U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE RG 1.236

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Technical Lead: Paul Clifford

PRESSURIZED-WATER REACTOR CONTROL ROD EJECTION AND BOILING-WATER REACTOR CONTROL ROD DROP ACCIDENTS

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes methods and procedures that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable when analyzing the nuclear reactor's initial response to a postulated control rod ejection (CRE) accident for pressurized-water reactors (PWRs) and a postulated control rod drop (CRD) accident for boiling-water reactors (BWRs). It describes analytical limits and guidance for analyzing the short-term reactivity insertion and demonstrating compliance with 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 28, "Reactivity Limits." It also defines fuel cladding failure thresholds, including ductile failure, brittle failure, and pellet-clad mechanical interaction (PCMI), to support radiological consequence assessments. To facilitate implementation, this guide also provides acceptable analytical models for cladding hydrogen uptake and transient fission gas release (FGR).

Applicability

This guide applies to applicants and reactor licensees subject to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2).

Applicable Regulations

- 10 CFR Part 50 provides for the licensing of production and utilization facilities.
    - 10 CFR Part 50, Appendix A, GDC 28, "Reactivity Limits," requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure

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vessel internals to impair significantly the capability to cool the core. GDC 28 also requires that these postulated reactivity accidents consider rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold-water addition. Plants whose construction permits were issued before the GDC in 10 CFR Part 50, Appendix A, were promulgated in February 1971 have similar design criteria associated with reactivity limits.

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 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," provides guidance for reviewing PWR CRE accidents.

    - SRP Section 15.4.9, "Spectrum of Rod Drop Accidents (BWR)," provides guidance for reviewing BWR CRD accidents.

    - SRP Section 4.2, "Fuel System Design," provides guidance for reviewing reactor fuel designs.

    - SRP Section 4.2, Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents," provides guidance for reviewing both PWR CRE and BWR CRD accidents.

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Ref. 4), provides guidance for calculating radiological consequences for design-basis accidents.

- RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors" (Ref. 5), provides guidance for calculating radiological consequences for design-basis accidents.

- RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors" (Ref. 6), provides guidance for evaluating CRE for PWRs.

- RG 1.203, "Transient and Accident Analysis methods" (Ref. 7), describes a process that the staff considers acceptable for use in developing and accessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.

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 issues or postulated events, 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 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), approval numbers 3150-0011 and 3150-0151. Send comments regarding this information collection to the Information Services Branch (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, DC20503; e- mail: oira_submission@omb.eop.gov.

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B. DISCUSSION

Reason for Issuance

This guide provides guidance on acceptable analytical methods, assumptions, and limits for evaluating the nuclear reactor's initial response to a postulated PWR CRE and a postulated BWR CRD accident, based on empirical data from in-pile, prompt power pulse test programs and analyses from several international publications that examine fuel rod performance under prompt power excursion conditions. Appendix B provides acceptable transient FGR models and Appendix C provides acceptable cladding hydrogen uptake models.

Background

Reactivity insertion accidents are safety significant because of their potential ability to challenge fuel rod integrity, fuel bundle geometry, and the integrity of the reactor pressure boundary. The uncontrolled movement of a single control rod out of the core results in a positive reactivity insertion that promptly increases local core power and is considered the limiting reactivity insertion accident. Of the various postulated single failures of the CRD system which may initiate an uncontrolled movement of a single control rod, the PWR CRE and BWR CRD are considered the most limiting scenarios for the current operating fleet. The following examples of event descriptions and sequence of events were extracted from an existing plant's Updated Final Safety Analysis Report (UFSAR).

> PWR CRE:
> The rod ejection accident is caused by a failure of a control rod drive mechanism housing, which allows a control rod to be rapidly ejected from the reactor by the Reactor Coolant System (RCS) pressure. The control rod is ejected in 0.15 seconds from the fully inserted position. A power excursion will result, and if the reactivity worth of the ejected control rod is large enough, the reactor will become prompt critical. The resulting power excursion will be limited by the fuel temperature feedback and the accident will be terminated when the Reactor Protective System (RPS) trips the reactor on high neutron flux or high RCS pressure. RCS pressure increases due to the core power excursion, and pressurizer spray, the pressurizer PORV, and the pressurizer code safety valves will respond to mitigate the pressure increase. If a rod ejection were to occur, the nuclear design of the reactor and limits on control rod insertion will limit any potential fuel damage to acceptable levels. Cladding failure can result from the core power excursion and the highly peaked core power distribution near the ejected rod location. The failure of the control rod drive mechanism housing also constitutes a 1.50 inch diameter small-break loss-of-coolant accident (SBLOCA). The Emergency Core Cooling System (ECCS) will actuate on low RCS pressure or high Reactor Building pressure and will maintain core cooling. This type of SBLOCA is bounded by the limiting SBLOCA analyses.
>
> BWR CRD:
> The accidents that result in releases of radioactive material from the fuel with the nuclear system process barrier, primary containment, and secondary containment initially intact are the results of various failures of the Control Rod Drive System. Examples of such failures are collet finger failures in one control rod drive mechanism, a control drive system pressure regulator malfunction, and a control rod drive mechanism ball check valve failure. None of the single failures associated with the control rods or the control rod system results in a greater release of radioactive material from the fuel than the release that results when a single control rod drops out of the core after being disconnected from its drive and after the drive has been retracted to the fully withdrawn position. Thus, this control rod drop accident is established as the design basis accident for the category of accidents resulting in radioactive material release from the fuel with

all other barriers initially intact. A highly improbable combination of actual events would be required for the design basis control rod drop accident to occur. The actual events required are as follows: (1) failure of the rod-to-drive coupling, (2) sticking of the control rod in its fully inserted position as the drive is withdrawn, (3) full withdrawal of the control rod drive, (4) failure of the operator to notice the lack of response of neutron monitoring channels as the rod drive is withdrawn, and (5) failure of the operator to verify rod coupling.

The above postulated accident scenarios result in a positive reactivity insertion that promptly increases local core power. Fuel temperatures rapidly increase, causing fuel pellet thermal expansion. The reactivity excursion is initially mitigated by the Doppler feedback and delayed neutron effects followed by a reactor trip. The prompt thermal expansion of the fuel pellet, which can be exacerbated at high burnups by gaseous fission product swelling, may cause the fuel cladding to fail by PCMI. The potential for failure is enhanced by the presence of hydrogen in the cladding. Depending on the initial conditions, fuel cladding may also fail in a brittle fashion from oxygen-induced embrittlement or in a ductile fashion from rod ballooning and subsequent rupture. Any fuel rod that experiences cladding failure will release a portion of its fission product inventory to the reactor coolant system. Radiological consequences resulting from the release of these fission products should be limited to meet applicable regulations.

The NRC staff initially provided guidance for PWR CRE in RG 1.77 in 1974. The state of knowledge of fuel rod performance under prompt power excursion conditions has increased significantly since publication of that guidance. This knowledge prompted the need for new guidance to build on the enhanced database drawn from operating experience and controlled experiments. The empirical database has expanded from the earlier Special Power Excursion Test Reactor (SPERT) and Transient Reactor Test Facility (TREAT) research programs (which formed the basis of the initial RG 1.77 analytical limits) to include test results from the Power Burst Facility (PBF) as well as significant, more recent contributions from international research programs at the CABRI research reactor (France), Nuclear Safety Research Reactor (NSRR) (Japan), Impulse Graphite Reactor (IGR) (Russian Federation), and Fast Pulse Graphite Reactor (BIGR) (Russian Federation).

In 2007, the staff provided interim acceptance criteria and guidance in Appendix B of Section 4.2 of the staff's standard review plan (SRP) in NUREG-0800. The basis for the revision was provided in NRC Memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Interim Acceptance Criteria and Guidance" (Ref. 8). In 2015, the staff evaluated newly published empirical data and analyses and identified further changes to guidance in the NRC memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1" (Ref. 9).

As a result of further evaluation and more recent controlled experiments, the NRC staff decided to update and consolidate its guidance in this new RG 1.236. RG 1.236 contains state-of-the-art guidance which replaces the legacy guidance found in RG 1.77 and SRP Section 4.2 Appendix B. The NRC staff believed that RG 1.77 was no longer applicable or acceptable for contemporary analytical methods and fuel designs. In addition, RG 1.77 was only applicable to PWRs whereas RG 1.236 addresses both PWRs and BWRs. For those reasons, RG 1.77 was withdrawn concurrently with issuance of RG 1.236. Although withdrawn, current licensees with RG 1.77 in their licensing basis may continue to use it, and withdrawal does not affect any existing licenses or agreements.

Harmonization with 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 standards for protecting people and the environment from harmful effects of ionizing radiation. These standards provide a system of safety fundamentals, safety requirements, and safety guides reflecting an international consensus on what constitutes a high level of safety. The following IAEA Safety Guide and Technical Report include technical principles and guidance to be considered for analyzing a postulated CRE accident for PWRs and a postulated CRD accident for BWRs.

- IAEA SSG-52, “Design of the Reactor Core for Nuclear Power Plants” (2019) has information on fuel pellet-cladding interaction (Ref. 10).

- IAEA Technical Report Series No. 354, “Reactivity Accidents,” (1993) contains two main parts. The first and larger part (Sections 1–4) provides an overview of reactivity accidents: how they can arise, the basic principles of the defense against them, methods of analysis and acceptance criteria. The second part summarizes more recent work on beyond-design-basis reactivity accidents, experiments on fuel behavior, methods for analyzing design basis accidents, and beyond deign basis accidents (Ref. 11).

This RG incorporates similar design and performance guidelines and is consistent with the safety principles provided in these publications.

Documents Discussed in Staff Regulatory Guidance

This RG endorses the use of one or more codes or standards developed by external organizations and other third party guidance documents. These codes, standards and third party guidance documents may contain references to other codes, standards or third party guidance documents (“secondary references”). If a secondary reference has itself been incorporated by reference into NRC regulations as a requirement, then licensees and applicants must comply with that standard as set forth in the regulation. If the secondary reference has been endorsed in a RG as an acceptable approach for meeting an NRC requirement, then the standard constitutes a method acceptable to the NRC staff for meeting that regulatory requirement as described in the specific RG. If the secondary reference has neither been incorporated by reference into NRC regulations nor endorsed in a RG, then the secondary reference is neither a legally-binding requirement nor a “generic” NRC approved acceptable approach for meeting an NRC requirement. However, licensees and applicants may consider and use the information in the secondary reference, if appropriately justified, consistent with current regulatory practice, and consistent with applicable NRC requirements.

C. STAFF REGULATORY GUIDANCE

This section describes analytical methods and limits that the staff of the NRC considers acceptable for use when analyzing a postulated PWR CRE accident and a postulated BWR CRD accident.

1. Limits on Applicability

The analytical limits and guidance described are not applicable to anticipated operational occurrences (AOOs) and other postulated accidents involving positive reactivity insertion (e.g., PWR excess load, PWR inadvertent bank withdrawal, PWR steam line rupture, BWR turbine trip without bypass, BWR rod withdrawal error). Furthermore, depending on design features, reactor kinetics, and accident progression, this guide may not apply directly to advanced light-water reactors (LWRs) and small or modular LWRs. The staff will consider application of this guide beyond PWR CRE and BWR CRD, as well as the range of applicability described below, on a case-by-case basis.

The applicability of the fuel rod cladding failure thresholds, fission product release fractions, and allowable limits on damaged core coolability provided in this guidance is limited as follows.

1.1 The applicability of this guidance is limited to approved LWR fuel rod designs comprising slightly enriched uranium dioxide (UO_2) ceramic pellets (up to 5.0 wt% uranium-235) within cylindrical zirconium-based cladding, including designs with or without barrier lined cladding, an integral fuel burnable absorber (e.g., gadolinium), or a pellet central annulus.

1.1.1 The applicability of this guidance to future LWR fuel rods designs (e.g., doped pellets, changes in fuel pellet microstructure or density, changes in zirconium alloy cladding microstructure or composition, coated zirconium alloy cladding) will be addressed on a case-by-case basis.

1.1.2 The guidance is not applicable to non-UO_2 fuels, such as mixed oxide (MOX) fuel rod designs, and non-zirconium based cladding alloys.

1.1.3 The applicability of this guidance is limited to a maximum fuel rod average burnup of 68 GWd/MTU.

1.1.4 The guidance is not applicable to fuel rods with pre-existing cladding failure (i.e., leaking, waterlogged) or excessive cladding oxidation exhibiting localized imperfections (e.g., spallation, hydride blisters).

1.2 As described in Section 3.2, separate PCMI cladding failure thresholds are provided for different initial cladding temperatures and different cladding thermal annealing treatments.

1.2.1 The high-temperature PCMI cladding failure threshold curves apply to initial, local reactor conditions involving cladding temperatures at or above 500 degrees Fahrenheit (F). For initial cladding temperatures below 500 degrees F, the low-temperature PCMI cladding failure threshold curves apply.

1.2.2 The recrystallized annealed (RXA) PCMI cladding failure threshold curves apply to cladding that has undergone final thermal treatment that produces an RXA metallurgical state, while the stress relief annealed (SRA) PCMI cladding failure threshold curves apply to cladding that has undergone final thermal treatment that produces an SRA metallurgical state. For any other metallurgical condition, the licensee or applicant should justify its similarity to either the SRA or RXA metallurgical condition.

2. Physics and Thermal-Hydraulics Analytical Methods and Assumptions

The staff considers the following analytical inputs, assumptions, and methods to be acceptable for evaluating the postulated CRE and CRD accidents.

2.1 Methods and Models

2.1.1 Accident analyses should be performed using NRC-approved analytical models and application methodologies. The analytical models and computer codes used should be documented and justified, and the conservatism of the models and codes should be evaluated. Comparison with experiments or more sophisticated spatial kinetics codes should be performed. In particular, the importance of two- or three-dimensional flux characteristics and changes in flux shapes should be investigated, and the conservatism of the flux shapes used for reactivity input and feedback, peak energy deposition, total energy, and gross heat transfer to the coolant should be evaluated. Also, sensitivity studies on variations of the Doppler effect, power distribution, pellet radial power profile, fuel element heat transfer parameters, and other relevant parameters should be included.

In statistically based accident analyses, analytical uncertainties should be quantified, and their application fully justified.

The staff recognizes that, if the guidance provided in this section is employed for the evaluation of postulated CRE and CRD accidents, the overlapping portions of RG 1.203 do not need to be applied.

2.1.2 The computer code used for calculating the transient should be a coupled thermal, hydrodynamic, and nuclear model with the following capabilities: (1) incorporation of all major reactivity feedback mechanisms, (2) at least six delayed neutron groups, (3) both axial and radial segmentation of the fuel element, (4) coolant flow provision, and (5) control rod scram initiation.

2.1.3 Calculations should be based on design-specific information.

2.1.4 Fuel enthalpy calculations should account for burnup-related effects on reactor kinetics (e.g., β_{eff}, l*, rod worth, Doppler effect) and fuel performance (e.g., pellet radial power distribution, fuel thermal conductivity, fuel-clad gap conductivity, fuel melting temperature).

2.2 Initial Conditions

2.2.1 PWR CRE Initial Conditions

2.2.1.1 Accident analyses should consider the full range of cycle operation from beginning of cycle (BOC) to end of cycle (EOC).

2.2.1.2 Accident analyses at hot zero power should encompass both (1) BOC following core reload and (2) restart following recent power operation.

2.2.1.3 Accident analyses should consider the full range of power operation including intermediate power levels up to hot full-power conditions. These calculations should consider power-dependent core operating limits (e.g., control rod insertion limits, rod power peaking limits, axial and azimuthal power distribution limits). At conditions where certain core operating limits do not apply, the analysis should consider the potential for wider operating conditions resulting from xenon oscillations or plant maneuvering.

When properly justified, cycle-independent bounding evaluations that demonstrate that regions of power operation are less limiting are an acceptable analytical approach to reduce the number of cases analyzed. For example, during CRE scenarios initiated from at-power conditions, credit for power-dependent insertion limits in the technical specifications may be used to demonstrate that these particular events are of less significance with respect to coolable geometry.

2.2.1.4 Uncontrolled worth of an ejected rod should be calculated based on the following conditions: (1) the range of control rod positions allowed at a given power level and (2) additional fully or partially inserted misaligned or inoperable rod(s) if allowed. Applicants do not need to consider dropped or misaligned rods which are being recovered within technical specifications limiting conditions for operation (TS LCO) completion times.

Sufficient parametric studies should be performed to determine the worth of the limiting control rod for the allowed configurations highlighted above. The evaluation methodology should account for (1) calculation uncertainties in neutronic parameters (e.g., neutron cross sections) and (2) allowed power asymmetries.

2.2.1.5 Because of burnup-dependent and corrosion-dependent factors that tend to reduce cladding failure thresholds and allowable limits on core coolability during fuel rod lifetime, the limiting initial conditions may involve locations other than the maximum uncontrolled rod worth defined in Regulatory Position C.2.2.1.4 (e.g., uncontrolled rod motion at a core location adjacent to higher burnup fuel assemblies). For this reason, a more comprehensive search for the limiting conditions may be necessary to ensure that the total number of fuel rod failures is not underestimated, and allowable limits are satisfied. Applicants may need to survey a larger population of PWR ejected rod core locations and exposure points to identify the limiting scenarios.

When properly justified, combining burnup-dependent parameters to create an artificial, composite worst time-in-life (e.g., end-of-life cladding hydrogen content combined with maximum ejected worth) is an acceptable analytical approach to reducing the number of cases analyzed.

2.2.1.6 The reactivity insertion rate should be determined from differential control rod worth curves and calculated transient rod position versus time curves.

2.2.1.7 The rate of ejection should be calculated based on the maximum pressure differential and the weight and cross-sectional area of the control rod and drive shaft, assuming no pressure barrier restriction.

2.2.1.8 The initial reactor coolant pressure, core inlet temperature, and flow rate used in the analysis should be conservatively chosen, depending on the transient phenomenon being investigated. The range of values should encompass the allowable operating range and monitoring uncertainties.

2.2.1.9 Fuel thermal properties (e.g., fuel-clad gap thermal conductivity, fuel thermal conductivity) should cover the full range over the fuel rod’s lifetime and should be conservatively selected based on the transient phenomenon being investigated. Time-in-life specific fuel properties may be used for a given burnup-specific statepoint analysis.

2.2.1.10 The moderator reactivity feedback resulting from voids, coolant pressure changes, and coolant temperature changes should be calculated based on the various assumed conditions of the fuel and moderator using standard transport and diffusion theory codes. If boric acid shim is used in the moderator, the highest boron concentration corresponding to the initial reactor state should be assumed. If applicable, the range of values should encompass the allowable operating range (i.e., technical specifications in the core operating limits report) and any applicable analytical uncertainties.

2.2.1.11 Calculations of the Doppler reactivity feedback should be based on and compared with available experimental data. Since the Doppler feedback reflects the change in reactivity as a function of fuel temperature, uncertainties in predicting the coefficient, as well as in predicting fuel temperatures at different power levels, should be reflected by conservative application of Doppler feedback.

2.2.1.12 Control rod reactivity insertion during trip versus time should be obtained by combining the differential rod worth curve with a rod velocity curve based on maximum design limit values for scram insertion times. Alternatively, reactivity may be calculated using control rod velocity during trip based on maximum design limit values for scram insertion times. Any loss of available scram reactivity resulting from allowable rod insertion should be quantified.

2.2.1.13 The reactor trip delay time, or the amount of time that elapses between the instant the sensed parameter (e.g., pressure, neutron flux) reaches the level for which protective action is required and the onset of negative reactivity insertion, should be based on maximum values of the following: (1) time required for the instrument channel to produce a signal, (2) time for the trip breaker to open, (3) time for the control rod motion to initiate, and (4) time required before control rods enter the core if the tips lie outside the core. The response of the reactor protection system should allow for inoperable or out-of-service components and single failures.

2.2.2 BWR CRD Initial Conditions

2.2.2.1 Accident analyses should consider the full range of cycle operation from BOC to EOC.

2.2.2.2 Accident analyses at cold zero power conditions should encompass both BOC following core reload and restart following recent power operation.

2.2.2.3 Accident analyses should consider the full range of power operation including intermediate power levels up to hot full-power conditions. At conditions where certain core operating limits do not apply, the analysis should consider the potential for wider operating conditions as the result of plant maneuvering.

When properly justified, cycle-independent bounding evaluations that demonstrate that regions of power operation are less limiting are an acceptable analytical approach to reduce the number of cases analyzed. For example, credit for the rod worth minimizer system or void reactivity feedback during CRD scenarios initiated from at-power conditions may be used to demonstrate that these particular events are of less significance.

2.2.2.4 Uncontrolled worth for a dropped blade should be calculated based on the following conditions: (1) the range of control blade positions allowed at a given power level, (2) additional fully or partially inserted misaligned or inoperable blade(s) if allowed, and (3) any out-of-sequence control blades that may be inserted for fuel leaker power suppression. Applicants do not need to consider uncontrolled withdrawal (as the initiating event) of an inoperable blade that has been locked in place and cannot physically move. However, the impact of that inoperable blade on the worth of other blades needs to be considered.

Sufficient parametric studies should be performed to determine the worth of the limiting control blade for the allowed configurations highlighted above. The evaluation methodology should account for (1) calculation uncertainties in neutronic parameters (e.g., neutron cross sections) and (2) allowed power asymmetries.

Credit for additional control blade banking, such as from within the banked position withdrawal sequence (BPWS) or another similar banking scheme may be used to reduce the control blade reactivity worth during the event. The licensee’s reload analysis should fully reflect the required bank positions that were assumed in the control rod drop accident (CRDA) analysis and any additional control blade banking beyond the minimum required in the BPWS.

2.2.2.5 Because of burnup-dependent and corrosion-dependent factors that tend to reduce cladding failure thresholds and allowable limits on core coolability during fuel rod lifetime, the limiting initial conditions may involve locations other than the maximum uncontrolled blade worth defined in Regulatory Position 2.2.2.4 (e.g., uncontrolled blade motion at a core location adjacent to higher burnup fuel assemblies). For this reason, a more comprehensive search for the limiting conditions may be necessary to ensure that the total number of fuel rod failures is not underestimated, and allowable limits are satisfied. Applicants may need to survey a larger population of BWR blade drop core locations and exposure points to identify the limiting scenarios.

When properly justified, combining burnup-dependent parameters to create an artificial, composite worst time-in-life (e.g., end-of-life cladding hydrogen content combined with maximum ejected worth) is an acceptable analytical approach to reduce the number of cases analyzed.

2.2.2.6 The reactivity insertion rate should be determined from differential control blade worth curves and calculated transient blade position versus time curves.

2.2.2.7 Credit may be taken for the velocity limiter when determining the rate of withdrawal caused by gravitational forces.

2.2.2.8 The initial reactor coolant pressure, core inlet temperature, and flow rate used in the analysis should be conservatively chosen, depending on the transient phenomenon being investigated. The range of values should encompass the allowable operating range and monitoring uncertainties.

2.2.2.9 Fuel thermal properties (e.g., fuel-clad gap thermal conductivity, fuel thermal conductivity) should cover the full range over the fuel rod’s lifetime and should be conservatively selected based on the transient phenomenon being investigated. Time-in-life specific fuel properties may be used for a given burnup-specific statepoint analysis.

2.2.2.10 The moderator reactivity feedback resulting from voids, coolant pressure changes, and coolant temperature changes should be calculated based on the various assumed conditions of the fuel and moderator using standard transport and diffusion theory codes. If applicable, the range of values should encompass the allowable operating range (i.e., technical specifications in the core operating limits report) and any applicable analytical uncertainties.

2.2.2.11 Calculations of the Doppler reactivity feedback should be based on and compared with available experimental data. Since the Doppler feedback reflects the change in reactivity as a function of fuel temperature, uncertainties in predicting the coefficient, as well as in predicting fuel temperatures at different power levels, should be reflected by conservative application of Doppler feedback.

2.2.2.12 Control blade reactivity insertion during trip versus time should be obtained by combining the differential blade worth curve with a velocity curve based on maximum design limit values for scram insertion times. Alternatively, reactivity may be calculated using control blade velocity during trip based on maximum design limit values for scram insertion times. Any loss of available scram reactivity resulting from allowable rod insertion should be quantified.

2.2.2.13 The reactor trip delay time, or the amount of time that elapses between the instant the sensed parameter (e.g., pressure, neutron flux) reaches the level for which protective action is required and the onset of negative reactivity insertion, should be based on maximum values of the following: (1) time required for the instrument channel to produce a signal, (2) time for the trip breaker to open, (3) time for the control blade motion to initiate, and (4) time required before control blades enter the core if the tips lie outside the core. The response of the reactor protection system should include allowances for inoperable or out-of-service components and single failures.

2.3 Predicting the Total Number of Fuel Rod Failures

2.3.1 At each initial statepoint, the total number of failed rods that should be considered in the radiological assessment is equal to the sum of all fuel rods failing each of the cladding failure thresholds described in Regulatory Position C.3 of this guide. Applicants do not need to double-count fuel rods that are predicted to fail more than one of these thresholds.

2.3.2 Figure 1 provides an acceptable high-temperature cladding failure threshold as a function of cladding differential pressure. In the application of Figure 1, the cladding differential pressure should include both the initial, pretransient rod internal gas pressure plus any increase associated with transient FGR. An approved fuel rod thermal-mechanical performance code should be used to predict the initial, pretransient rod internal conditions (e.g., moles of fission gas, void volume, FGR, rod internal pressure). The amount of transient FGR may be calculated using the burnup-dependent correlations in Appendix B.

2.3.3 Because of the large variation in predicted radial average fuel enthalpy rise along the axial length of a fuel rod, the applicant may elect to (1) calculate transient FGR for selected axial regions and (2) combine each axial contribution, along with the pretransient gas inventory, within the calculation of total rod internal pressure.

2.3.4 In the application of the PCMI cladding failure thresholds, an NRC-approved alloy-specific cladding corrosion and hydrogen uptake model (or the appropriate model from Appendix C of this guidance) should be used to predict the initial, pretransient cladding hydrogen content. These approved models should account for the influence of (1) time at temperature (e.g., residence time, operating temperatures, steaming rate), (2) cladding fluence (e.g., dissolution of second-phase precipitates), (3) enhanced hydrogen uptake mechanisms (e.g., shadow corrosion, proximity to dissimilar metal), and (4) crud deposition, either directly or implicitly through the supporting database.

2.3.4.1 As an alternative, Appendix C presents acceptable alloy-specific hydrogen uptake models to estimate pretransient cladding hydrogen content.

2.3.4.2 The measured and estimated cladding hydrogen content in the empirical database used to develop the PCMI failure curves is based on total hydrogen content, including any hydrogen present in the oxide layer. Therefore, total hydrogen content should be used to implement these curves. If an applicant elects to use its own approved alloy-specific hydrogen model that separates out hydrogen in the oxide layer, then these curves would no longer apply.

2.3.4.3 The midwall cladding temperature at the start of the transient should be used to define the excess hydrogen in the cladding. Use of the Kearns solubility correlation (Ref. 12) is acceptable.

2.3.4.4 Because of the large variation in predicted radial average fuel enthalpy rise along the axial length of a fuel rod, along with potential axial variability in cladding hydrogen content, the applicant may need to perform multiple calculations to identify the limiting axial position. Alternatively, the PCMI cladding threshold corresponding to the predicted peak axial hydrogen content may be used to bound the entire fuel rod.

2.3.5 Because of the thermomechanical treatment of the cladding material under fabrication and its effect on the final cladding microstructure, zirconium hydride platelets will precipitate in a preferential orientation. For SRA cladding, most zirconium hydride platelets will precipitate in a circumferential orientation, whereas RXA cladding tends to exhibit randomly oriented zirconium hydride platelets. In addition to fabrication-related effects, the hydride orientation is also affected by the stress state prevailing during hydride precipitation (Ref. 12). Hydride reorientation from the circumferential direction to the radial direction is possible when the fuel rod cladding is loaded in tension beyond the hydride reorientation stress threshold (Refs. 13 and 14). Each applicant should address the possibility of hydride reorientation because of power maneuvering or reactor shutdown consistent with the criteria in NUREG-0800, Section 4.2.

2.3.6 Fuel cladding failure may occur almost instantaneously during the prompt fuel enthalpy rise (because of PCMI) or may occur as total fuel enthalpy (prompt plus delayed), heat flux, and cladding temperature increase. For the purpose of calculating fuel enthalpy for assessing PCMI failures, the prompt fuel enthalpy rise is defined as the radial average fuel enthalpy rise at the time corresponding to one pulse width after the peak of the prompt pulse. For assessing high-temperature cladding failures, the total radial average fuel enthalpy (prompt plus delayed) should be used.

2.4 Reactor Coolant System Pressure

The pressure surge should be calculated on the basis of conventional heat transfer from the fuel, a conservative metal-water reaction threshold, and prompt heat generation in the coolant to determine the variation of heat flux with time and the volume surge. The volume surge should then be used in the calculation of the pressure transient, accounting for fluid transport in the system, heat transfer to the steam generators, and the action of the pressurizer relief and safety valves, as appropriate. For PWRs, no accounting should be taken of the possible pressure reduction caused by the assumed failure of the control rod pressure housing on the RCS peak pressure, or other parameters for comparison to fuel failure thresholds.

3. Fuel Rod Cladding Failure Thresholds

Depending on the energy deposition level and the heat transfer from the rod, the following phenomena can occur: fuel temperatures increase and may approach melting temperatures (both rim and/or centerline), rapid fuel pellet thermal expansion may promote PCMI cladding failure, and local heat flux may exceed critical heat flux conditions, prompting fuel cladding temperatures to rise, leading to other potential fuel failure mechanisms.

The following sections define acceptable fuel rod cladding failure thresholds that encompass each degradation mechanism and failure mode. To ensure a conservative assessment of onsite and offsite radiological consequences, each of these failure modes should be quantified, and the sum total number of failed fuel rods should not be underestimated.

Conservative alternative fuel rod cladding failure criteria may be used if they are adequately justified by analytical methods and supported by sufficient experimental data. Alternative cladding failure criteria will be addressed on a case-by-case basis.

3.1 High-Temperature Cladding Failure Threshold

Figure 1 shows the empirically based high-temperature cladding failure threshold. This composite failure threshold encompasses both brittle and ductile failure modes and should be applied for events with prompt critical excursions (i.e. ejected rod worth or drop rod worth greater than or equal to $1.00). Because ductile failure depends on cladding temperature and differential pressure (i .e., rod internal pressure minus reactor pressure), the composite failure threshold is expressed in peak radial average fuel enthalpy (calories per gram (cal/g)) versus fuel cladding differential pressure (megapascals (MPa)).

For prompt critical scenarios which experience a prolonged power level following the prompt pulse, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., departure from nucleate boiling and critical power ratios).

For non-prompt critical excursions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits.

3.2 PCMI Cladding Failure Threshold

Figures 2 through 5 show the empirically based PCMI cladding failure thresholds. Because fuel cladding ductility is sensitive to hydrogen content, zirconium hydride orientation, and initial temperature, separate PCMI failure curves are provided for RXA and SRA cladding types at both low initial cladding temperature conditions (i.e., below 500 degrees F down to BWR cold startup) and high initial cladding temperature conditions (i.e., at or above 500 degrees F). The RXA cladding failure threshold is further refined for cladding designs with and without a barrier liner (e.g., sponge or low alloy cladding inside diameter liner). The SRA cladding failure threshold is applicable regardless of the presence of a barrier liner. The PCMI cladding failure threshold is expressed in peak radial average fuel enthalpy rise (Δcal/g) versus excess cladding hydrogen content (wppm). Excess cladding hydrogen content refers to the portion of total hydrogen content in the form of zirconium hydrides (i.e., it does not include hydrogen in solution).

3.3 Molten Fuel Cladding Failure Threshold

Fuel cladding failure is presumed if predicted fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions.

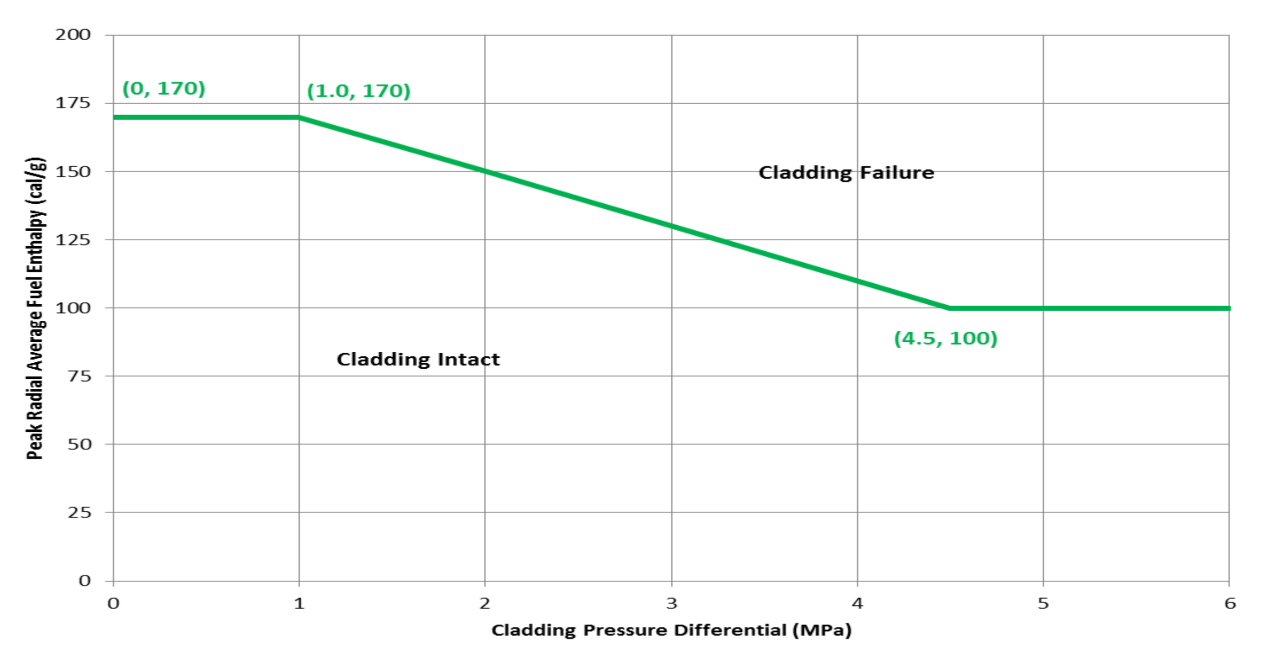

Figure 1. High-Temperature Cladding Failure Threshold

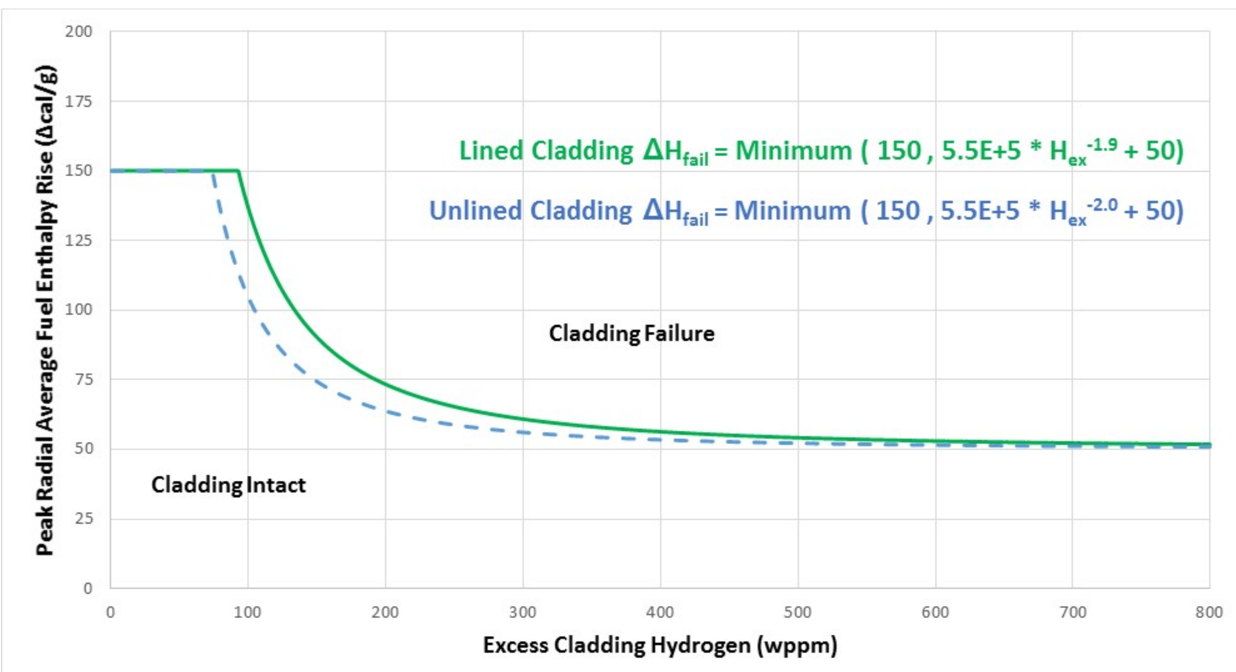

Figure 2. PCMI Cladding Failure Threshold—RXA Cladding at or above 500 Degrees F

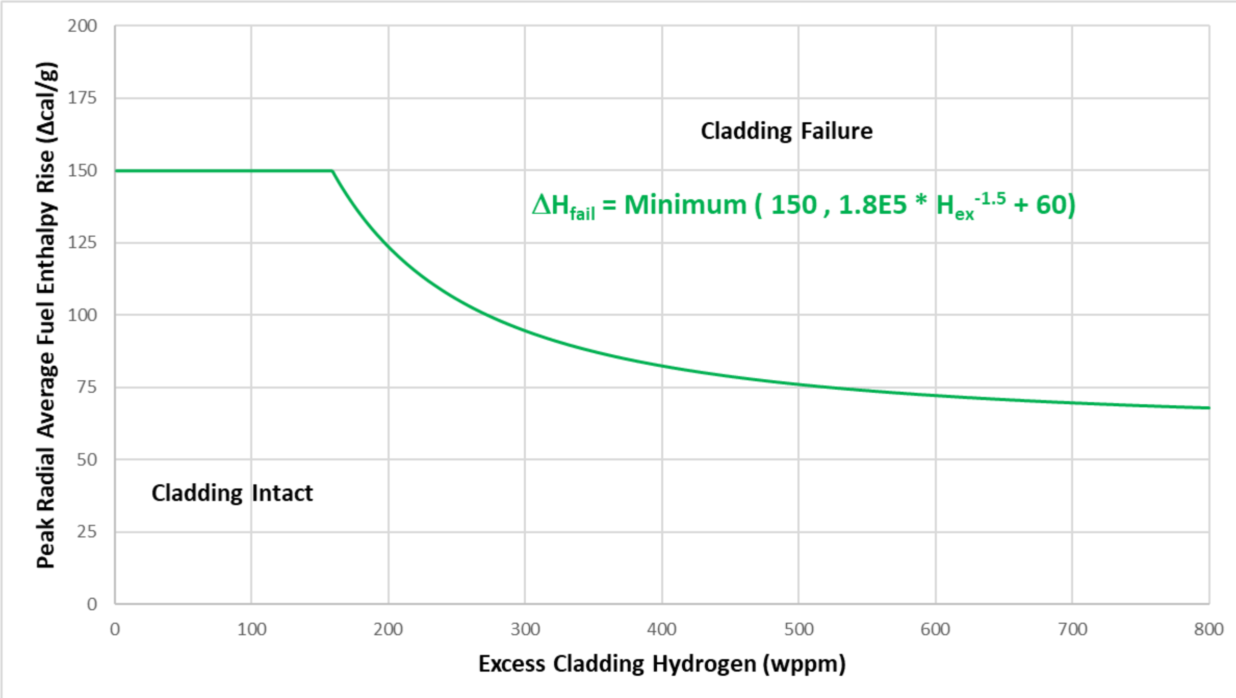

Figure 3. PCMI Cladding Failure Threshold—SRA Cladding at or above 500 Degrees F

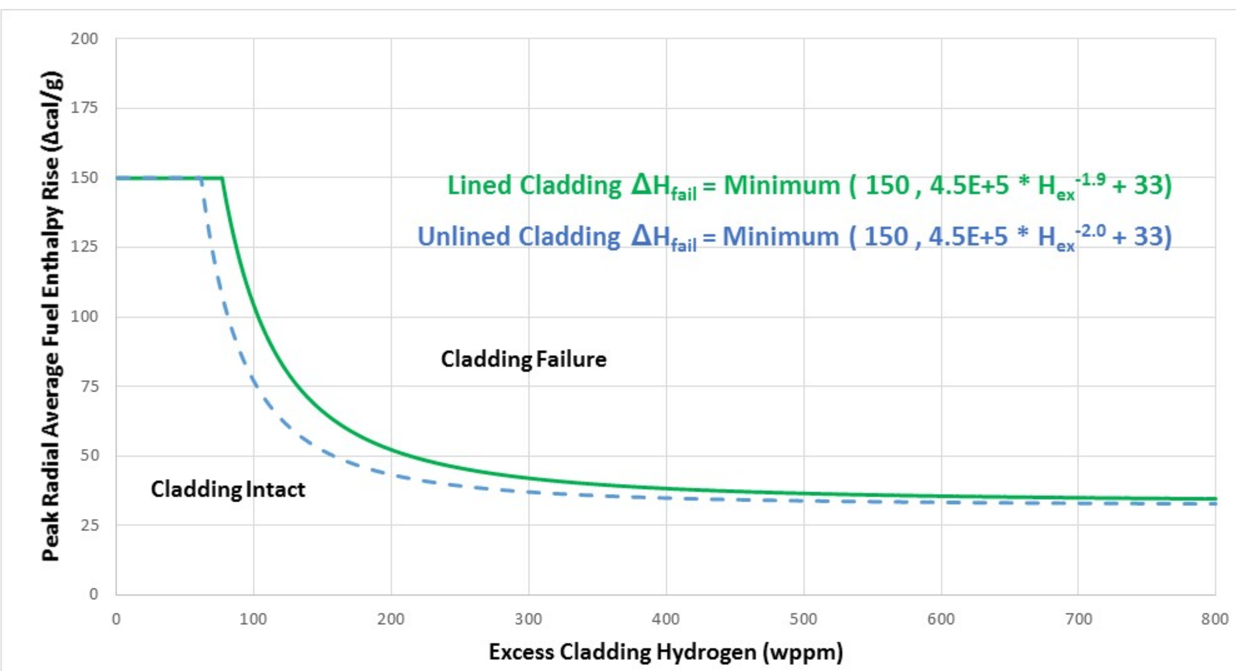

Figure 4. PCMI Cladding Failure Threshold—RXA Cladding below 500 Degrees F

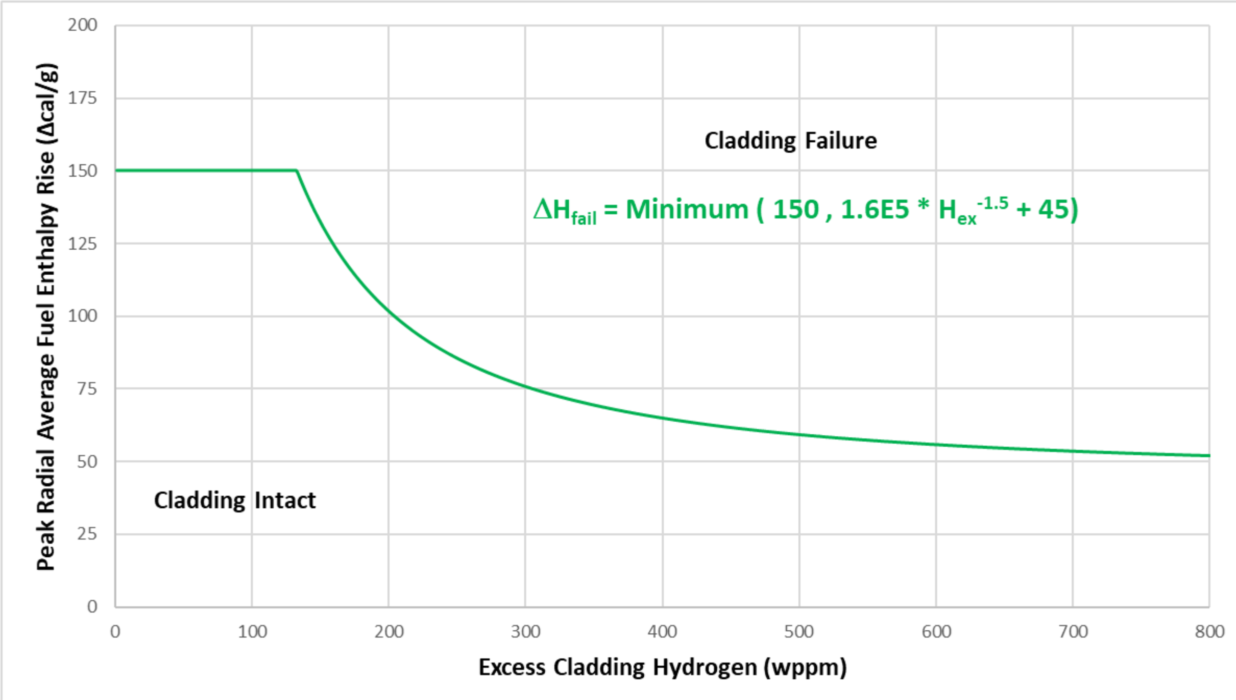

Figure 5. PCMI Cladding Failure Threshold—SRA Cladding below 500 Degrees F

4. Allowable Limits on Radiological Consequences

RG 1.183 and RG 1.195 contain the accident dose radiological consequences criteria for CRD and CRE accidents.

5. Allowable Limits on Reactor Coolant System Pressure

For new license applications, the maximum reactor coolant system pressure should be limited to the value that will prevent stresses from exceeding Emergency Condition (Service Level C), as defined in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 15). For existing plants, the allowable limits for the reactor pressure boundary specified in the plant's updated final safety analysis report should be maintained.

6. Allowable Limits on Damaged Core Coolability

Limiting peak radial average fuel enthalpy to prevent catastrophic fuel rod failure and avoiding molten fuel-coolant interaction is an acceptable metric to demonstrate that there is limited damage to core geometry and that the core remains amenable to cooling. The following restrictions should be met:

a. Peak radial average fuel enthalpy should remain below 230 cal/g.

b. A limited amount of fuel melting is acceptable provided that it is less than 10 percent of fuel volume. If fuel melting occurs, the peak fuel temperature in the outer 90 percent of the fuel volume should remain below incipient fuel melting conditions.

For fresh and low-burnup fuel rods, the peak radial average fuel enthalpy restriction will likely be more limiting than the limited fuel melt restriction. However, because of the effects of edge-peaked pellet radial power distribution and lower solidus temperature, medium- to high-burnup fuel rods are more likely to experience fuel melting in the pellet periphery under prompt power excursion conditions. For these medium- to high-burnup rods, fuel melting outside the centerline region should be precluded, and this restriction will likely be more limiting than the peak radial average fuel enthalpy restriction.

D. IMPLEMENTATION

The NRC staff may use this regulatory guide as a reference in its regulatory processes, such as licensing, inspection, or enforcement. However, the NRC staff does not intend to use the guidance in this regulatory guide to support NRC staff actions in a manner that would constitute backfitting as that term is defined in 10 CFR 50.109, "Backfitting," and as described in NRC Management Directive 8.4 (Ref. 16), "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes forward fitting as that term is defined and described in Management Directive 8.4. 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 file a backfitting or forward fitting appeal with the NRC in accordance with the process in Management Directive 8.4.

E. REFERENCES [1]

1. *U.S. Code of Federal Regulations* (CFR), Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Washington, DC.

2. CFR, Title 10, Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.

3. U.S. Nuclear Regulatory Commission (NRC), NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Washington, DC.

4. NRC, Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Washington, DC.

5. NRC, RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," Washington, DC.

6. NRC, RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Washington, DC.

7. NRC, RG 1.203, "Transient and Accident Analysis methods," Washington, DC.

8. NRC, Memorandum, Ralph Landry to Thomas Martin, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Interim Acceptance Criteria and Guidance," January 19, 2007 (ADAMS Accession No. ML070220400).

9. NRC, Memorandum, Paul M. Clifford to Timothy J. McGinty, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1," March 16, 2015 (ADAMS Accession No. ML14188C423).

10. International Atomic Agency (IAEA) Specific Safety Guide (SSG)-52, "Design of the Reactor Core for Nuclear Power Plants," 2019, Vienna, Austria.[2]

11. IAEA Technical Report Series No. 354, "Reactivity Accidents," 1993, Vienna, Austria.

12. Kearns, J.J., "Terminal Solubility and Partitioning of Hydrides in the Alpha Phase of Zirconium, Zircaloy-2 and Zircaloy-4," *Journal of Nuclear Materials*, 22:292–303, 1967.[3]

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[1] Publicly available NRC published documents are available electronically through the NRC Library 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 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.

[3] The Journal of Nuclear Materials is a publication of Elsevier Inc., 360 Park Avenue South, New York, NY 10010, telephone: 212-989-5800. Copies of Elsevier Books and Journals can be purchased at their web site: https://www.elsevier.com/books-and-journals.

13. Organization for Economic Cooperation and Development, Nuclear Energy Agency, State-of-the-Art Report, "Nuclear Fuel Behaviour under Reactivity-Initiated Accident (RIA) Conditions," ISBN 978-92-99113-2, 2010.[4]

14. Billone, M.C., T. Burtseva, Z. Han, and Y. Liu, "Embrittlement and DBTT of High-Burnup PWR Fuel Cladding Alloys," Used Fuel Disposition Campaign, Argonne National Laboratory, FCRD-UFD-2013-000401, ANL-13/16, September 30, 2013.[5]

15. American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code*, Section III, Division 1, "Rules for Construction of Nuclear Facility Components," New York, NY.[6]

16. NRC, Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," Washington, DC.

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4 Publically available on OECD-NEA Web site: https://www.oecd-nea.org/nsd/reports/2010/nea6847-behaviour-RIA.pdf.

5 Publically available on U.S. Department of Energy Web site: https://www.energy.gov/sites/prod/files/2013/12/f5/EmbrittlementDBTTHighBrnup%20PWRFuelClad%20Alloys.pdf

6 Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Two Park Avenue, New York, NY 10016-5990; telephone: 800-843-2763. Purchase information is available through the ASME Web site store at http://www.asme.org/Codes/Publications/.

APPENDIX A

ACRONYMS AND ABBREVIATIONS

ADAMS	Agencywide Documents Access and Management System
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
BIGR	Fast Pulse Graphite Reactor
BOC	beginning of cycle
BPWS	bank position withdrawal sequence
BU	burnup
BWR	boiling-water reactor
cal/g	calories per gram
CRDM	control rod drive mechanism
CRD	control rod (blade) drop
CRE	control rod ejection
Cs	Cesium
EOC	end of cycle
FGR	fission gas release
GDC	general design criteria
GWd/MTU	gigawatt-days per metric ton of uranium
I	Iodine
IGR	Impulse Graphite Reactor
Kr	Krypton
kW/ft	kilowatts per foot
LOCA	Loss-of-coolant accident

LWR	light-water reactor
MPa	Megapascal
MOX	Mixed Oxide
NSRR	Nuclear Safety Research Reactor
OMB	Office of Management and Budget
PBF	Power Burst Facility
PCMI	pellet-clad mechanical interaction
PNNL	Pacific Northwest National Laboratory
PWR	pressurized-water reactor
R/B	release to birth
RXA	recrystallized annealed
SPERT	Special Power Excursion Test Reactor
SRA	stress relief annealed
SRP	Standard Review Plan (NUREG-0800)
Te	Tellurium
TREAT	Transient Reactor Test Facility
UO_2	uranium dioxide
wppm	weight parts per million
Xe	Xenon
βeff	effective delayed neutron fraction
ΔH	change in radial average fuel enthalpy
$\Delta\rho$	change in reactivity
l*	average neutron lifetime

APPENDIX B

TRANSIENT FISSION GAS RELEASE

This appendix provides guidance on transient fission gas release (FGR). Unlike steady-state FGR (into the rod plenum) which is controlled by diffusion during normal operations, pellet fracturing and grain boundary separation are the primary mechanisms for FGR during the transient. The release of additional fission gas during the transient will contribute to higher rod internal pressure.

The empirical database suggests that transient FGR is sensitive to both local fuel burnup and peak radial average fuel enthalpy rise. As a result, separate low-burnup and high-burnup transient FGR correlations, shown in Figure B-1, are provided as a function of peak radial average fuel enthalpy rise (Δ calories per gram (cal/g)). NRC memorandum titled "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1," dated March 16, 2015 (Ref. B-1), documents the derivation of the transient FGR correlations, including the application of uncertainties.

pellet burnup (BU) < 50 gigawatt-days per metric ton of uranium (GWd/MTU)
transient FGR = maximum [(0.26 * ΔH) – 13) / 100, 0]

pellet BU ≥ 50 GWd/MTU
transient FGR = maximum [(0.26 * ΔH) – 5) / 100, 0]

where:
FGR = fission gas release, fraction
ΔH = increase in radial average fuel enthalpy, Δcal/g

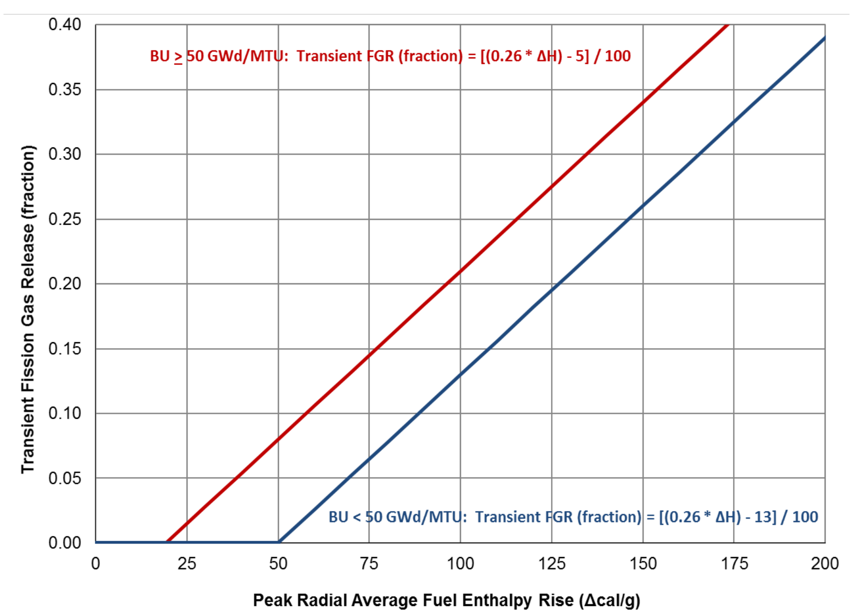

Figure B-1. Transient Fission Gas Release

REFERENCES[7]

B-1 NRC, Memorandum, Paul M. Clifford to Timothy J. McGinty, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1," March 16, 2015 (ADAMS Accession No. ML14188C423).

7 Publicly available NRC published documents are available electronically through the NRC Library 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.

APPENDIX C

HYDROGEN UPTAKE MODELS FOR FUEL ROD CLADDING

The purpose of this appendix is to provide acceptable fuel rod cladding hydrogen uptake models for commercial zirconium alloys to aid in the implementation of threshold curves for hydrogen-dependent, pellet-clad mechanical interaction cladding failure. These models are also acceptable for implementing other hydrogen-dependent fuel performance requirements (e.g., emergency core cooling system) analytical limits on peak cladding temperature and integral time-at-temperature (expressed as equivalent cladding reacted and calculated using the Cathcart-Pawel correlation [Ref. C-1]) as a function of pretransient cladding hydrogen content.

C-1. Zirconium Cladding Alloys in Pressurized-Water Reactors

Corrosion rates and the amount of corrosion at fuel discharge vary widely across the pressurized-water reactor (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 can predict a best estimate corrosion thickness as a function of effective full-power days and local operating conditions (fuel duty).

An examination of the empirical database of measured cladding hydrogen content for the current commercial zirconium alloys reveals that PWR cladding alloys do not exhibit the same breakaway hydrogen uptake at higher fluence levels as observed in Zircaloy-2 data for boiling-water reactors (BWRs). However, the pickup fraction does appear to be alloy specific. With consideration of the extent, uncertainty, and variability of the supporting database, the staff developed the following upper bound pickup fractions:

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.

C-2. Zircaloy-2 Cladding in Boiling-Water Reactors

An examination of the empirical database of measured cladding hydrogen content for legacy and modern commercial BWR Zircaloy-2 cladding alloys reveals that a constant hydrogen pickup fraction does not fit the observed cladding hydrogen data. Given the allowable range in composition within the ASTM specification for Zircaloy-2 (ASTM B351/B351M, “Standard Specification for Hot-Rolled and Cold-Finished Zirconium and Zirconium Alloy Bars, Rod, and Wire for Nuclear Application” [Ref. C-2]) and the degree of flexibility and variability in manufacturing procedures, the staff chose to adopt the more conservative legacy hydrogen uptake model.

An acceptable burnup-dependent 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, weight parts per million

BU = local axial burnup, gigawatt-days per metric ton of uranium (GWd/MTU)

C-3. Applicability

The hydrogen models apply to 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.

REFERENCES[8]

C-1 ORNL/NUREG-17, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies," August 1977.

C-2 American Society of 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," Revision 11, 2016.[9]

8 Publicly available NRC published documents are available electronically through the NRC Library 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

9 Copies of American Society of Testing and Materials (ASTM) standards and purchase information are available through the ASTM Web site store at http://www.astm.org/Standards/standards-and-publications.html.