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Technical Lead
M. Bales

REGULATORY GUIDE

Public availability of this draft document is intended to inform stakeholders of the current status of the NRC staff's preliminary draft final rule package and associated documents for § 50.46c of Title 10 of the Code of Federal Regulations (10 CFR). These preliminary draft documents are in support of an October 22, 2015, Category 3 public meeting, and a November 2, 2015, Advisory Committee on Reactor Safeguards (ACRS) subcommittee meeting.

This draft document has not been subject to all levels of NRC management review. Accordingly, it ***is*** *incomplete and may be error in one or more respects. The document may be subject to further revision before the staff provides the final draft rule language package to the Commission (currently scheduled to be provided to the Commission in February 2016.*

REGULATORY GUIDE 1.224

(Draft was issued as DG-1263, dated March 2014)

ESTABLISHING ANALYTICAL LIMITS FOR ZIRCONIUM-ALLOY CLADDING MATERIAL

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes an approach to establish analytical limits for post-quench ductility and breakaway oxidation for zirconium-alloy cladding material that the U.S. Nuclear Regulatory Commission (NRC) accepts in the implementation of the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46c, "Emergency Core Cooling System (ECCS) Performance during Loss of Coolant Accidents (LOCA)" (Ref. 1), subsection (g), "Fuel system designs: uranium oxide or mixed uranium-plutonium oxide pellets within cylindrical zirconium-alloy cladding." This RG also provides guidance on how to consider oxygen diffusion from inside surfaces in ECCS evaluation to meet the requirements of Section (g)(2)(i) of 10 CFR 50.46c.

Applicable Rules and Regulations

- Regulations contained in 10 CFR 50.46c require that analytical limits on peak cladding temperature and integral time at temperature be established that correspond to the measured

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ductile-to-brittle transition for the zirconium-alloy cladding material based on an NRC-approved experimental technique.

Related Guidance

- Regulatory Guide 1.222, "Measuring Breakaway Oxidation Behavior" (Ref. 2), describes a method that the NRC staff considers acceptable to measure and periodically confirm the breakaway oxidation behavior of a zirconium-alloy cladding material.

- Regulatory Guide 1.223, "Determining Post-Quench Ductility" (Ref. 3), describes a method that the NRC staff considers acceptable for measuring the ductile-to-brittle transition for a zirconium-alloy cladding.

Purpose of Regulatory Guides

The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG contains information collection requirements covered by 10 CFR Part 50 that the Office of Management and Budget (OMB) approved under OMB control number 3150-0011. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays a currently valid OMB control number.

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B. DISCUSSION

Reason for Issuance

This RG provides guidance to implement the performance-based rule in 10 CFR 50.46c. The rule requires that analytical limits be established for zirconium-alloy cladding material. This RG describes an acceptable approach to establish analytical limits on post-quench ductility and breakaway oxidation behavior for the zirconium-alloy cladding material. This RG also provides guidance regarding the development of hydrogen uptake models, which are needed to establish hydrogen-dependent post-quench ductility analytical limits. Finally, this RG provides guidance on how to consider oxygen diffusion from inside surfaces in an ECCS evaluation to meet the requirements of 10 CFR 50.46c(g)(2)(i).

Background

In 1996, the NRC initiated a fuel-cladding research program intended to investigate the behavior of high-exposure fuel cladding under accident conditions. This program included an extensive LOCA research and testing program at Argonne National Laboratory (ANL) (See NUREG/CR-6967, "Cladding Embrittlement during Postulated Loss-of-Coolant Accidents" (Ref. 4)), 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. 5)), and the Halden Reactor Project (see IFE/KR/E 2008/004, "LOCA Testing of High Burnup PWR Fuel in the HBWR. Additional PIE on the Cladding of the Segment 650.5" (Ref. 6)), to develop the body of technical information needed to evaluate LOCA regulations for high-exposure fuel. The research findings have been summarized in Research Information Letter (RIL) 0801, "Technical Basis for Revision of Embrittlement Criteria in 10 CFR 50.46" (Ref. 7).

The research results revealed that hydrogen, which is absorbed into the cladding during the burnup-related corrosion process under normal operation, 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 as hydrogen pickup increases. 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 RIL-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 OD, oxygen ingress from the cladding ID would further limit integral time at temperature to nil ductility.

Existing Embrittlement Database

The majority of the cladding embrittlement experimental results from the NRC's LOCA research program are summarized in NUREG/CR-6967. Since the publication of NUREG/CR-6967 in 2008, additional testing was conducted, focusing on cladding materials with hydrogen contents in the 200- to 350-weight parts per million (wppm) range (Refs. 8–9, 11). Additional oxidation and post-quench

ductility (PQD) tests were conducted with cladding samples sectioned from high-burnup ZIRLO®[1] defueled segments characterized by a 25–30 micrometers corrosion-layer thickness and 300–340 wppm of hydrogen in the cladding metal before oxidation (Ref. 8). Also, the ductility data for an oxidation sample with about 600-wppm hydrogen were reassessed (Ref. 9). In addition, since the publication of NUREG-6967, oxidation and PQD tests were conducted with pre-hydrided cladding samples containing 200–300 wppm of hydrogen (Ref. 9).

The tests that were conducted after the publication of NUREG/CR-6967 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.

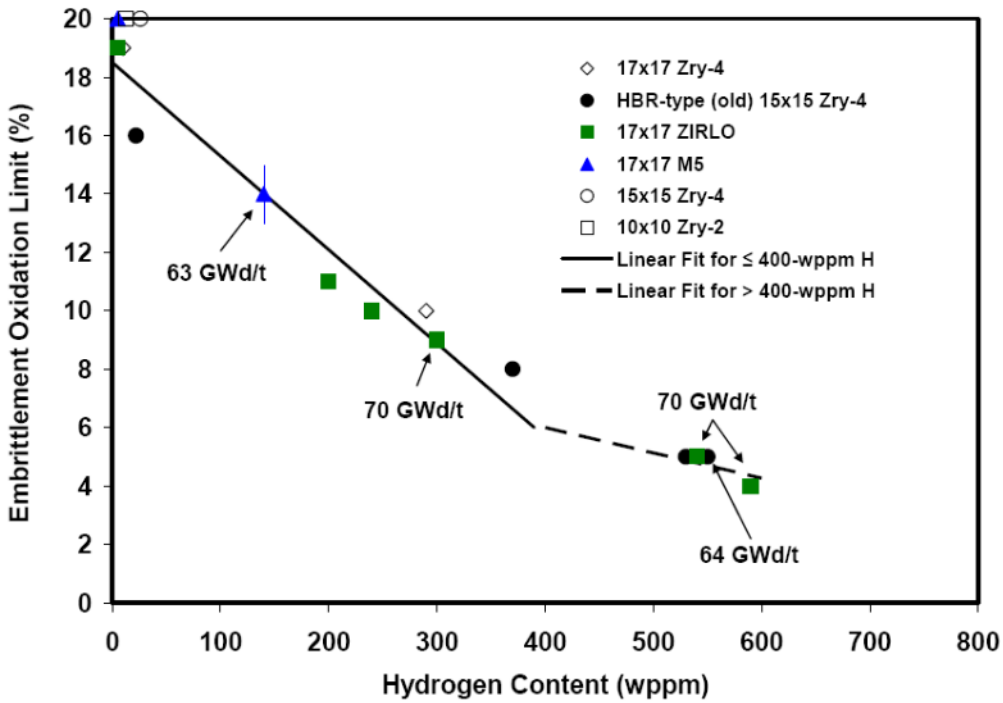

Figure 1. Ductile-to-brittle transition oxidation level (CP-ECR) as a function of pretest hydrogen content in cladding metal for as-fabricated, prehydrided, and high-burnup cladding materials. Samples were oxidized at ≤1,200 °C ±10 °C and quenched at 800 °C. For high-burnup cladding with about 550-wppm hydrogen, embrittlement occurred during the heating ramp at 1,160–1,180 °C peak oxidation temperatures (Ref. 8).

Experimental Techniques

The regulations in 10 CFR 50.46c require that post-quench ductility analytical limits be established that correspond to the measured ductile-to-brittle transition for the zirconium-alloy cladding material based on an NRC-approved experimental technique. One NRC-approved experimental technique for measuring the ductile-to-brittle transition for the zirconium-alloy cladding material is provided in RG 1.223, but other methods may be submitted for staff review and approval. The regulations in 10 CFR 50.46c, also requires that breakaway oxidation behavior is measured for zirconium-alloy cladding material based on an NRC-approved experimental technique. One NRC-approved experimental technique

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

for measuring breakaway oxidation behavior for zirconium-alloy cladding material is provided in RG 1.222, but other methods may be submitted for staff 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 RG 1.222 and RG 1.223, or a different NRC-approved experimental technique.

Harmonization with International Standards

The International Atomic Energy Agency (IAEA) has established a series of safety guides and standards constituting a high level of safety for protecting people and the environment. IAEA safety guides present international good practices and increasingly reflects best practices to help users striving to achieve high levels of safety. Pertinent to this RG, IAEA Safety Guide NS-G-1.9, "Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants" (Ref. 11), issued September 2004, provides recommendations and guidance to regulatory bodies, nuclear power plant designers and licensees on the design of the Reactor Coolant System and associated systems, including the ECCS, to maintain the integrity of the fuel cladding. This RG describes testing related to the development of analytical limits for ECCS performance evaluation and is consistent with the basics safety principles provided in IAEA Safety Guide NS-G-1.9.

C. STAFF REGULATORY GUIDANCE

This RG describes an approach to establish analytical limits for post-quench ductility (Section C.1) and breakaway oxidation (Section C.2) for zirconium-alloy cladding material that the NRC considers acceptable to implement the requirements 10 CFR 50.46c(g)(1)(ii) and (g)(1)(iii). In addition, Section C.3 of this RG provides guidance to implement the requirements of paragraph (g)(2)(i), which states that ECCS evaluation models must consider 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.

C.1 Analytical Limits for Post Quench Ductility

The regulations in 10 CFR 50.46c require that analytical limits on peak cladding temperature and integral time at temperature be established that 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 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 degrees Celsius (C), or 2,200 degrees Fahrenheit (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®[2] and Optimized ZIRLO™[3]

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 ZIRLO™ was not tested as part of NRC's LOCA research program. Westinghouse submitted a comment on DG-1263 requesting that Optimized ZIRLO™ 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 ZIRLO™. The basis of the staff's finding is documented in Reference 19.

[2] M5 is a registered trademark of AREVA.

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

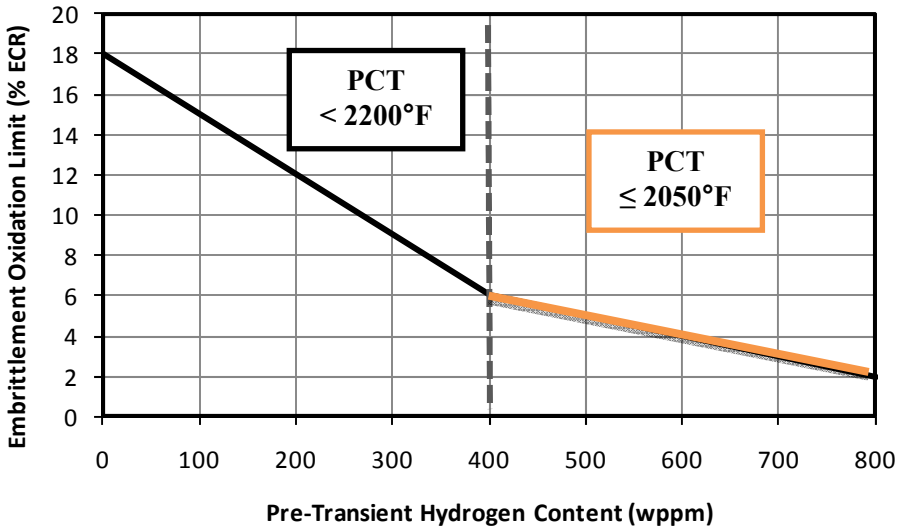

Figure 2. Acceptable analytical limits for peak cladding temperature and integral time at temperature (as calculated in local oxidation calculations using the CP correlation)

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 Cathcart-Pawel (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 degrees C (2,200 degrees 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 degrees C (2,200 degrees F) for materials with less than 400-wppm cladding hydrogen content and 1,121 degrees C (2,050 degrees 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 degrees C (2,200 degrees F) and quench temperature of 800 degrees C (1,472 degrees F). In the NRC test program, experiments were conducted at maximum oxidation temperatures less than or equal to 1,200 plus or minus 10 degrees C (2,192 plus or minus 18 degrees F) and quenched at 800 degrees C (1,472 degrees 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 degrees C (275 degrees 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 degrees C or 275 degrees 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 (Ref. 10). Use of Figure 2 as an acceptable analytical limit necessitates that the LOCA method employ the CP correlation to calculate local oxidation. However, it should be noted that Appendix K LOCA methods (Ref. 12) can continue to be used to calculate the metal-water reaction heat generation rate using the Baker-Just correlation. The CP set of correlations was only validated for temperatures greater than or equal to 1,000 degrees C (1,832 degrees F) and therefore the lower temperature threshold should be 1,000 degrees C (1,832 degrees F) for this calculation.

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 (DBT CP-ECR) for the cladding material is greater than or equal to the analytical limits provided in Figure 2. The DBT 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 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-6867 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 alloy 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. 13) 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 alloy 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, testing on irradiated material is required and the irradiated testing must confirm the desired ECR limit curve. If the DBT CP-ECR

determined from tests on irradiated material is less than the DBT CP-ECR determined from as-received and hydrogen pre-charged samples, 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 the NRC staff's review and approval of the new fuel design (i.e., 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 RG 1.223) 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 DBT 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 DBT CP-ECR should also be identified though 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 NRC staff's review and approval of the new or existing fuel design (i.e., 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 RG 1.223) 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.

C.1.D Methodology for Establishing Analytical Limits for Post-Quench Ductility at Peak Oxidation Temperatures Less than 1,204 degrees C (2,200 degrees 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 degrees C (2,200 degrees F). The DBT 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 DBT CP-ECR should also be identified though 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 degrees C (2,200 degrees 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 degrees C (2,192 plus or minus 18 degrees F) and quenched at 800 degrees C (1,472 degrees F). Some ECCSs may perform such that the maximum oxidation temperature is significantly below 1,204 degrees C (2,200 degrees 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 degrees C (2,200 degrees F), the applicant should provide experimental results as part of the documentation supporting the NRC staff's review and approval of the new fuel design or existing fuel design (i.e., 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 RG 1.223) 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 alloy-specific cladding hydrogen uptake model will be required if a licensee chooses to use the hydrogen-dependent embrittlement threshold provided in this regulatory guide. 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

Demonstrating that ECCS performance is such that local oxidation and peak cladding temperature are calculated below the established analytical limit is acceptable to demonstrate compliance with 10 CFR 50.46c. 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 peak cladding temperature and local oxidation (surrogate for time at temperature). Combinations of initial conditions and uncertainties will vary between 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," and best-estimate methods. Separate cases might be necessary to identify the limiting scenario for peak cladding temperature relative to local oxidation and vice versa. The applicant should demonstrate that predicted peak cladding temperature remains below the lesser of the regulatory limit of 2,200 degrees F and the maximum oxidation PQD temperature. 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 cladding hydrogen content, burnup, fuel rod power, or a combination. For example, peak cladding temperature and local oxidation calculations would 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 peak cladding temperature 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. These higher exposure bundles operate at substantially lower power and are usually placed along the core periphery. For re-inserted bundles manufactured prior to the effective date of the 50.46c rule and comprised of either currently available commercial cladding alloys (e.g., Zry-2, Zry-4) or legacy zirconium alloys no longer commercially available, the analytical limits in Figure 2 should be applied to show compliance with 50.46c.

C.2 Breakaway Oxidation

C.2.A Breakaway Oxidation Analytical Limits

The regulations in 10 CFR 50.46c(g)(1)(iii) require that an analytical time limit that has been shown to preclude breakaway oxidation using an NRC-approved experimental technique must be determined and specified for each zirconium-based cladding alloy. 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 the NRC staff's review and approval of the new or existing fuel design (i.e., 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 RG 1.222) 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 degrees C 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 degrees C.

Licensees may re-insert older, irradiated fuel bundles residing in the spent fuel pool. These higher exposure bundles operate at substantially lower power and are usually placed along the core periphery. For re-inserted bundles manufactured prior to the effective date of the 50.46c rule and comprised of currently available commercial cladding alloys (e.g., Zry-2, Zry-4), the analytical limit established for the current versions of those commercial alloys should be applied to show compliance with 50.46c. For re-inserted bundles manufactured prior to the effective date of the 50.46c rule and comprised of legacy zirconium alloys no longer commercially available, a default analytical limit of 3,500 seconds should be applied to show compliance with 50.46c.

C.2.B Temperature of Susceptibility for Breakaway Oxidation

The regulations in 10 CFR 50.46c (g)(1)(iii) require that the total time that the cladding is predicted to remain above the temperature that the zirconium-alloy cladding material has been shown to be susceptible to breakaway oxidation must be less that the analytical limit. Eight hundred degrees

Celsius (1,472 degrees 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. 14), zirconium alloys have been shown to be susceptible to the breakaway oxidation phenomenon at temperatures as low as 650 degrees C (1,202 degrees F). At 650 degrees C (1,202 degrees F), it took more than 4 hours (beyond LOCA-relevant times) for Zircaloy-4 to accumulate 200-wppm hydrogen, while at 800 degrees C (1,472 degrees F), the time to accumulate 200-wppm hydrogen was only 1 hour (within LOCA-relevant times). Thus, time spent in steam at less than or equal to 650 degrees C (1,202 degrees 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. 15) included pre-oxidation of seven samples at 800 degrees C (1,472 degrees 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 degrees C (1,472 degrees F) was benign with regard to breakaway oxidation.

C.2.C Evaluating ECCS Performance for Breakaway Oxidation

The regulations in 10 CFR 50.46c(g)(1)(iii) require that the total time the cladding is predicted to remain above the temperature that the zirconium-alloy cladding material has been shown to be susceptible to breakaway oxidation must be less that the analytical limit.

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 800 degrees C (1,472 degrees 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 Considering Oxygen Diffusion from the Cladding Inside Surface in ECCS Evaluation Models

The regulations in 10 CFR 50.46c (g)(2)(i) state that ECCS evaluation models must consider 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. 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.3.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 (OD), oxygen diffusion from the cladding inside diameter (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 due to 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. 16). The integral LOCA testing confirmed that continued exposure to high temperature steam environment 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 pre-existing cladding hydrogen content.

C.3.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.

A threshold for the onset of this inside surface oxidation source other than 30 GWd/MTU may be proposed by an applicant and provided as part of the documentation supporting the NRC staff's review and approval of the new or existing fuel design (i.e., 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. It should be noted 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

The purpose of this section is to provide information on how applicants and licensees[4] may use this guide and information regarding the NRC's plans for using this RG. In addition, it describes how the NRC staff complies with 10 CFR 50.109, "Backfitting" and any applicable finality provisions in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

Use by Applicants and Licensees

Applicants and licensees may voluntarily[5] use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this RG may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations. Current licensees may continue to use guidance the NRC found acceptable for complying with the identified regulations as long as their current licensing basis remains unchanged.

Licensees may use the information in this RG for actions that do not require NRC review and approval such as changes to a facility design under 10 CFR 50.59, "Changes, Tests, and Experiments." Licensees may use the information in this RG or applicable parts to resolve regulatory or inspection issues.

Use by NRC Staff

The NRC staff does not intend or approve any imposition or backfitting of the guidance in this RG. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this RG, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this RG to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action that would require the use of this RG. Examples of such unplanned NRC regulatory actions include issuance of an order requiring the use of the RG, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this RG, generic communication or promulgation of a rule requiring the use of this RG without further backfit consideration.

During regulatory discussions on plant specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this RG, as one acceptable means of meeting the underlying NRC regulatory requirement. Such discussions would not ordinarily be considered backfitting even if prior versions of this RG are part of the licensing basis of the facility. However, unless this RG is part of the licensing basis for a facility, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this RG constitutes a violation.

If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this new or revised RG and (2) the specific subject matter of this RG is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this RG or provide an equivalent alternative process that demonstrates compliance with the

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4 In this section, "licensees" refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52; and "applicants," refers to applicants for licenses and permits for (or relating to) nuclear power plants under 10 CFR Parts 50 and 52, and applicants for standard design approvals and standard design certifications under 10 CFR Part 52.

5 In this section, "voluntary" and "voluntarily" mean that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.

underlying NRC regulatory requirements. This is not considered backfitting as defined in 10 CFR 50.109(a)(1) or a violation of any of the issue finality provisions in 10 CFR Part 52.

Additionally, an existing applicant may be required to comply with new rules, orders, or guidance if 10 CFR 50.109(a)(3) applies.

If a licensee believes that the NRC is either using this RG or requesting or requiring the licensee to implement the methods or processes in this RG in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 17), and in NUREG-1409, "Backfitting Guidelines," (Ref. 18).

GLOSSARY

breakaway oxidation The fuel-cladding oxidation phenomenon in which the oxygen weight gain rate deviates from normal oxidation 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.

integral-time-at-temperature The process of embrittlement at high temperature is dependent on the diffusion of oxygen into the cladding and as such depends on the cladding temperature and the span of time for which the cladding is exposed to that temperature. A convenient way to establish this influence is through the application of a prediction of the oxide layer increase during the transient. ECR during the transient is calculated with the Cathcart-Pawel correlation and is used to integrate time-at-temperate in this Regulatory Guide.

loss-of-coolant accident A postulated accident that would result from in 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 post-test 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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2. NRC, RG 1.222, "Measuring Breakaway Oxidation Behavior," Washington, DC, DATE.

3. NRC, RG 1.223, "Determining Post Quench Ductility," Washington, DC, DATE.

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5. NRC, NUREG/IA-0211, "Experimental Study of Embrittlement of Zr-1%Nb VVER Cladding under LOCA-Relevant Conditions," Washington, DC, March 2005.

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11. IAEA Safety Guide NS-G-1.9, "Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants," issued September 2004.[7, 8]

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6 All NRC documents that are publicly available may be accessed through the Electronic Reading Room on NRC's public Web site 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 The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

7 Copies of International Atomic Energy Agency (IAEA) documents may be obtained through their Web site: WWW.IAEA.Org/ or by writing the International Atomic Energy Agency P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria. Telephone (+431) 2600-0, Fax (+431) 2600-7, or E-Mail at Official.Mail@IAEA.Org

8 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.

12. *Code of Federal Regulations* (CFR) Title 10, *Energy,* 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," Appendix K, "ECCS Evaluation Models."

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16. NRC, NUREG-2119, "Mechanical Behavior of Ballooned and Ruptured Cladding." Washington DC, February 2012.

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19. NRC, Memorandum from P. M. Clifford to T. J. McGinty, "Audit Report: Applicability of 50.46c Post Quench Ductility Analytical Limits to Westinghouse Optimized ZIRLO™ Cladding Material," July 28, 2015 (ADAMS Accession No. ML15209A314).

[9] 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.

[10] Copies of this paper may be purchased through ScienceDirect Website: http://www.sciencedirect.com

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 ECCS 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 database. Table 1 of Reference A-1 (reproduced 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 1 Hydrogen models in FRAPCON-3.4 and new model

Alloy	FRAPCON-3.4		New Model	
	Model	Std. dev.	Model	Std. dev.
BWR				
Zry-2 pre 1998	Eq. 2	10 ppm[1] NA[2]	Eq. 2	10 ppm[1] 54 ppm[2]
Zry-2 post 1998	Eq. 3	11 ppm[1] 61 ppm[2]	Eq. 3	13 ppm[1] 60 ppm[2]
PWR				
Zry-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 BWR Zircaloy-2

For boiling water reactor (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:

$$H = 47.8 \exp[-1.3/(1+BU)] + 0.316BU \qquad BU < 50 \text{ GWd/MTU}$$
$$H = 28.9 + \exp[0.117(BU-20)] \qquad BU > 50 \text{ GWd/MTU}$$

Modern alloys:

$$H = 22.8 + \exp[0.117(BU-20)]$$

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 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 on 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.

$$H = (47.8 \exp[-1.3/(1+BU)] + 0.316BU) * 1.40 \qquad BU < 50 \text{ GWd/MTU}$$
$$H = (28.9 + \exp[0.117(BU-20)]) * 1.40 \qquad BU > 50 \text{ GWd/MTU}$$

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-1 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 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, EFPD). 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 EFPD 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 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's 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 over 280 measurements. Figure A-3 shows predicted versus measured hydrogen concentration using the above 15.3 percent pickup fraction. With over 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 PNNL'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-1. 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

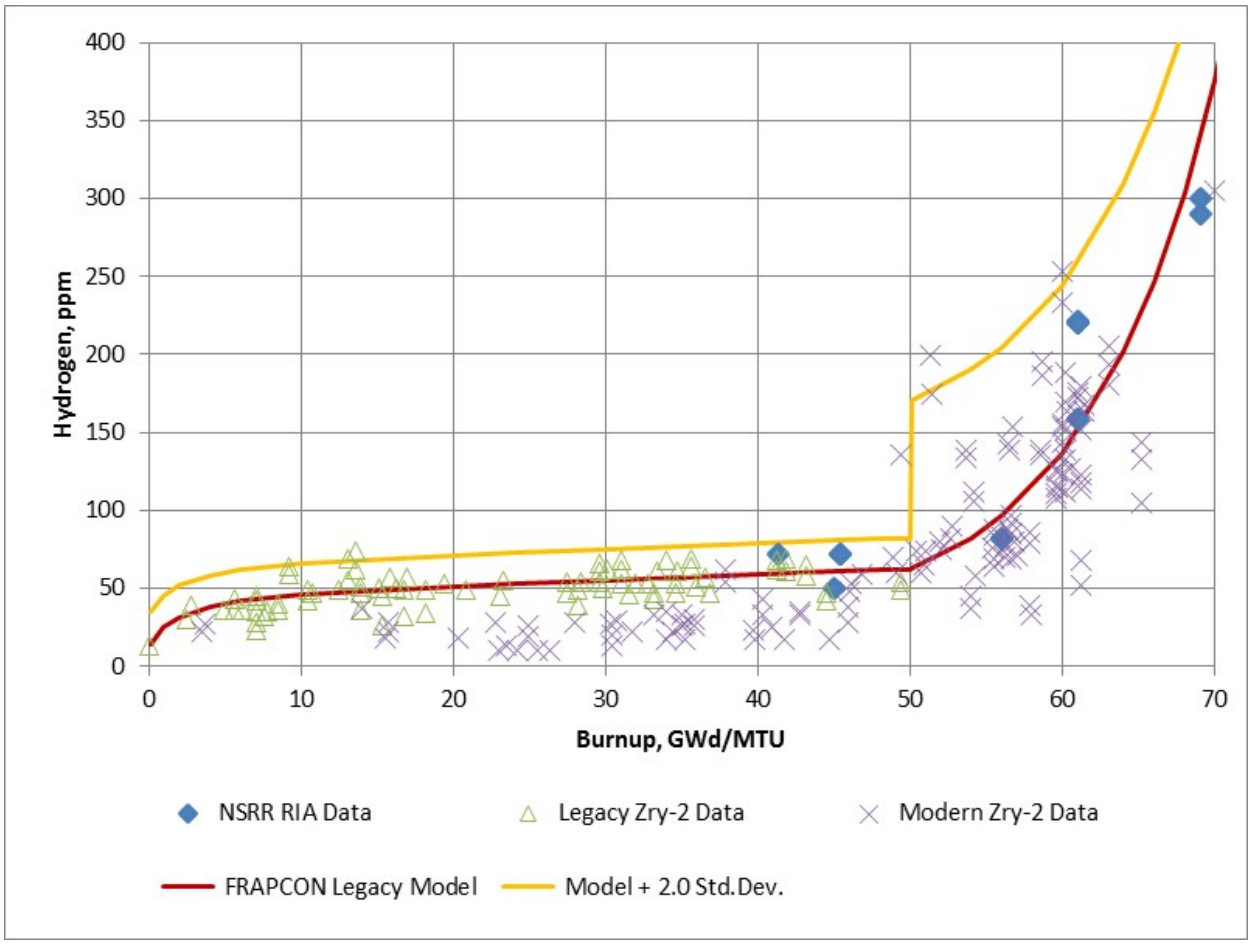

Figure A-1. Zircaloy-2 Hydrogen Model, +2-Sigma Prediction Versus Data

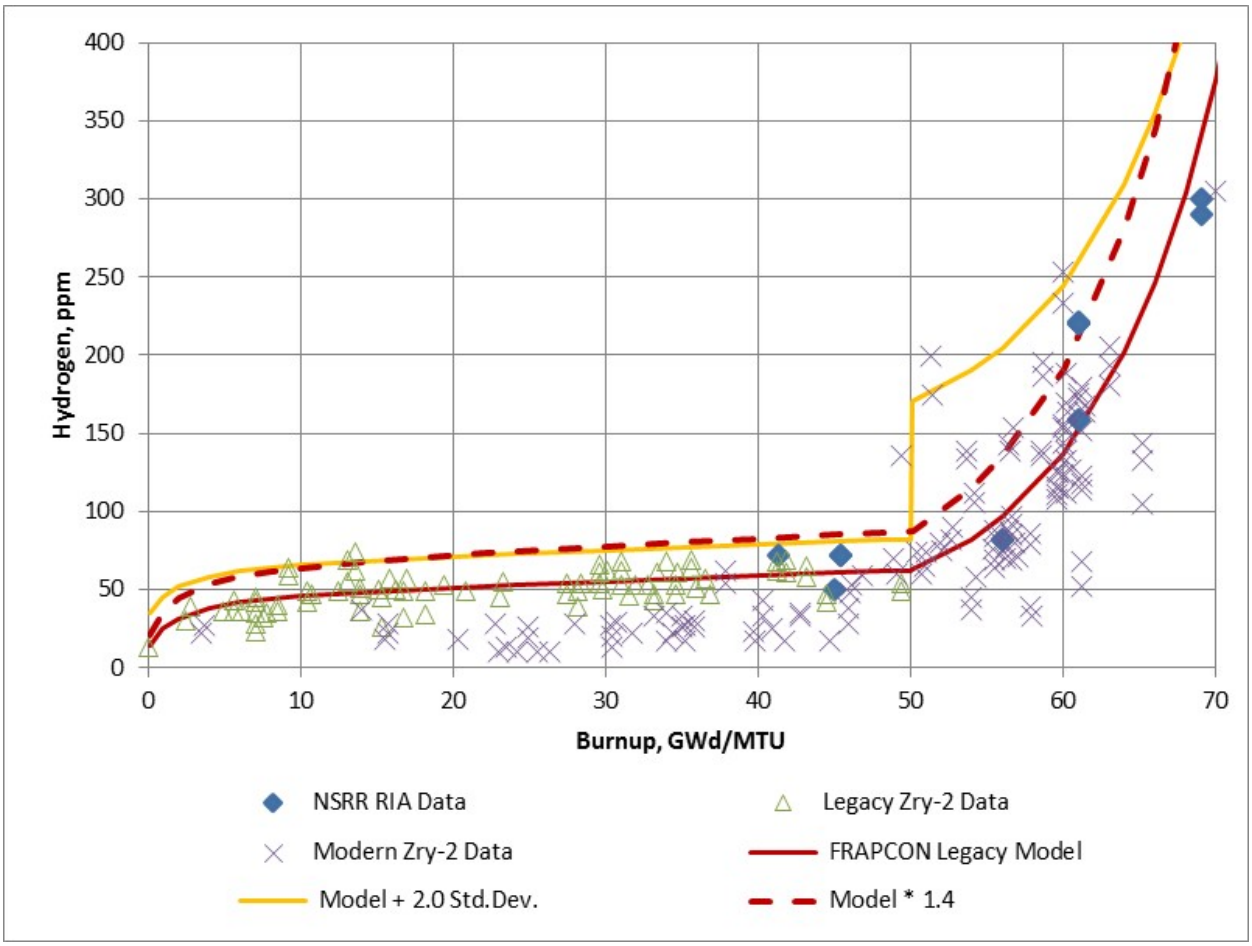

Figure A-2. Zircaloy-2 Hydrogen Model, 1.40 Multiplier Prediction Versus Data

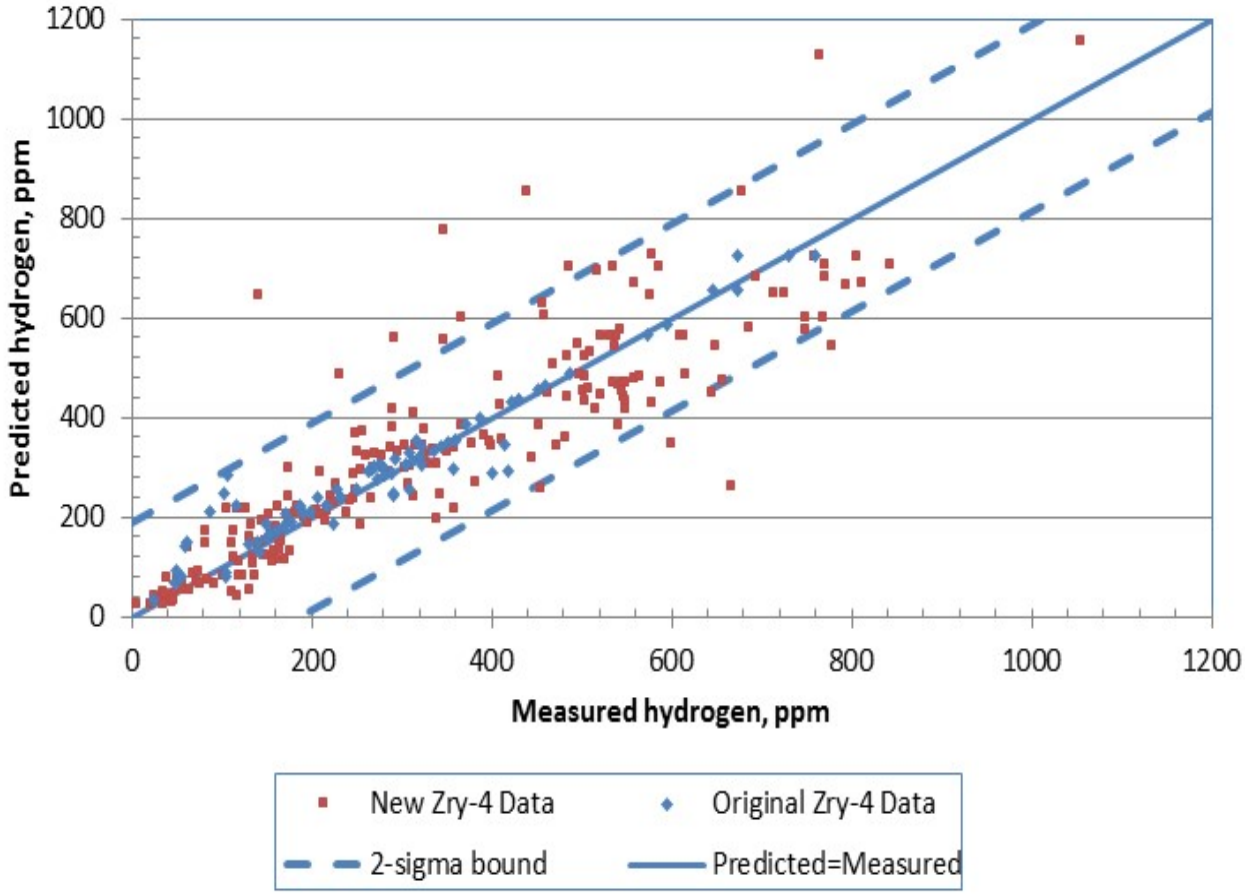

Figure A-3. Zircaloy-4 Predicted Versus Measured Hydrogen Content, 15.3% Pickup (Figure 4 of Reference A-1)

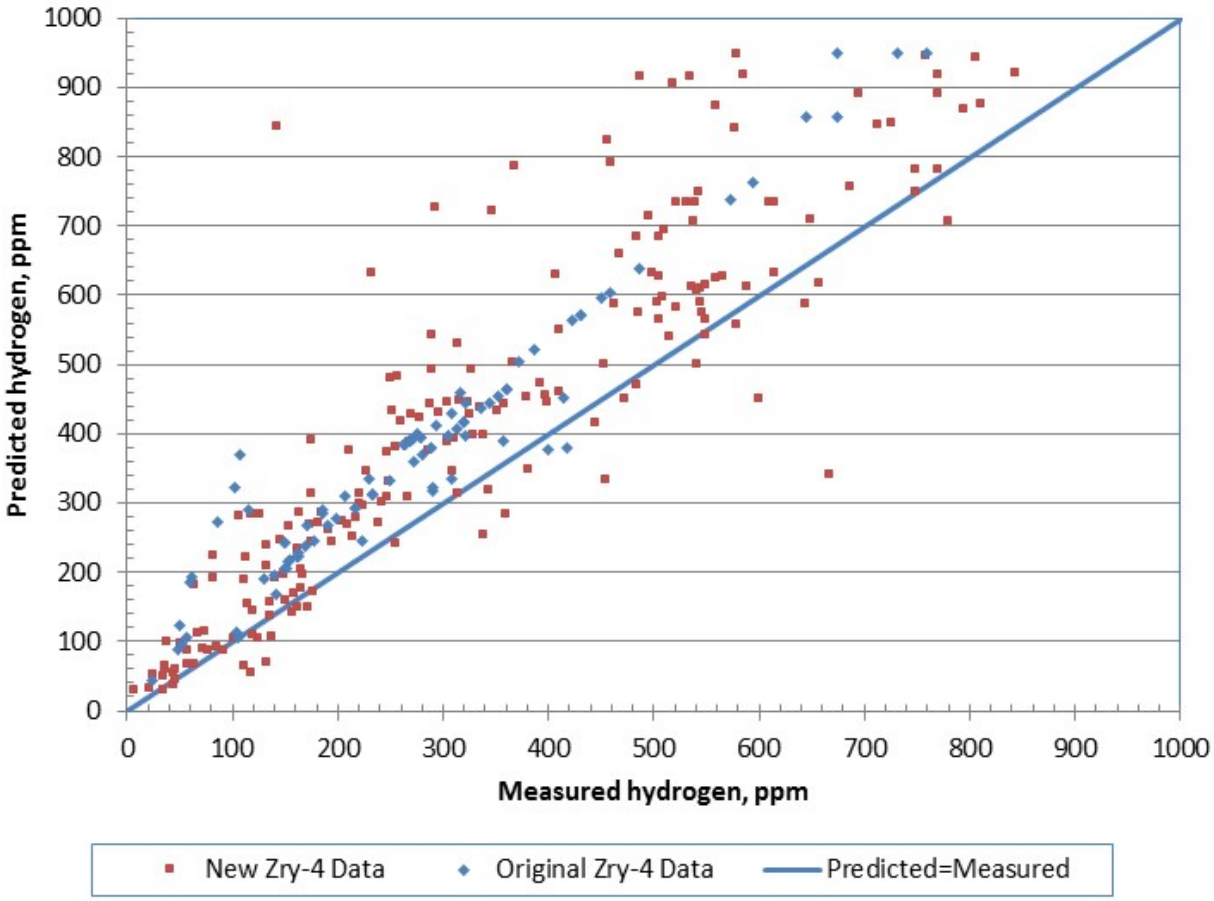

Figure A-4. Zircaloy-4 Predicted Versus Measured Hydrogen Content, 20.0% Pickup

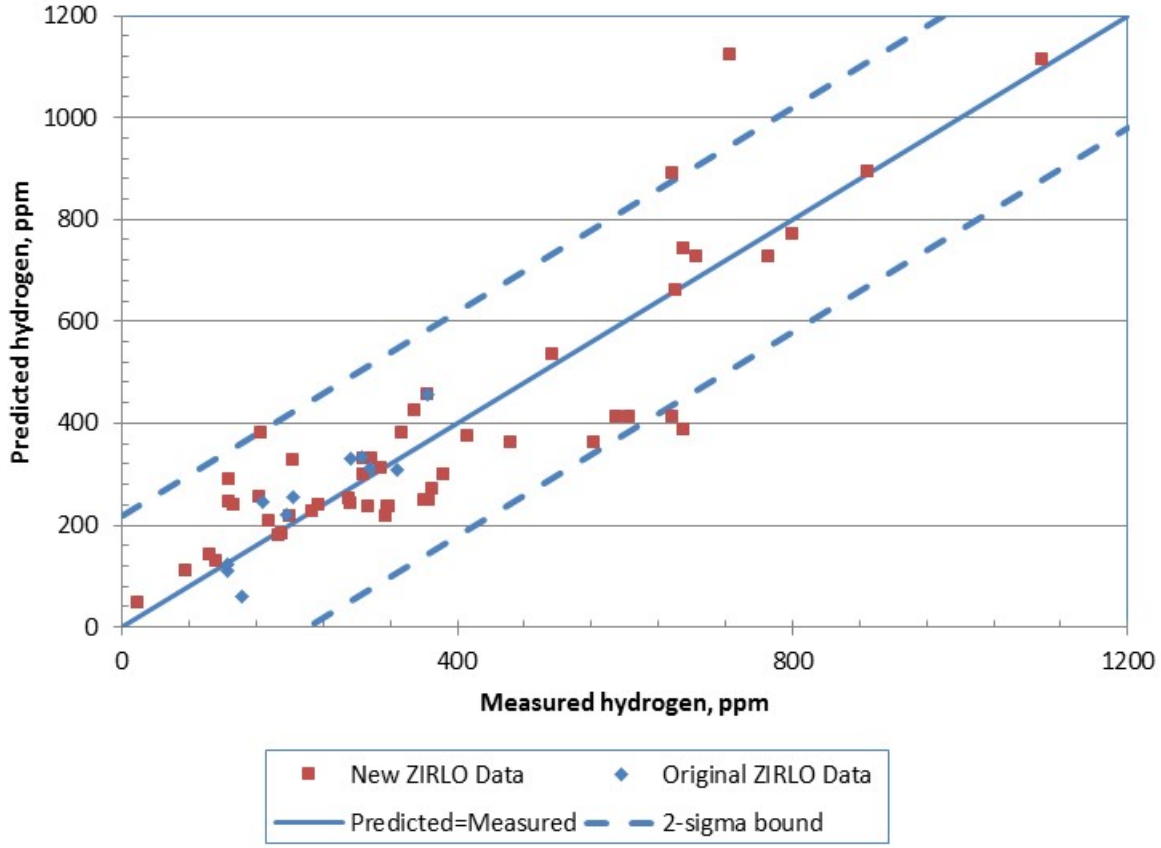

Figure. A-5: ZIRLO® Predicted Versus Measured Hydrogen Content, 17.3% Pickup (Figure 6 of Reference A-1)

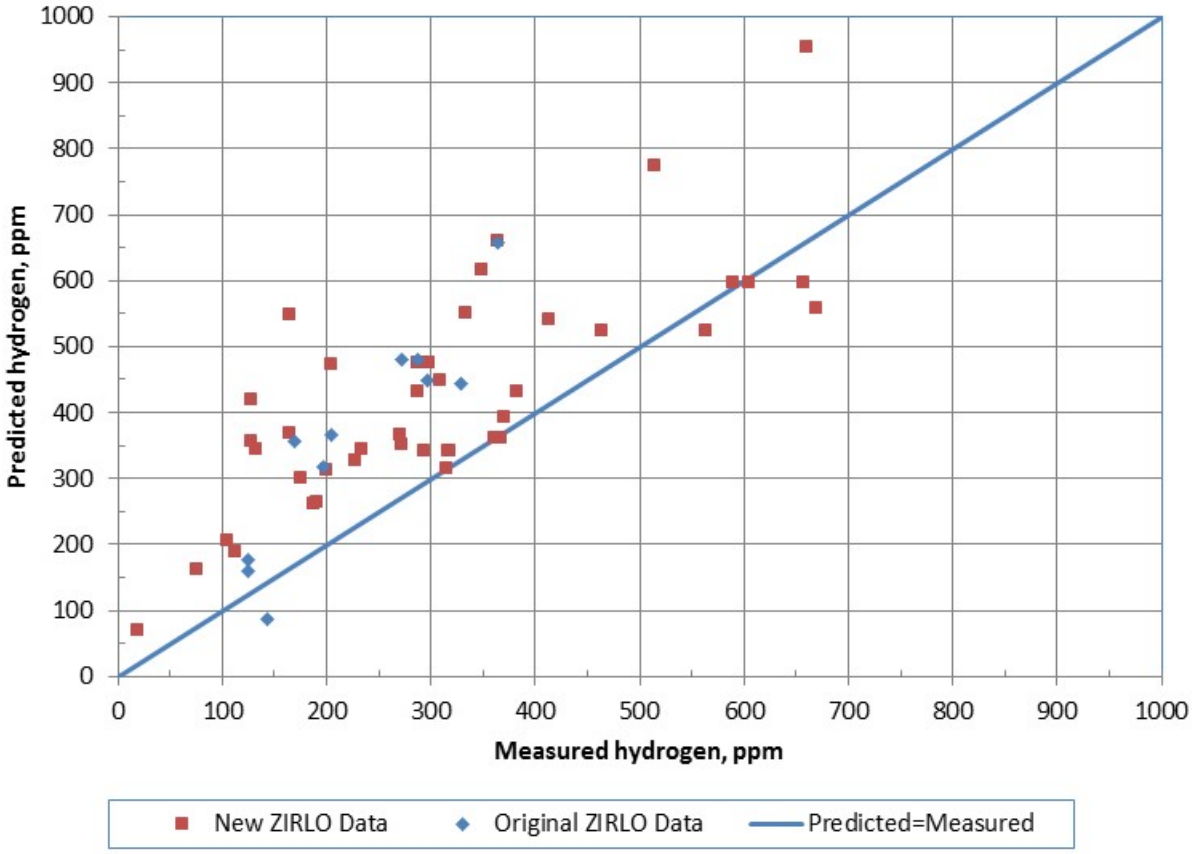

Figure A-6. ZIRLO® Predicted Versus Measured Hydrogen Content, 25.0% Pickup

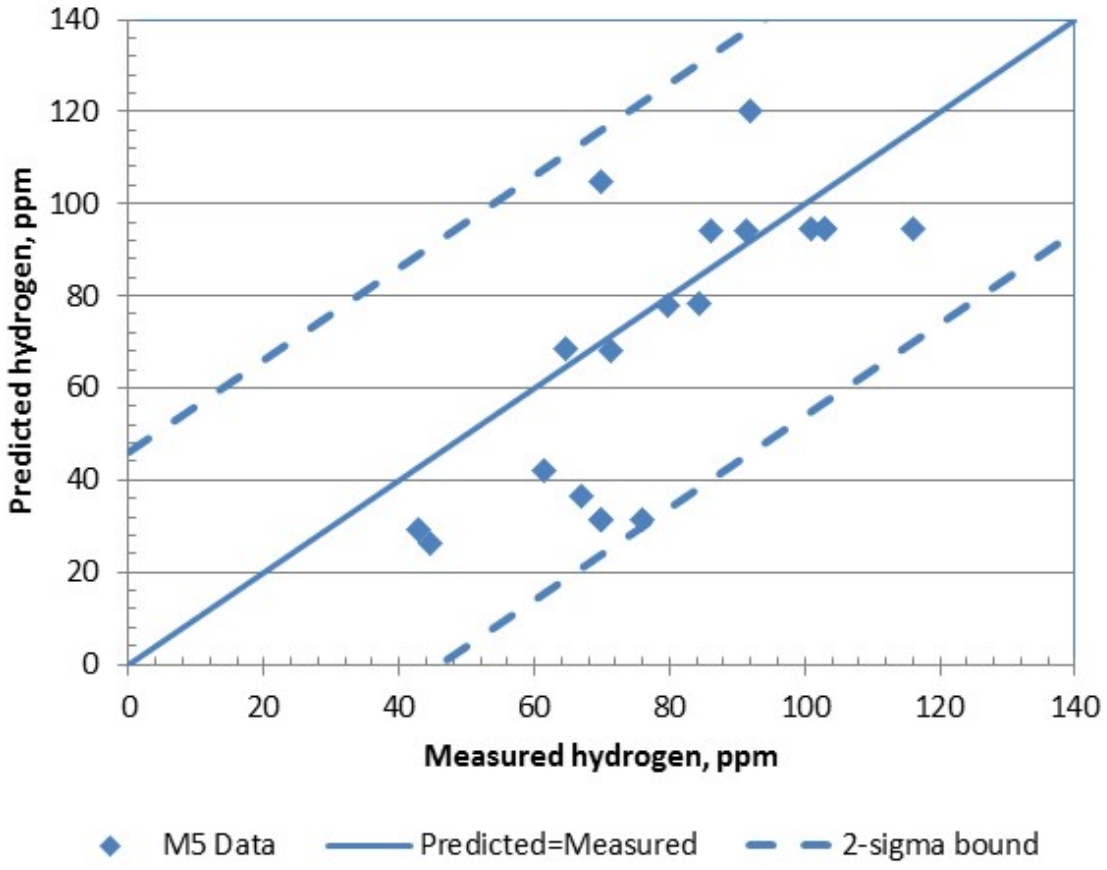

Figure A-7. M5® Predicted Versus Measured Hydrogen Content, 10% Pickup (Figure 8 of Reference A-1)

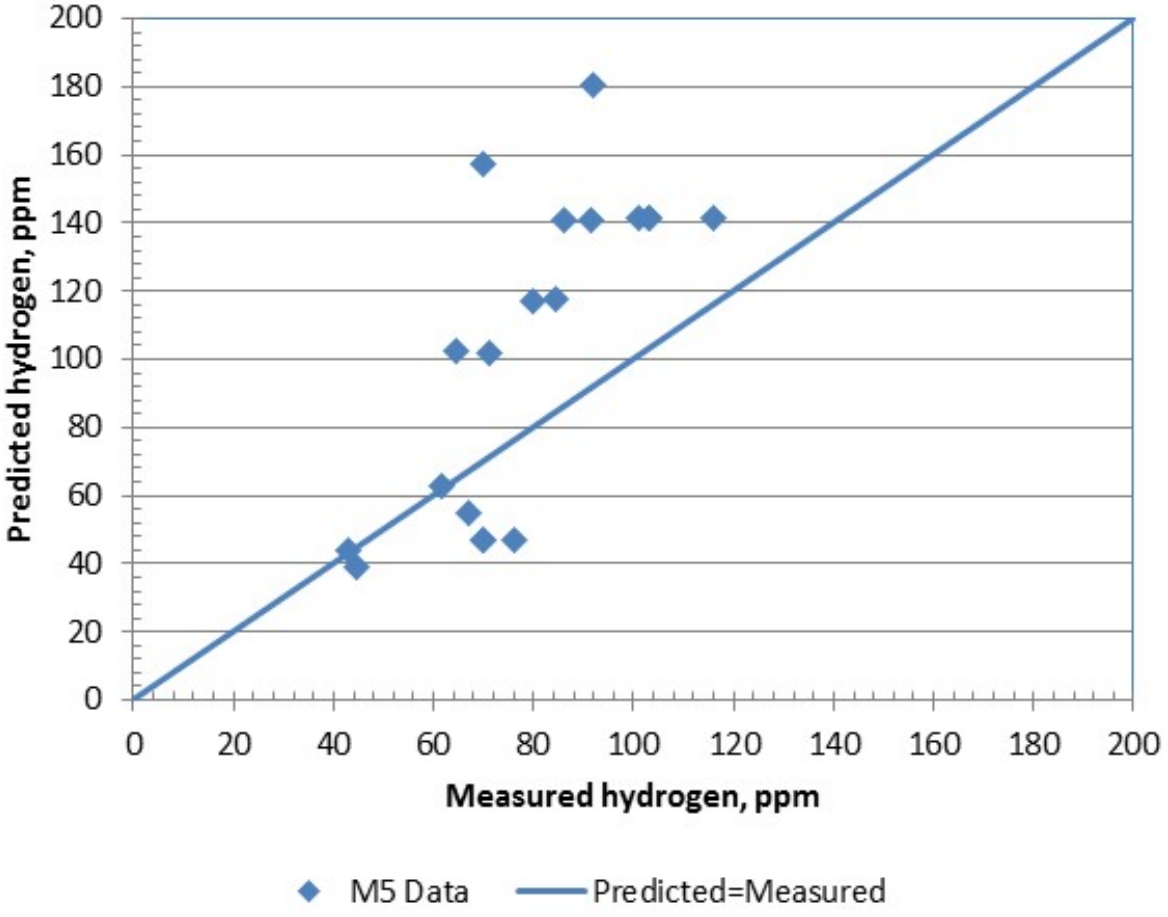

Figure A-8. M5® Predicted Versus Measured Hydrogen Content, 15% Pickup

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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

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1 All NRC documents that are publicly available may be accessed through the Electronic Reading Room on the NRC's public Web site 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 The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or 800-397-4209; fax 301-415-3548; or e-mail pdr.resource@nrc.gov.

2 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.

3 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.