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THE U. S. NUCLEAR REGULATORY COMMISSION'S STRATEGY FOR REVISING THE RIA ACCEPTANCE CRITERIA

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Abstract

In March 2007, the U.S. Nuclear Regulatory Commission (NRC) issued interim criteria and guidance for the reactivity-initiated accident (RIA) within NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 4.2, "Fuel System Design," Appendix B. Since its publication, re-examination of experimental data and improved analytical models have prompted further revision to the RIA criteria and guidance. The purpose of this paper is to describe proposed changes to Regulatory Guide (RG) 1.77 and to the Standard Review Plan.

1. Introduction

Reactivity-initiated accidents (RIAs) consist of postulated accidents which involve a sudden and rapid insertion of positive reactivity. These accident scenarios include a control rod ejection (CRE) for pressurized water reactors (PWRs) and a control rod drop accident (CRDA) for boiling water reactors (BWRs). The uncontrolled movement of a single control rod out of the core results in a prompt positive reactivity insertion, which increases local core power. Fuel temperatures rapidly increase, resulting in fuel pellet thermal expansion, an increase in cladding temperature, and cladding strain.

Regulatory criteria were established in 1974 to limit the extent of fuel rod fragmentation, thus preventing damage to the reactor coolant pressure boundary and ensuring core cooling capability. More recent results from RIA test programs in the United States, France, Japan, and Russia indicated that cladding failure may occur at much lower fuel enthalpy levels—particularly in highly corroded cladding. Consequently, the NRC issued revised criteria and guidance. In 2007, Revision 3 of SRP Section 4.2, Appendix B (Reference 1), was issued and captured the following changes in regulatory position and staff guidance:

1. The fuel cladding failure criteria was revised to include separate PWR and BWR criteria for both high cladding temperature failure and pellet-to-cladding mechanical interaction (PCMI) failure mechanisms.

2. The core coolability criteria was revised to specifically address both short-term (e.g., fuel-to-coolant interaction, rod burst) and long-term (e.g., fuel rod ballooning, flow blockage) phenomena which challenge coolable geometry and reactor pressure boundary integrity.

3. The fission-product inventory for dose calculations was revised to specifically account for transient-induced fission gas release.

2. Deficiencies Addressed by Interim RIA Criteria and Guidance

Previous regulatory criteria for RIA were designed to prevent extensive fragmentation of the fuel rod. The criteria did not recognize PCMI as a cladding failure mechanism during RIA nor did they account for effects associated with high exposure fuel and in-service cladding corrosion. As a result, fuel rod cladding failure may occur below the criteria specified in previous regulatory guidance. As a result, the associated radiological assessments may be non-conservative.

Previous radiological guidance does not account for the transient-induced fission gas release reported in several RIA test programs. The total fission-product inventory available for release includes the steady-state gap inventory plus any fission gas released from the pellet during the event. As a result, radiological assessments may be non-conservative.

Fuel enthalpy limits provided in RG 1.77 were incorrectly interpreted from the original SPERT and TREAT data. Further, this criteria did not account for fuel rod ballooning, fuel fragmentation, fuel dispersal, and the associated fuel-to-coolant interaction. As a result, the criteria specified in RG 1.77 do not adequately address core coolability requirements.

The revised RIA criteria and guidance provided in Appendix B of the Standard Review Plan, Section 4.2, addressed these deficiencies. The NRC issued these interim RIA criteria and guidance to support the initial licensing of the new reactor fleet. In addition, these interim criteria provided a target for the U.S. industry to develop improved analytical methods which would allow a more deliberate implementation once the NRC issued final criteria and guidance.

3. Development of Final RIA Criteria and Guidance

The NRC staff is working to finalize the RIA criteria and guidance. This effort involves several steps, including (1) revising technical and regulatory basis documents, (2) revising Regulatory Guide 1.77 and Appendix B of the Standard Review Plan, Section 4.2, (3) conducting public workshops and a comment session, (4) addressing public comments, and (5) issuing final documents.

Utilizing recent experimental data from the Nuclear Safety Research Reactor hot capsule tests along with improved analytical models, the NRC is conducting a critical assessment of the interim PCMI cladding failure threshold. This assessment evaluates the effects of the following attributes on the failure threshold:

1. Hydride morphology
2. Initial cladding temperature
3. Cladding alloy composition and heat treatment
4. MOX fuel
5. Power pulse width
6. Fuel design (e.g., initial gap size)

The assessment also evaluates the threshold’s applicability to future fuel rod designs and cladding alloys.

To account for the first-order effect of hydride orientation, the staff is developing separate cladding failure thresholds for cold worked stress relief annealed (CWSR) and fully recrystallized annealed (RXA) zirconium alloy cladding both at PWR operating conditions and BWR cold start-up conditions.

The NRC is also considering the Electric Power Research Institute (EPRI) fuel reliability program's proposed alternate PWR and BWR PCMI cladding failure thresholds (Reference 2). EPRI's approach for developing the alternative PCMI cladding failure curves combined experimental data from a variety of sources, including integral RIA-simulation tests and separate effects tests, with Falcon transient fuel rod analysis calculations.

Figure 1 provides a comparison of the SRP interim, EPRI, draft CWSR, and draft RXA PCMI cladding failure threshold curves, as a function of increase in radial average fuel enthalpy versus cladding hydrogen content, applicable to PWR operating conditions. Similarly, Figure 2 illustrates the various PCMI cladding failure threshold curves applicable to BWR cold start-up conditions. The draft failure thresholds are still a work in progress and are therefore subject to change.

Figure 1: PWR "Hot" PCMI Cladding Failure Threshold

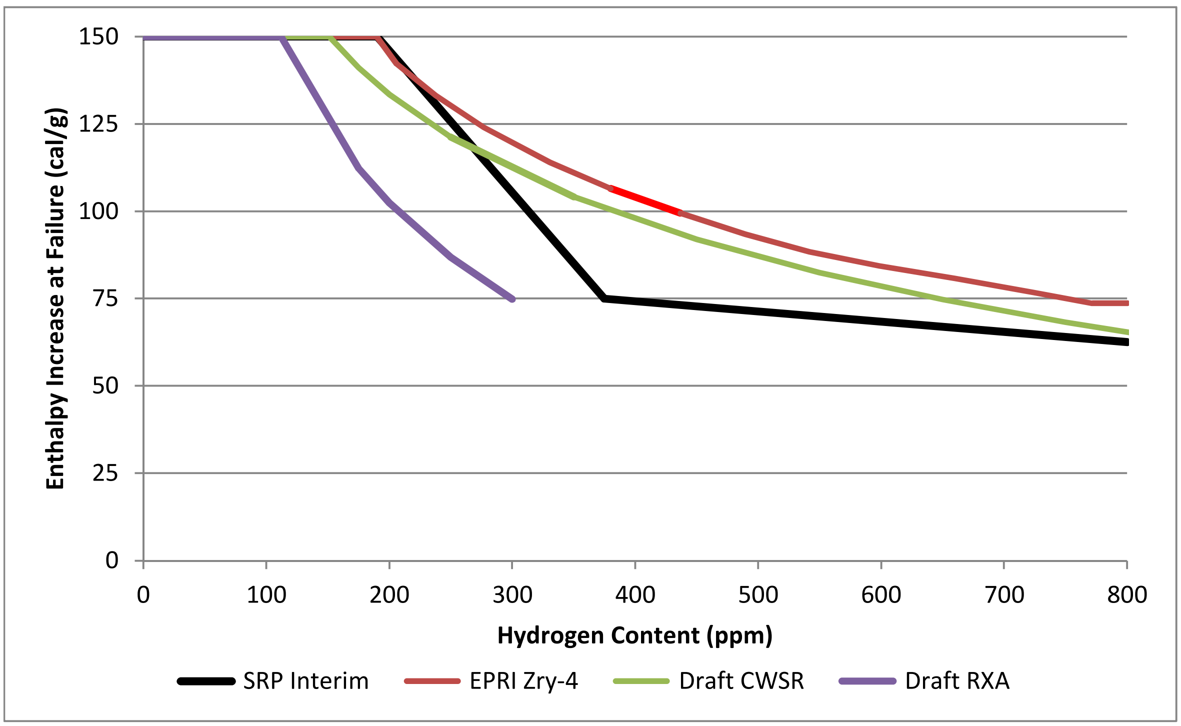
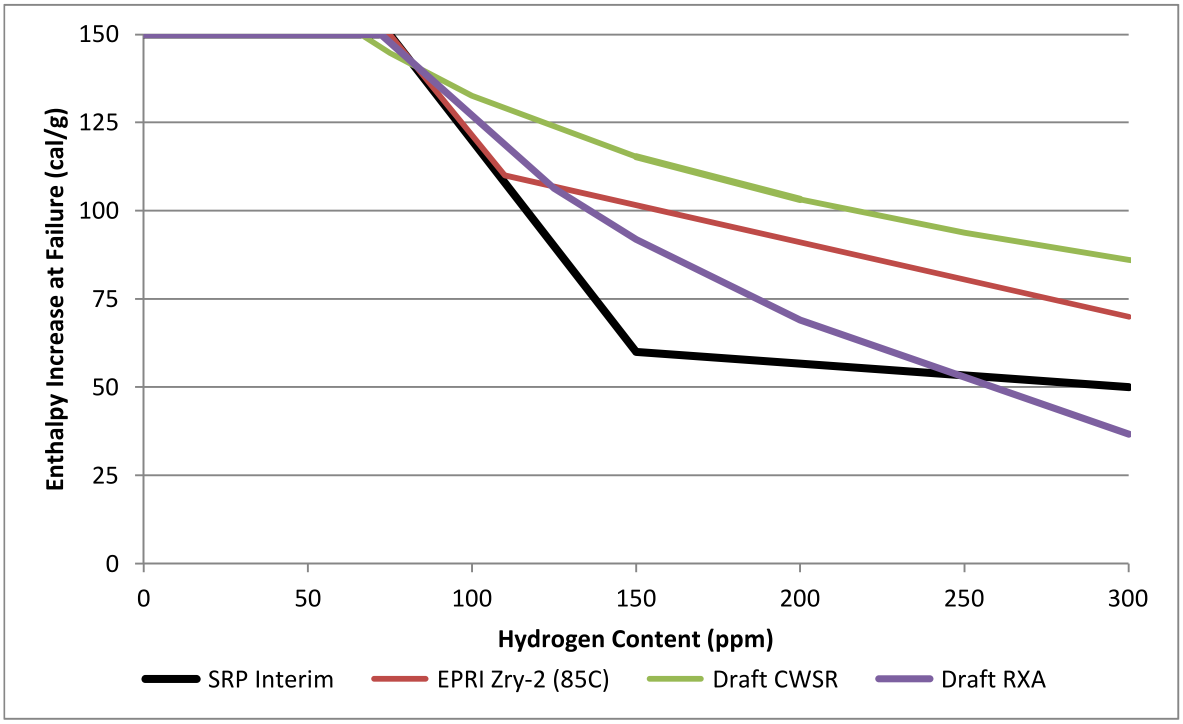


Figure 2: BWR “Cold” PCMI Cladding Failure Threshold



The NRC will be defining the range of applicability of the proposed PCMI cladding failure thresholds with respect to pulse width, cladding alloy, fuel design, and exposure. For example, the empirically-based PCMI cladding failure thresholds may not be applicable for PWR CRE and BWR CRDA scenarios which do not exhibit a prompt-critical, narrow pulse power excursion. Non-prompt scenarios are more likely to experience cladding failure due to high temperature cladding failure modes.

No new experimental data has become available to challenge the adequacy of the interim high temperature cladding failure criteria. As such, the following remains unchanged:

> The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure. For intermediate and full power conditions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g. DNBR or CPR).

The total number of fuel rods which must be considered in the radiological assessment is equal to the summation of all of the fuel rods failing each of the criteria above. Licensees do not need to double count fuel rods which are predicted to fail more than one of the criteria.

Based upon further assessment, the transient-induced fission gas release component of the total source term was revised. The new guidance was captured in a draft revision to RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” which was issued for public comment in 2010 (draft issued as DG-1199).

No new experimental data has become available to challenge the adequacy of the interim criteria for addressing short-term (e.g., fuel-to-coolant interaction, rod burst) and long-term (e.g., fuel rod ballooning, flow blockage) phenomena which challenge coolable geometry and reactor pressure boundary integrity. As such, the following remains unchanged:

1. Peak radial average fuel enthalpy must remain below 230 cal/g.

2. Peak fuel temperature must remain below incipient fuel melting conditions.

3. Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.

4. No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.

Fuel rod thermal-mechanical calculations, employed to demonstrate compliance to criteria #1 and #2 above, must be based upon design-specific information accounting for manufacturing tolerances and modeling uncertainties using NRC approved methods including burnup-enhanced effects on pellet power distribution, fuel thermal conductivity, and fuel melting temperature.

Upon completion of the technical and regulatory basis documents, the NRC staff will develop draft revisions to Regulatory Guide 1.77 and Appendix B of the Standard Review Plan, Section 4.2. The NRC will issue these draft documents for comment, and public workshops will be conducted to facilitate stakeholder involvement. After addressing public comments, the NRC will issue the final RIA criteria and guidance documents.

4. Implementation Strategy

As part of the process for revising RG 1.77 and Appendix B of the Standard Review Plan, Section 4.2, the staff will complete a backfit determination pursuit with 10 CFR 50.109. If the proposed change in regulatory staff position qualifies as either an exception (e.g., compliance, adequate protection) or cost-justified substantial increase in safety under the provisions of 10 CFR 50.109, the staff will propose an implementation schedule for applying the final criteria and guidance to both the current reactor fleet and the new reactor fleet.

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 RIA and (2) the specific subject matter of this new guidance 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 new guidance or provide an equivalent alternative process that demonstrates compliance with the 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.

REFERENCES

1. NUREG-0800, Standard Review Plan, Chapter 4.2, "Fuel System Design," Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents," March 2007.

2. EPRI Technical Report 1021036, "Fuel Reliability Program: Proposed RIA Acceptance Criteria," December 2010.