brittle transition for the zirconium-alloy cladding material.

For a given zirconium alloy, an applicant is permitted to define an analytical limit on integral time at temperature (CP-ECR as a function of cladding hydrogen) corresponding to different peak cladding temperature analytical limits. This approach may provide margin for high-burnup, high-corrosion, low-power fuel rods where the calculated cladding temperature is significantly less than lower burnup fuel rods.

C.1.E Hydrogen Pickup Models

An applicant will need an alloy-specific cladding hydrogen uptake model if the applicant chooses to use the hydrogen-dependent embrittlement threshold provided in this RG. Appendix A of this RG provides acceptable fuel rod cladding hydrogen uptake models for the current commercial zirconium alloys.

C.1.F Demonstrating Compliance with the Analytical Limit for Post-Quench Ductility

Based on the approved ECCS evaluation models and methods, the applicant should identify the limiting combination of break size, break location, and initial conditions and assumptions that maximize predicted PCT and local oxidation (surrogate for time at temperature).

Appropriate combinations of initial conditions and uncertainties will vary for different design conditions as well as the regulatory requirements under which the evaluation model has been developed, including evaluation models that satisfy the following:

- the required and acceptable features of Appendix K, “ECCS Evaluation Models,” to 10 CFR Part 50
- the evaluation model requirements for realistically considering the behavior of the reactor system during a LOCA, with consideration of uncertainties, specified in 10 CFR 50.46(a)(3)(i), or for

postulated LOCAs smaller than the transition break size, in 10 CFR 50.46a(e)(2)

- the requirements for realistic methods (with no consideration of uncertainties), as described in 10 CFR 50.46a(e)(3), which are intended for postulated LOCAs larger than the transition break size where licensees have demonstrated such breaks may be addressed as beyond-design-basis events

Separate cases might be necessary to identify the limiting scenario for PCT relative to local oxidation and vice versa. The applicant should demonstrate that the predicted PCT remains below the lesser of 1,204°C (2,200°F) and the maximum oxidation PQD temperature for the relevant cladding hydrogen content. The applicant should also demonstrate that the maximum predicted local oxidation remains below the established PQD analytical limits.

Because of the strong relation between allowable local oxidation and cladding hydrogen content, the applicant may elect to subdivide the fuel rods within the core based on characteristics such as cladding hydrogen content, burnup, fuel rod power, or a combination thereof. For example, PCT and local oxidation calculations could be performed on three representative sets of fuel rods (e.g., 0–30 gigawatt-days per metric ton of uranium (GWd/MTU), 30–45 GWd/MTU, and 45–62 GWd/MTU) using bounding power histories for each fuel rod grouping. The predicted PCT and local oxidation would then be compared to the analytical limit for that range of burnup/hydrogen.

Licensees may re-insert older, irradiated fuel bundles residing in the spent fuel pool. Further, all fuel bundles delivered to a licensee prior to the date of the issuance of the license amendment authorizing the licensee's use of 10 CFR 50.46a, including bundles clad with a currently available commercial cladding alloy (e.g., Zircaloy-2, Zircaloy-4) or legacy zirconium alloy no longer commercially available, may use the analytical limits in figure 2 to show compliance with 10 CFR 50.46a.

C.2 Breakaway Oxidation

C.2.A Breakaway Oxidation Analytical Limits

In 10 CFR 50.46a, the NRC requires applicants to address cladding degradation phenomena, which, for licensees using zirconium-based alloys, includes breakaway oxidation.

To establish a zirconium-alloy-specific time limit for a new or existing fuel design, the applicant should provide experimental results, using an NRC-approved experimental technique, from testing for breakaway oxidation behavior as part of the documentation supporting its request for NRC approval of the new or existing fuel design (e.g., through a license amendment request or vendor topical report). The applicant should provide details of the experimental technique (unless the submitted material states that the experiments were conducted in accordance with DG-1261, Revision 1) and the results of experiments conducted.

Applicants may elect to establish the analytical time limit for breakaway oxidation with conservatism relative to the measured minimum time (i.e., reduce the time) to the onset of breakaway oxidation. This approach may reduce the likelihood of reassessing small-break LOCA cladding temperature histories in the event of a minor change in measured time to breakaway oxidation. For example, the minimum time to breakaway oxidation may be demonstrated to occur at 975°C (1,787°F) at a time of 4,000 seconds. An applicant may elect to establish an analytical limit of 3,000 seconds for the total accumulated time that the cladding may remain above 800°C (1,472°F).

Licensees may re-insert older, irradiated fuel bundles residing in the spent fuel pool. Further, all fuel bundles delivered to a licensee prior to the date of the issuance of the license amendment authorizing the licensee's use of 10 CFR 50.46a that are clad with a currently available commercial cladding alloy

(e.g., Zircaloy-2, Zircaloy-4), may use the analytical limit established for the current version of those commercial alloys to show compliance with 10 CFR 50.46a. All fuel bundles delivered to a licensee prior to the date of the issuance of the license amendment authorizing the licensee's use of 10 CFR 50.46a that are clad with a legacy zirconium alloy that is no longer commercially available may apply a default analytical limit of 3,500 seconds to show compliance with 10 CFR 50.46a.

C.2.B Temperature of Susceptibility for Breakaway Oxidation

The staff has determined that 800°C (1,472°F) is an acceptable temperature for use as the temperature that the zirconium-alloy has been shown to be susceptible to breakaway oxidation. Additional information regarding this temperature is provided below.

Based on data reported by Leistikow and Schanz (Ref. 21), zirconium alloys have been shown to be susceptible to the breakaway oxidation phenomenon at temperatures as low as 650°C (1,202°F). At 650°C (1,202°F), it took more than 4 hours (beyond LOCA-relevant times for conventional reactors) for Zircaloy-4 to accumulate 200 wppm hydrogen, while at 800°C (1,472°F), the time to accumulate 200 wppm hydrogen was only 1 hour (within LOCA-relevant times). Thus, time spent in steam at $\leq 650^{\circ}\text{C}$ (1,202°F) was benign with regard to breakaway oxidation and hydrogen accumulation because of the very low oxidation rate.

Recently, work documented in H. K. Yueh, et. al., "Changes in Cladding Properties under LOCA Conditions" (Ref. 22), included pre-oxidation of seven samples at 800°C (1,472°F) for 18,000 seconds. Although the pre-oxidation exercise was not designed to demonstrate resistance to breakaway oxidation, the observation that none of these samples showed any indication of breakaway oxidation for such a long time period suggests that the time spent in steam at less than or equal to 800°C (1,472°F) was benign with regard to breakaway oxidation.

C.2.C Evaluating ECCS Performance for Breakaway Oxidation

Based on the approved ECCS evaluation models and methods, the applicant should identify the limiting combination of break size, break location, and initial conditions and assumptions that maximize the total accumulated time that the cladding is predicted to remain above 800°C (1,472°F). The applicant should demonstrate that this time interval remains below the established alloy-specific breakaway oxidation analytical limit.

The applicant may credit operator actions to limit the duration at elevated temperatures, provided these actions are consistent with existing procedures and the timing of such actions is validated by operator training on the plant simulator or via a job performance measure.

C.3 An Analytical Limit on Combustible Gas Generation

The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam should not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

C.4 Considering Oxygen Diffusion from the Cladding Inside Surface in ECCS Evaluation Models

ECCS evaluation models must consider oxygen diffusion from the cladding inside surfaces if an oxygen source is present on the inside surface of the zirconium-alloy cladding at the onset of the LOCA. Two scenarios where an oxygen source should be assumed to be present on the inside surface of the

cladding at the onset of the LOCA are when cladding rupture is predicted to occur and when a fuel-cladding bond has formed. Guidance is provided below for how to consider these two scenarios in ECCS evaluation models.

C.4.A Application of Post-Quench Ductility Limits in the Rupture Region

In regions of the fuel rod where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling and rupture, it is acceptable to define the cladding thickness as the cladding cross-sectional area divided by the cladding circumference, taken at a horizontal plane at the elevation of the rupture. It is acceptable to calculate two-sided oxidation using the CP correlation and apply the analytical limit in figure 2 (or an alternative specified and acceptable analytical limit). Additional discussion is provided below.

During a postulated LOCA, fuel rods may be predicted to balloon and rupture because of elevated cladding temperature and differential pressure (between rod internal pressure and system pressure, which is decreasing because of a break in pressure boundary). The regions of the fuel rod near the ballooned and ruptured location will thus be exposed to oxidation from the inside surface of the cladding. Combined with oxygen diffusion from the cladding outside diameter, oxygen diffusion from the cladding ID would further limit integral time at temperature to reach the analytical limit in figure 2. In addition, local regions above and below the rupture opening will absorb significant hydrogen because of the steam oxidation on the ID, which may result in locally brittle regions above and below the rupture. Finally, the balloon region will experience wall thinning, which affects the calculation of ECR because the value is taken to be a percentage of the pre-oxidation cladding thickness.

The LOCA acceptance criteria that limit peak oxidation temperature and maximum oxidation level versus hydrogen content are based on retention of ductility. As discussed above, ductility will not be retained everywhere in the balloon region.

To investigate the mechanical behavior of ruptured fuel rods, the NRC conducted integral LOCA testing, designed to produce ballooning and rupture, on as-fabricated and hydrogen-charged cladding specimens and high-burnup fuel rod segments exposed to high temperature steam oxidation followed by quench (Ref. 14). The integral LOCA testing confirmed that continued exposure to high-temperature steam environments weakens the already-flawed region of the fuel rod surrounding the cladding rupture. Hence, limitations on integral time at temperature are necessary to preserve an acceptable amount of mechanical strength and fracture toughness. In addition, this research demonstrated that the degradation in strength and fracture toughness with prolonged exposure to steam oxidation was enhanced with preexisting cladding hydrogen content.

C.4.B Accounting for Oxygen Ingress on the Cladding Inside Surface due to the Fuel-Cladding Bond Layer

An acceptable approach to account for oxygen ingress on the cladding inside surface due to the fuel-cladding bond layer is to use twice the oxidation as on the exterior of the cladding for un-ruptured locations for fuel rods with a local exposure beyond 30 GWd/MTU. Accounting for oxygen ingress on the cladding ID because of the fuel-cladding bond layer for fuel rods with a local exposure beyond 30 GWd/MTU is considered conservative.

An applicant may propose a threshold for the onset of this inside surface oxidation source other than 30 GWd/MTU and provide it as part of the documentation supporting its request for NRC approval of the new or existing fuel design (e.g., license amendment request or vendor topical report). A threshold

other than 30 GWd/MTU could be supported by metallographic images of bonding layers as a function of burnup. The NRC notes that there would be no metal-water-reaction heat associated with this process on the ID, in contrast to the situation in a rupture node.

D. IMPLEMENTATION

Licensees generally are not required to comply with the guidance in this regulatory guide. If the NRC proposes to use this regulatory guide in an action that would constitute backfitting, as that term is defined in 10 CFR 50.109, “Backfitting,” and as described in NRC Management Directive 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests” (Ref. 23); affect the issue finality of an approval issued under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants”; or constitute forward fitting, as that term is defined in Management Directive 8.4, then the NRC staff will apply the applicable policy in Management Directive 8.4 to justify the action. If a licensee believes that the NRC is using this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may inform the NRC staff in accordance with Management Directive 8.4.

GLOSSARY

breakaway oxidation	The fuel-cladding oxidation phenomenon in which the weight gain rate deviates from normal kinetics. This change occurs with a rapid increase of hydrogen pickup during prolonged exposure to a high-temperature steam environment, which promotes loss of cladding ductility.
loss-of-coolant accident (LOCA)	A postulated accident that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.
offset strain	The value determined from a load-displacement curve by the following procedure: (1) linearize the initial loading curve, (2) use the slope of the initial loading curve to mathematically unload the sample at the peak load before a significant load drop (about 30–50 percent) indicating a through-wall crack along the length of the sample, and (3) determine the offset displacement (distance along the displacement axis between loading and unloading lines). This offset displacement is normalized to the outer diameter of the preoxidized cladding to determine a relative plastic strain.
permanent strain	The difference between the posttest outer diameter (after the sample is unloaded) and the pretest outer diameter of a cladding ring, normalized to the initial diameter of the cladding ring.

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⁴ Publicly available NRC published documents are available electronically through the NRC Library on the NRC’s public website at <http://www.nrc.gov/reading-rm/doc-collections/> and through the NRC’s Agencywide Documents Access and Management System (ADAMS) at <http://www.nrc.gov/reading-rm/adams.html>. For problems with ADAMS, contact the Public Document Room staff at 301-415-4737 or 800-397-4209, or e-mail pdr.resource@nrc.gov. The NRC Public Document Room (PDR), where you may also examine and order copies of publicly available documents, is open by appointment. To make an appointment to visit the PDR, please send an e-mail to pdr.resource@nrc.gov or call 800-397-4209 or 301-415-4737, between 8 a.m. and 4 p.m. eastern time (ET), Monday through Friday, except Federal holidays.

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⁵ Copies of International Atomic Energy Agency (IAEA) documents may be obtained through its website: www.iaea.org/ or by writing the International Atomic Energy Agency, P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria.

⁶ Copies of American Society for Testing and Materials (ASTM) standards may be purchased from ASTM, 100 Barr Harbor Drive, P.O. Box C700, West Conshohocken, Pennsylvania 19428-2959; telephone (610) 832-9585. Purchase information is available through the ASTM Web site at <http://www.astm.org>.

⁷ A copy of this document is available for review by the public at the NRC's Technical Library, by appointment, which is located at Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852; telephone: 301-415-7000; e-mail: Library.Resource@nrc.gov.

⁸ *Nuclear Engineering and Design* is a publication of Elsevier Inc., 230 Park Avenue, 7th floor, New York, NY 10169, telephone: 212-309-8100. Copies of Elsevier books and journals can be purchased at its website: <https://www.elsevier.com/books-and-journals>.

APPENDIX A

FUEL ROD CLADDING HYDROGEN UPTAKE MODELS

A-1 Scope and Purpose

The purpose of this appendix is to provide acceptable fuel rod cladding hydrogen uptake models for the current commercial zirconium alloys to aid in the implementation of the hydrogen-dependent emergency core cooling system performance requirements. Figure 2 of this regulatory guide (RG) provides acceptable analytical limits on peak cladding temperature and integral time-at-temperature (expressed as equivalent cladding reacted calculated using the Cathcart-Pawell correlation (CP-ECR)) as a function of pre-transient cladding hydrogen content. To support implementing these new requirements, steady-state cladding waterside corrosion and hydrogen pickup models are needed to translate these analytical limits to fuel burnup.

These models also are acceptable for implementing other hydrogen-dependent fuel performance requirements (e.g., reactivity initiated accident (RIA), pellet-to-cladding mechanical interaction (PCMI), cladding failure thresholds).

A-2 Discussion

Pacific Northwest National Laboratory (PNNL) compiled information on publicly available cladding hydrogen measurements and revised the hydrogen uptake models in the FRAPCON fuel rod performance code. In addition, PNNL quantified the standard deviation in these model predictions relative to the publicly available information compiled by PNNL. Table 1 of Reference A-1 (reproduced as table A-1 below) summarizes the recommended changes to the FRAPCON-3.4 hydrogen uptake models for Zircaloy-2, Zircaloy-4, M5®, and ZIRLO® based upon the expanded hydrogen database. The revised corrosion and hydrogen uptake models are documented in NUREG/CR-7022, Volume 1, Revision 1 (Ref. A-2).

Table A-1. Hydrogen models in FRAPCON-3.4 and new model

Alloy	FRAPCON-3.4		New Model	
	Model	Std. dev.	Model	Std. dev.
Boiling Water Reactor				
Zircaloy-2 pre-1998	Eq. 2	10 ppm ¹ NA ²	Eq. 2	10 ppm ¹ 54 ppm ²
Zircaloy-2 post-1998	Eq. 3	11 ppm ¹ 61 ppm ²	Eq. 3	13 ppm ¹ 60 ppm ²
Pressurized Water Reactor				
Zircaloy-4	15% ³	40 ppm	15.3% ³	94 ppm
ZIRLO®	12.5% ³	162 ppm	17.3% ³	110 ppm
M5®	10% ³	20 ppm	10% ³	23 ppm

¹ standard deviation below 50 GWd/MTU

² standard deviation above 50 GWd/MTU

³ pickup fraction

A-2.1 Boiling Water Reactor (BWR) Zircaloy-2

For BWR conditions, a constant hydrogen pickup fraction does not fit the observed cladding hydrogen data. As a result, FRAPCON-3.5 (Ref. A-2) uses a burnup-dependent hydrogen concentration model. In addition, the recommended Zircaloy-2 model is divided between modern alloys (with tighter control of composition and second phase precipitation particle size) and legacy alloys. The best-estimate hydrogen uptake models are listed below:

Legacy alloys:

$$H = \begin{cases} 47.8e^{-1.3/(1+BU)} + 0.316BU, & BU < 50 \text{ GWd/MTU} \\ 28.9 + e^{0.117(BU-20)}, & BU > 50 \text{ GWd/MTU} \end{cases}$$

Modern alloys:

$$H = 22.8 + e^{0.117(BU-20)}$$

Where:

H = total hydrogen, wppm

BU = local axial burnup, GWd/MTU

Given the allowable range in composition within the Zircaloy-2 ASTM specification (ASTM B351/B351M, "Standard Specification for Hot-Rolled and Cold-Finished Zirconium and Zirconium Alloy Bars, Rod, and Wire for Nuclear Application," Ref. A-3) and the degree of flexibility and variability in manufacturing procedures between the fuel vendors, the NRC staff has elected to adopt the more conservative legacy hydrogen uptake model. For this model, Table 1 of Reference A-1 provides a standard deviation on the model prediction of 10 wppm below 50 GWd/MTU and 54 wppm above 50 GWd/MTU. To account for variability and uncertainty, the staff decided to use a +2-sigma uncertainty band on the model prediction. Figure A-1 illustrates the best-estimate and +2-sigma model predictions along with the entire database. Examination of this figure reveals a discontinuity at 50 GWd/MTU where the larger standard deviation is first applied. In addition, application of the same standard deviation to even higher burnup suggests that the relative scatter in hydrogen content is becoming smaller. This is not likely the case. As a result, the staff developed a 1.40 multiplier of the model prediction that is approximately equal to 2-sigma at the lower burnup. This new model is shown in Figure A-2. The application of this multiplier removes the discontinuity and ensures that the model reflects a larger uncertainty at higher concentrations of hydrogen.

An acceptable BWR Zircaloy-2 hydrogen uptake model is provided below.

$$H = \begin{cases} 1.4(47.8e^{-1.3/(1+BU)} + 0.316BU), & BU < 50 \text{ GWd/MTU} \\ 1.4(28.9 + e^{0.117(BU-20)}), & BU > 50 \text{ GWd/MTU} \end{cases}$$
$$H = (47.8 \exp[-1.3/(1+BU)] + 0.316BU) * 1.40 \quad BU < 50 \text{ GWd/MTU}$$
$$H = (28.9 + \exp[0.117(BU-20)]) * 1.40 \quad BU > 50 \text{ GWd/MTU}$$

Where:

H = total hydrogen, wppm

BU = local axial burnup, GWd/MTU

References A-4 and A-5 describe an independent Zircaloy-2 hydrogen uptake model along with hydrogen data from various sources. A comparison of best-estimate predictions with the model above and the Heck model is provided in Table A-2 below. Examination of this table reveals reasonable agreement up to 60 GWd/MTU. At higher exposures the FRAPCON 3.4 model predicts higher hydrogen contents relative to the Heck model. Given the lack of data in this region, the more conservative FRAPCON model is preferable. In addition, examination of the data scatter shown in References A-4 and A-5 supports the 1.40 multiplier on the model prediction.

A-2.2 Pressurized Water Reactor (PWR) Zirconium Alloys

Corrosion rates and the amount of corrosion at fuel discharge vary widely across the PWR fleet because of alloy composition, operating conditions, and residence time (i.e., effective full power days). Fuel vendors have approved fuel performance analytical tools along with corrosion models. In general, these corrosion models are capable of predicting a best-estimate corrosion thickness as a function of effective full power days and local operating conditions (fuel duty).

Hydrogen data collected on PWR zirconium alloy cladding do not exhibit the same breakaway hydrogen uptake at higher fluence levels as observed in the BWR Zircaloy-2 data. However, the pickup fraction does appear to be alloy specific. As a result, the applicant should propose a constant hydrogen pickup fraction for each zirconium alloy.

These hydrogen pickup fractions should be used, along with a best-estimate prediction of the peak oxide thickness using an approved fuel rod thermal-mechanical model, to estimate the cladding hydrogen content.

Table 1 of Reference A-1 (reproduced as Table A-1 above) defines the following best-estimate hydrogen pickup fractions and standard deviation relative to the hydrogen database for Zircaloy-4, ZIRLO[®], and M5[®] cladding:

Zircaloy-4	- 15.3% pickup	94 wppm standard deviation
ZIRLO [®]	- 17.3% pickup	110 wppm standard deviation
M5 [®]	- 10.0% pickup	23 wppm standard deviation

Figure 4, 6, and 8 of Reference A-1 show predicted versus measured hydrogen concentration along with a +2-sigma band for Zircaloy-4, ZIRLO[®], and M5[®] cladding, respectively. Similar to the above BWR model, the staff has decided to apply a +2-sigma uncertainty band on the model prediction to account for variability and uncertainty in the database. However, the application of a constant, additive standard deviation has negative attributes including (1) it is overly conservative when applied to low burnup, low corrosion fuel rods and (2) there is no recognition for larger scatter in highly corroded fuel rods.

With consideration of the extent and variability of the supporting database, the staff developed upper bound pickup fractions. As described in Reference A-1, the expanded Zircaloy-4 hydrogen database has more than 280 measurements. Figure A-3 shows predicted versus measured hydrogen concentration using the above 15.3 percent pickup fraction. With more than 280 data points, a 95/95 non-parametric statistical upper bound could be derived. However, given all of the variables (e.g., alloy content, operating conditions) and uncertainties, there is no guarantee that the data are actually poolable. Instead, the staff elected to iterate on pickup fraction until a reasonable upper bound prediction was obtained. Figure A-4 shows predicted versus measured hydrogen content assuming a 20 percent pickup fraction. Examination of the figure reveals that a vast majority of the data are conservatively predicted.

For ZIRLO[®] cladding, the hydrogen database is limited to 60 data points. As such, a 95/95 non-parametric statistical upper bound would need to bound 100 percent of the data and likely be overly conservative. Figure A-5 shows predicted versus measured hydrogen content using PNNL's recommended 17.3 percent pickup fraction. For the reasons stated above, the staff elected to iterate on pickup fraction until a reasonable upper bound prediction was obtained. Employing a bounding pickup fraction of 25 percent shifts the predictions, as shown in Figure A-6. Examination of this figure reveals that a reasonable majority of the data are conservatively predicted.

For Optimized ZIRLO™ cladding, applicants may use the bounding 25 percent pickup fraction along with an approved alloy-specific corrosion model.

For M5® cladding, the hydrogen database is limited to less than 20 data points. As such, a 95/95 non-parametric statistical upper bound of this database is not possible. Figure A-7 shows predicted versus measured hydrogen content using PNPL's recommended 10.0 percent pickup fraction. For the reasons stated above, the staff elected to iterate on pickup fraction until a reasonable upper bound prediction was obtained. Employing a bounding pickup fraction of 15 percent shifts the predictions, as shown in Figure A-8. Examination of this figure reveals that a reasonable majority of the data are conservatively predicted.

Based on the above discussion, the staff finds the following bounding hydrogen pickup fractions acceptable:

Zircaloy-4	- 20% hydrogen absorption
ZIRLO®	- 25% hydrogen absorption
Optimized ZIRLO™	- 25% hydrogen absorption
M5®	- 15% hydrogen absorption

These hydrogen pickup fractions should be used, along with a best-estimate prediction of the peak oxide thickness using an approved fuel rod thermal-mechanical model, to estimate the cladding hydrogen content.

Applicability

The hydrogen models are applicable to currently approved commercial alloys up to their respective limits on fuel rod burnup, corrosion, and residence time.

The hydrogen models are not applicable to fuel rods that experience oxide spallation.

Table A-2. Comparison of Hydrogen Uptake Models

Local Exposure (GWd/MTU)	Best-Estimate Hydrogen Prediction	
	FRAPCON-3.4 Legacy	Heck (2008)
0	15	18
10	46	42
20	51	48
30	55	48
40	59	55
50	62	80
60	137	150
70	376	260

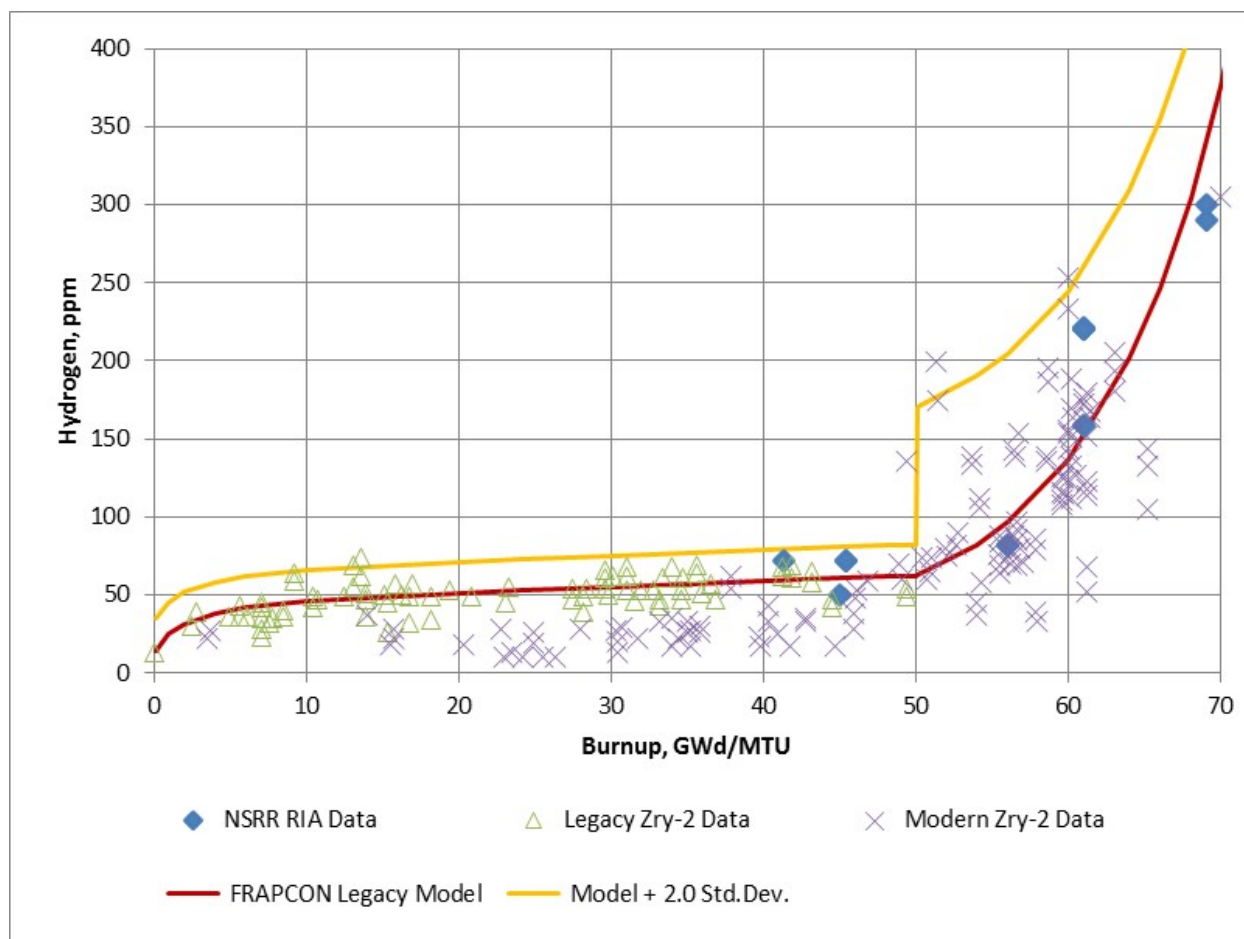


Figure A-1. Zircaloy-2 Hydrogen Model, +2-Sigma Prediction Versus Data

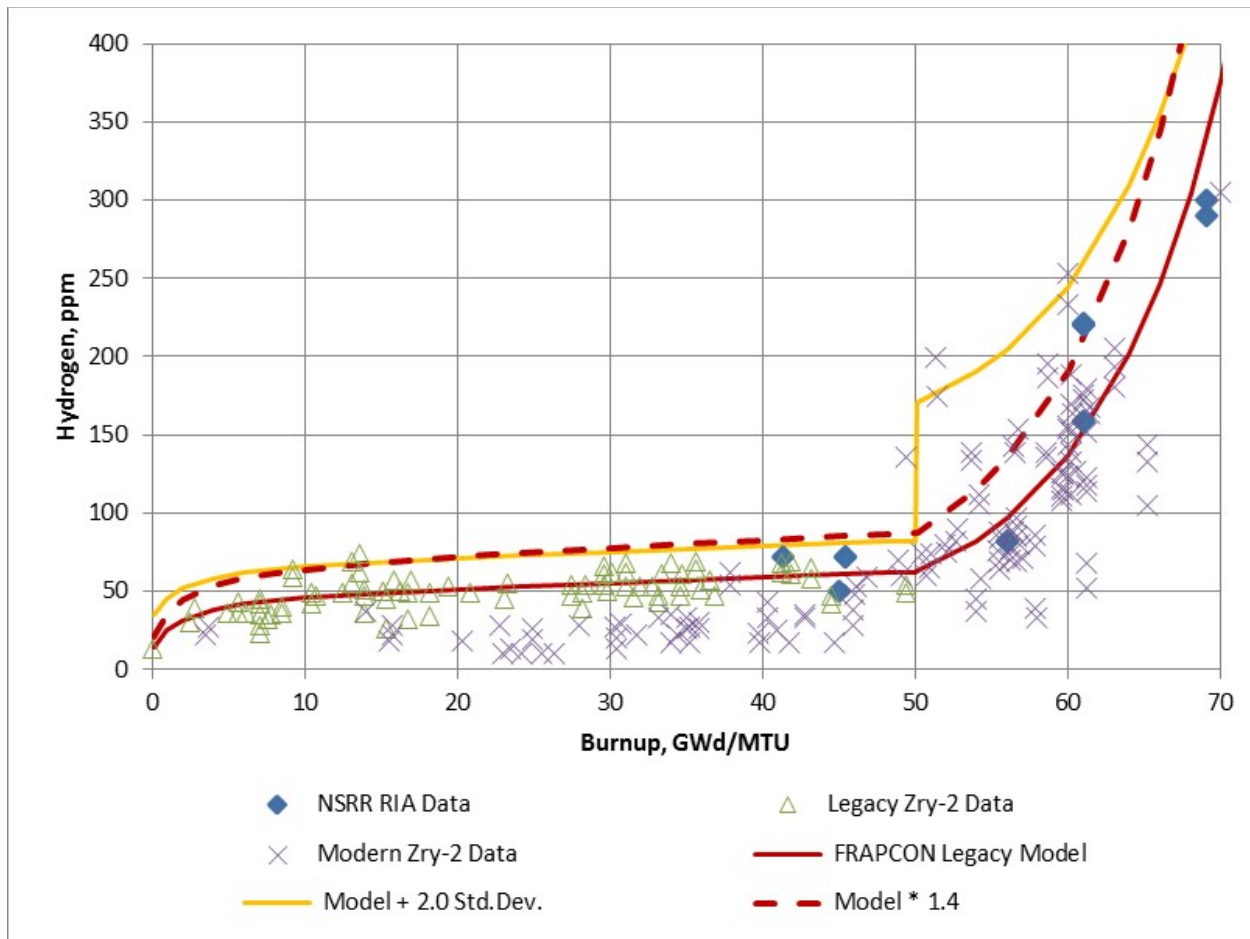


Figure A-2. Zircaloy-2 Hydrogen Model, 1.40 Multiplier Prediction Versus Data

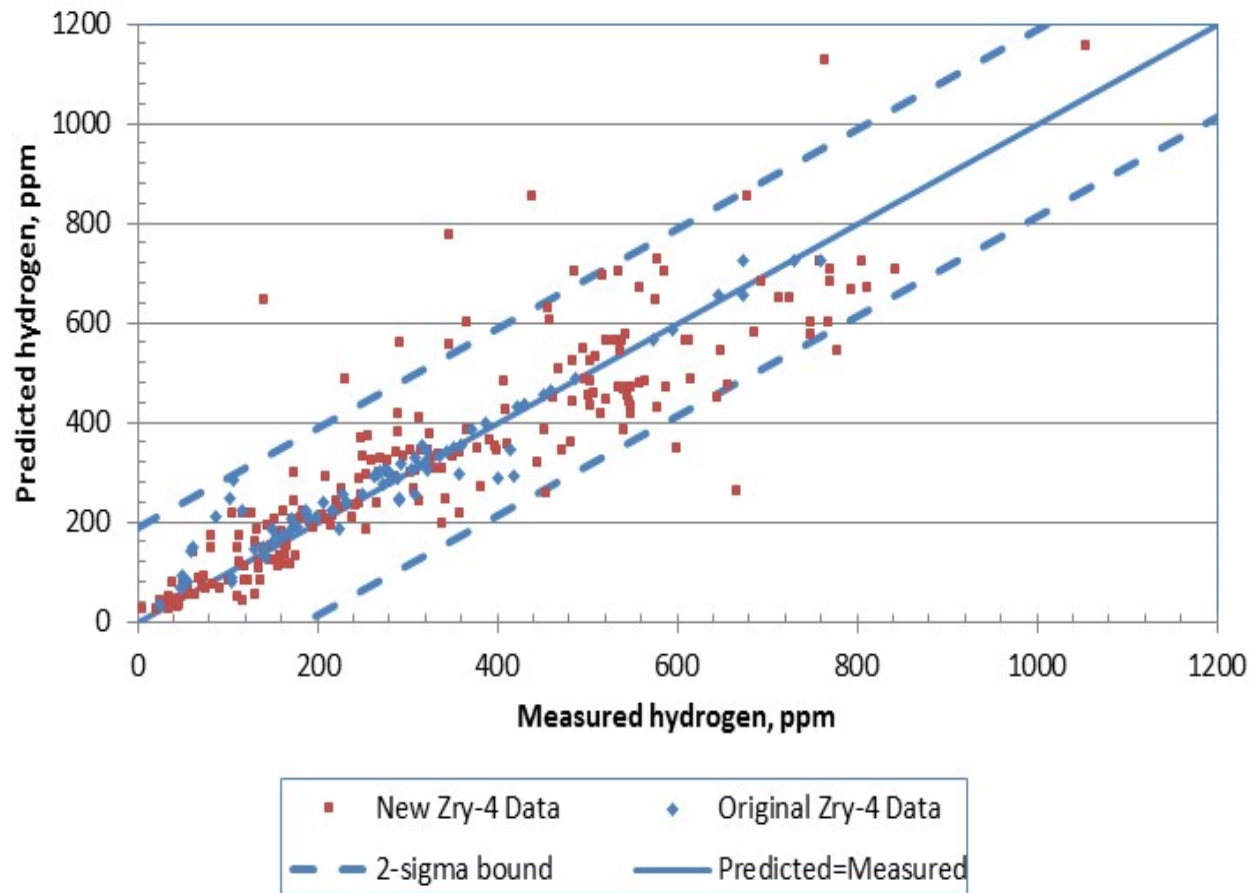


Figure A-3. Zircaloy-4 Predicted Versus Measured Hydrogen Content, 15.3% Pickup (Figure 4 of Reference A-1)

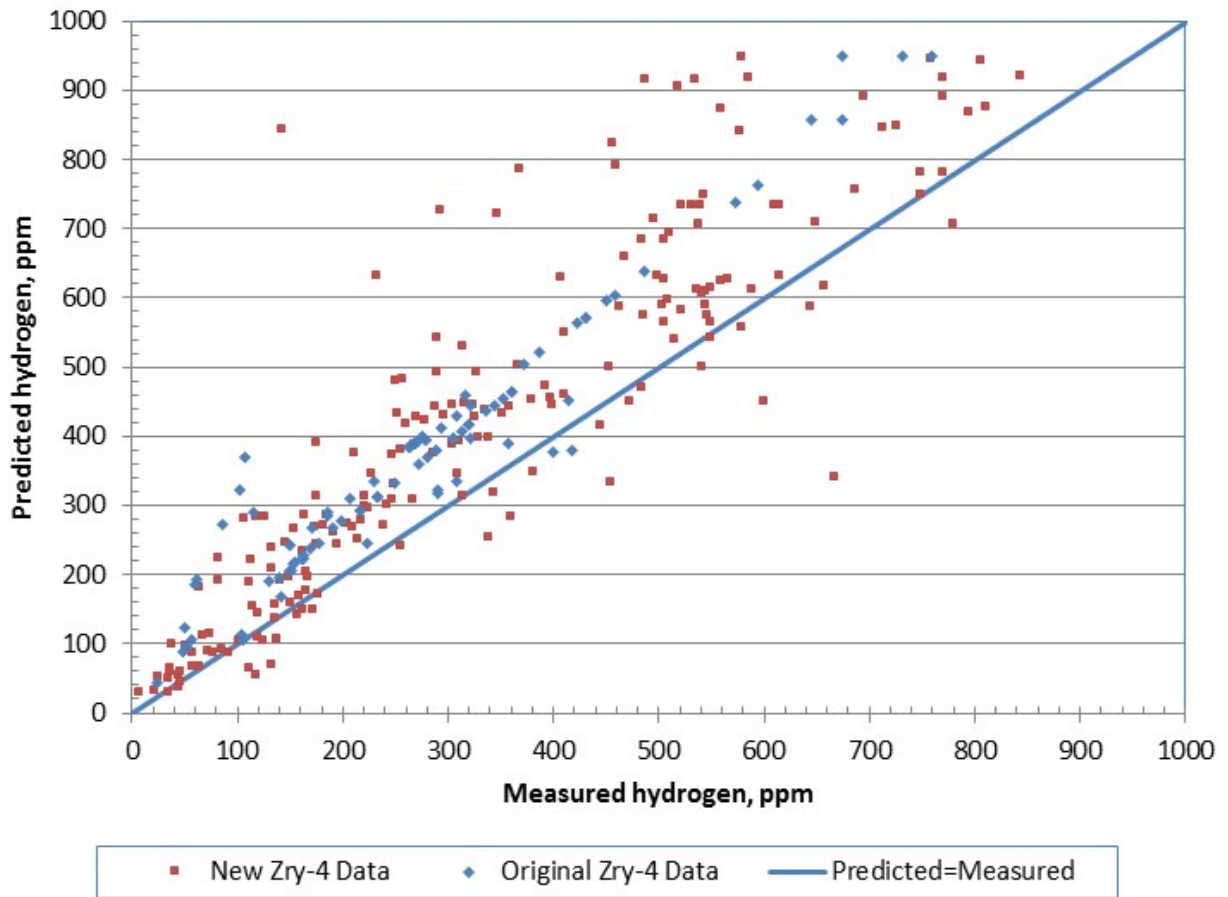


Figure A-4. Zircaloy-4 Predicted Versus Measured Hydrogen Content, 20.0% Pickup

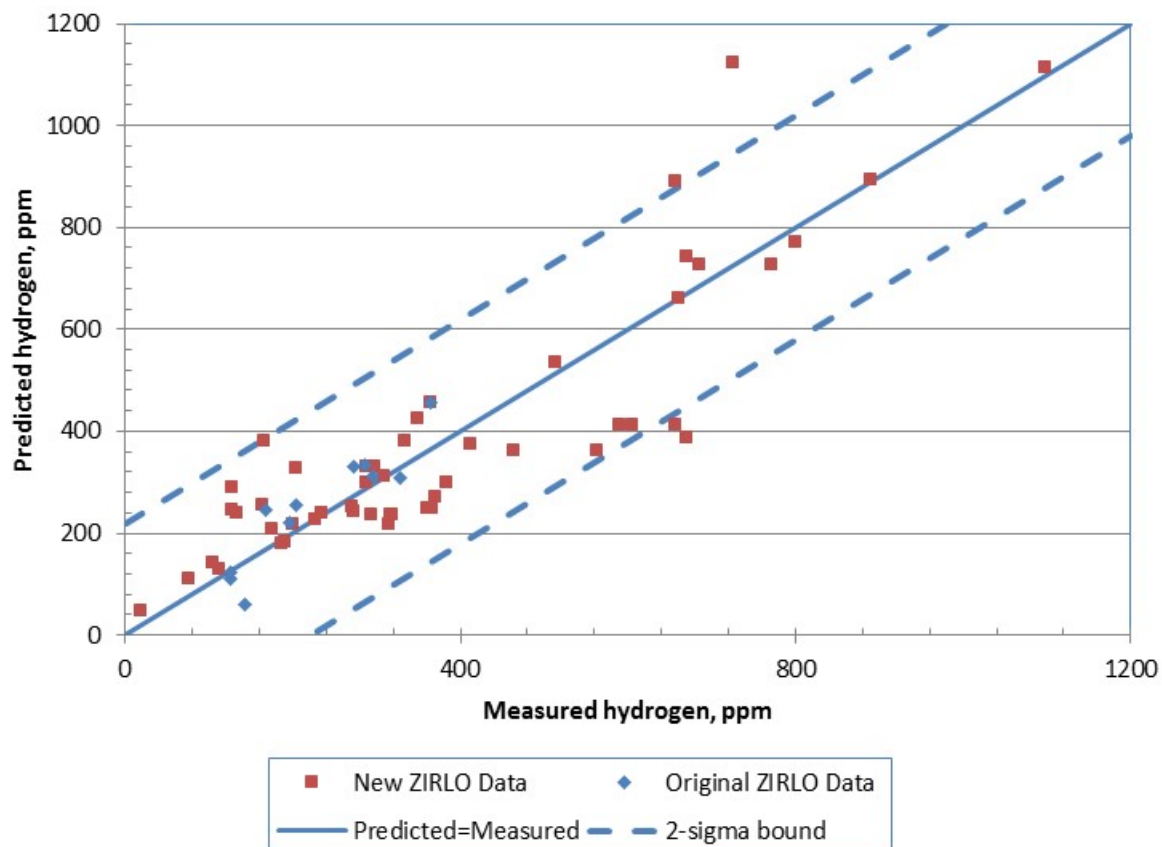


Figure. A-5: ZIRLO[®] Predicted Versus Measured Hydrogen Content, 17.3% Pickup (Figure 6 of Reference A-1)

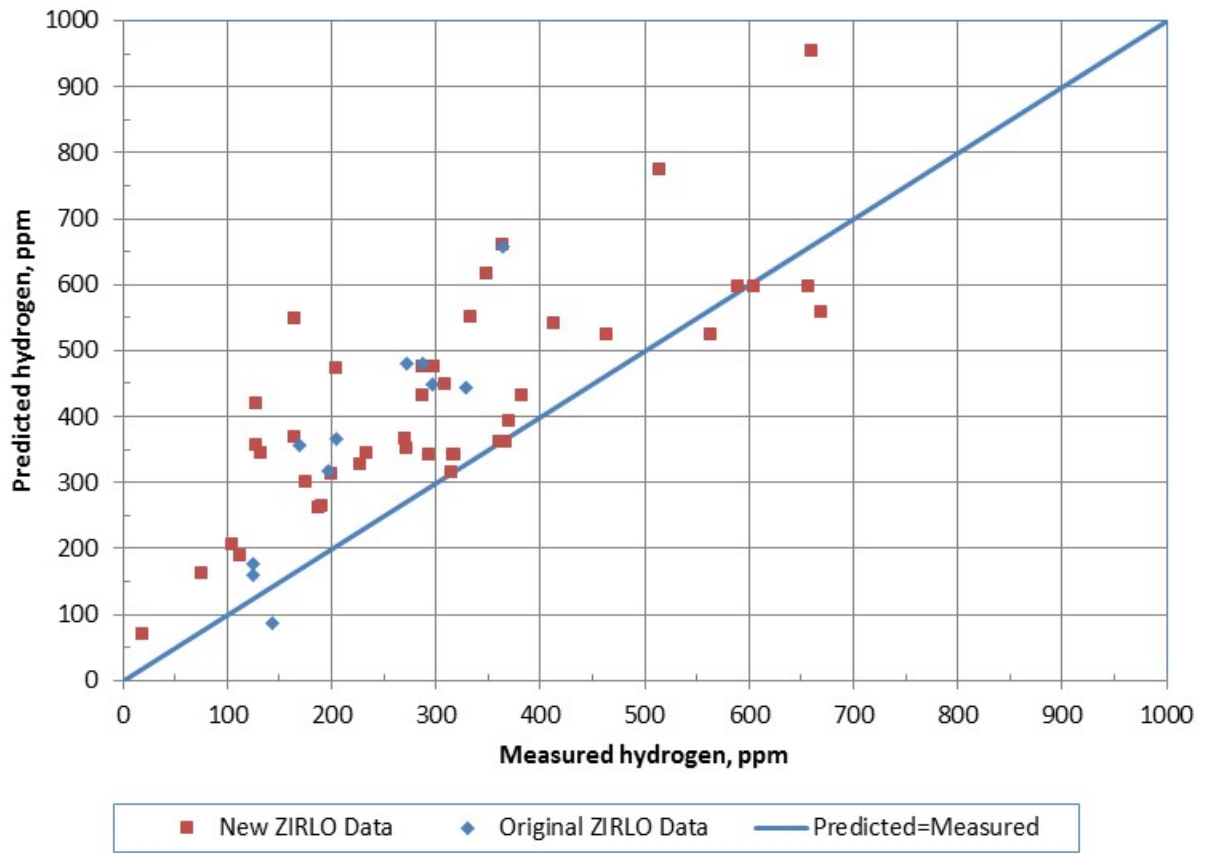


Figure A-6. ZIRLO® Predicted Versus Measured Hydrogen Content, 25.0% Pickup

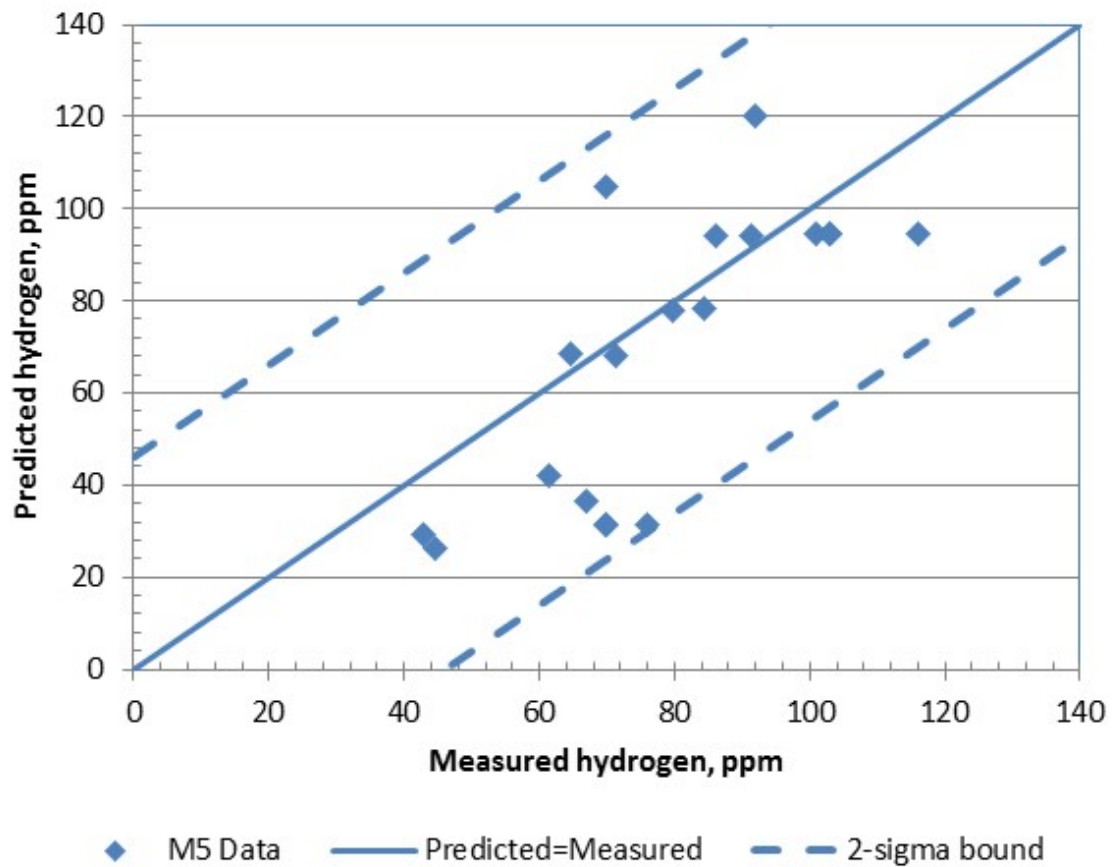


Figure A-7. M5[®] Predicted Versus Measured Hydrogen Content, 10% Pickup (Figure 8 of Reference A-1)

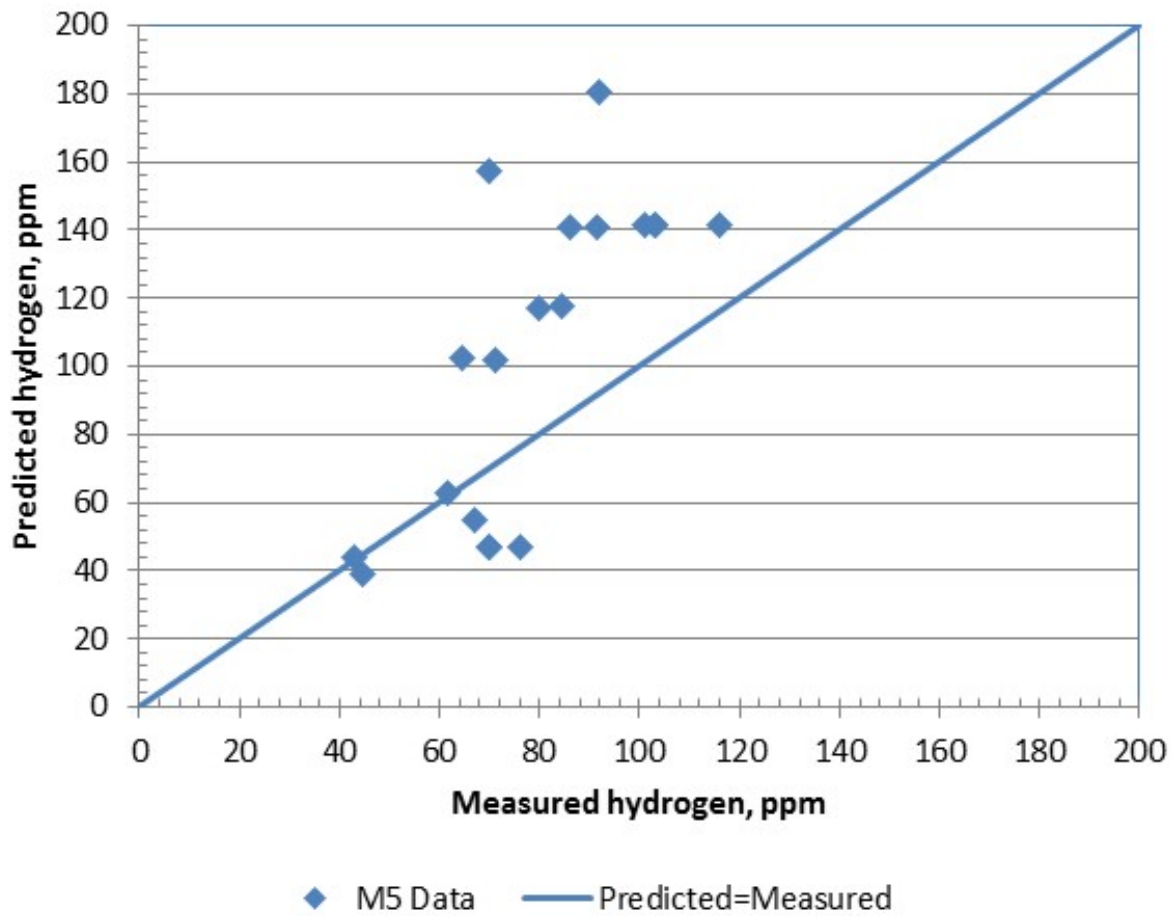


Figure A-8. M5® Predicted Versus Measured Hydrogen Content, 15% Pickup

APPENDIX A REFERENCES

- A-1. Geelhood, K., and C. Beyer, “Hydrogen Pickup Models for Zircaloy-2, Zircaloy-4, M5TM and ZIRLOTM,” 2011 Water Reactor Fuel Performance Meeting, Chengdu, China, September 11-14, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12093A469).
- A-2. NRC, NUREG/CR-7022, “FRAPCON-3.5: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behaviour of Oxide Fuel Rods for High Burnup,” Volume 1, Revision 1, Washington, DC, October 2014.
- A-3. American Society for Testing and Materials (ASTM) B351/B351M, “Standard Specification for Hot-Rolled and Cold-Finished Zirconium and Zirconium Alloy Bars, Rod, and Wire for Nuclear Application.” ASTM International, West Conshohocken, PA, 2005. ^{1,2}
- A-4. Rudling, P., “Zr Alloy Corrosion and Hydrogen Pickup,” ANT International, December 2013 (ADAMS Accession No. ML15253A227).
- A-5. Heck, C., “BWR Control Rod Drop Accident: Methodology, Application and Regulatory Compliance,” Public Workshop on Reactivity Insertion Accident (RIA) Regulatory Criteria And Guidance, Rockville, MD, September 25, 2008.

¹ Copies of American Society for Testing and Materials (ASTM) standards may be purchased from ASTM, 100 Barr Harbor Drive, P.O. Box C700, West Conshohocken, Pennsylvania 19428-2959; telephone (610) 832-9585. Purchase information is available through the ASTM Web site at <http://www.astm.org>.

² A copy of this document is available for review by the public at the NRC’s Technical Library, by appointment, which is located at Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852; telephone: 301-415-7000; e-mail: Library.Resource@nrc.gov.