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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 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 defines fuel cladding failure thresholds for ductile failure, brittle failure, and pellet-clad mechanical interaction (PCMI) and provides an algorithm for calculating radionuclide release fractions for use in assessing radiological consequences. It also describes analytical limits and guidance for demonstrating compliance with applicable regulations governing reactivity limits.

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.
    - o 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (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 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,

This RG is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not received final staff review or approval and does not represent an NRC final staff position. Public comments are being solicited on this draft guide and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal-rulemaking Web site, http://www.regulations.gov, by searching for Docket ID: NRC-2016-0233. Alternatively, comments may be submitted to the Rules, Announcements, and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Comments must be submitted by the date indicated in the *Federal Register* notice.

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steamline rupture, changes in reactor coolant temperature and pressure, and cold water addition. Plants whose construction permits were issued before 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.

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 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 incorporates empirical data from in-pile, prompt power pulse test programs and analyses from several international publications on fuel rod performance under prompt power excursion conditions to provide guidance on acceptable analytical methods, assumptions, and limits for evaluating a postulated PWR CRE and a postulated BWR CRD accident. To assist with the implementation of this guidance, Appendix B provides acceptable fission product release fractions and Appendix C provides acceptable cladding hydrogen uptake models.

Background

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 has 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 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. 7). 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. 8). This memorandum, as amended by public comments (Ref. 9), documents the empirical database, as well as the technical and regulatory bases for this guide. To reflect the latest state of knowledge, this guide presents that information.

A PWR CRE event is postulated to occur because of a mechanical failure that causes an instantaneous circumferential rupture of the control element drive mechanism (CEDM) housing or its associated nozzle. This results in the reactor coolant system pressure ejecting the control rod and drive shaft to the fully withdrawn position. The CEDM housings are capable of withstanding throughout their design life all normal operating loads, including the steady-state and transient operating conditions specified for the reactor vessel. Hence, such a failure is considered to be a very low probability event.

A BWR CRD event is postulated to occur because of the following sequence of events: a control rod (blade) inserted into the core becomes decoupled from its drive mechanism, the drive mechanism is subsequently withdrawn, the control blade is assumed to be stuck in place, and later, the control rod suddenly falls free and drops to the control rod drive position. This results in the removal of large negative reactivity from the core and a localized power excursion. By postulating the worst possible combination of rod worth and core conditions, this accident encompasses the consequences of all such reactivity control system excursions.

The uncontrolled movement of a single control rod out of the core results 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, which 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.

GDC 28 of 10 CFR Part 50, Appendix A, requires reactivity control systems to be designed with appropriate limits on potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than local yielding nor sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to significantly impair 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, steamline rupture, changes in reactor coolant temperature and pressure, and cold water addition. Reactivity insertion accidents, such as PWR CRE and BWR CRD, directly affect the core by challenging the array geometry of fuel rod bundles. Rapid local power excursions may cause gross failure of fuel rods and loss of a coolable core geometry. Furthermore, molten fuel ejected from failed rods will interact with the reactor coolant, producing a pressure pulse that may challenge the integrity of the reactor pressure boundary.

Harmonization with International Standards

The NRC staff reviewed guidance from the International Atomic Energy Agency, International Organization for Standardization, and International Electrotechnical Commission and did not identify any standards that provided useful guidance to NRC staff, applicants, or licensees.

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 may not directly apply to anticipated operational occurrences and other postulated accidents involving positive reactivity insertion (e.g., PWR excess load, PWR inadvertent bank withdrawal, PWR steamline 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 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.

1.2 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.

1.2.3 Because of the dominant role liner fuel test results played in the development of the RXA PCMI cladding failure threshold curves and the influence of the sponge or low-alloy zirconium liner on the initial hydride distribution, the applicability of these failure threshold curves for nonliner cladding designs is limited to cladding with less than 70 weight parts per million (wppm) excess hydrogen.

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.

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 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 The maximum uncontrolled worth of an ejected rod should be calculated based on fully or partially inserted misaligned or inoperable rod or rods if allowed. Sufficient parametric studies should be performed to determine the worth of the most reactive control rod of all inserted control rods 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.2.2.4 (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 coefficients 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 coefficient of reactivity 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 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 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, 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 The maximum 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 blades if allowed, and (3) any out-of-sequence control blades that may be inserted for fuel leaker power suppression. Sufficient parametric studies should be performed to determine the worth of the most reactive control blade of all inserted control rods 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 within the bank position withdrawal sequence (BPWS) may be used to reduce the control blade reactivity worth during the event. The licensee's reload analysis should fully reflect 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 coefficients 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 coefficient of reactivity 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 fission gas release (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 Figure B-2.

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 several 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 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. 10) 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. 11). 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. 11 and 12). Each applicant should address the possibility of hydride reorientation because of power maneuvering or reactor shutdown.

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.

2.3.7 For plants in which gross failure (sufficient to allow a control rod to be ejected rapidly from the core) of a control rod drive mechanism housing is not considered credible, fuel failure predictions do not need to consider any reactor coolant system depressurization resulting from a mechanistic evaluation of a ruptured control rod drive mechanism housing. If credible, it should be shown that failure of one control rod housing will not lead to failure of other control rod housings.

2.4 Fission Product Release Fractions

Because of the large variation in predicted fuel radial average enthalpy rise along the axial length of a fuel rod, the applicant may elect to (1) calculate the transient fission product release fraction for each radionuclide for several axial regions and (2) combine each axial contribution, along with the pretransient, steady-state inventories, to obtain the total radiological source term for dose calculations. Appendix B gives more information and guidance.

2.5 Reactor Coolant System Peak Pressure

For PWRs, 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. No credit should be taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

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 and bounding 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 initiated from reactor startup conditions up to 5-percent reactor power operating conditions. Because ductile failure depends on both 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 at-power operating conditions (i.e., above 5-percent reactor power), fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., departure from nucleate boiling and critical power ratios).

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 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).

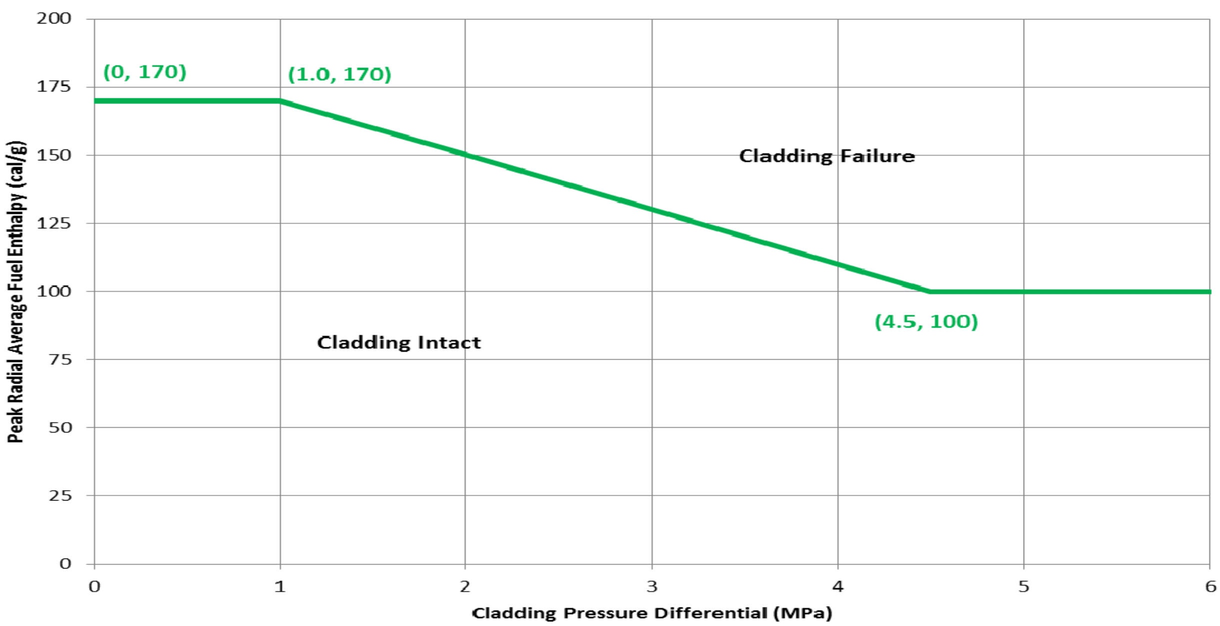

Figure 1. High-Temperature Cladding Failure Threshold

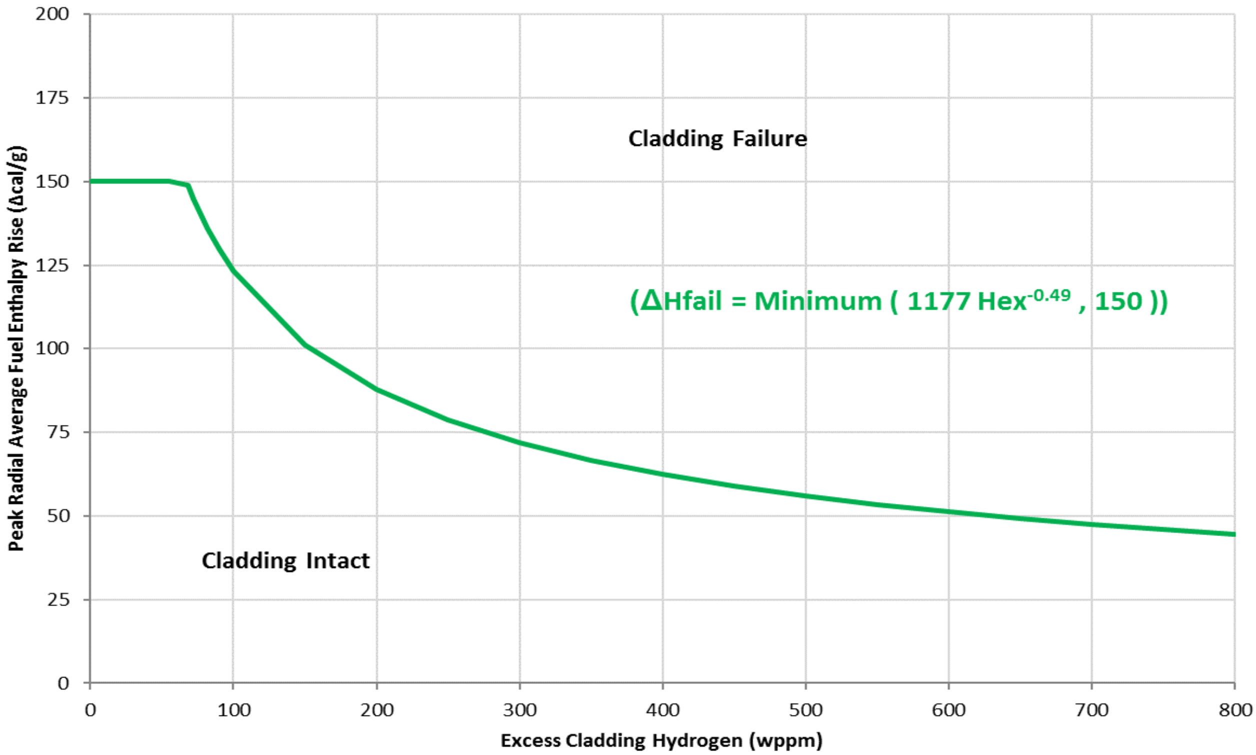

Figure 2. PCMI Cladding Failure Threshold—RXA Cladding at or above 500 Degrees F

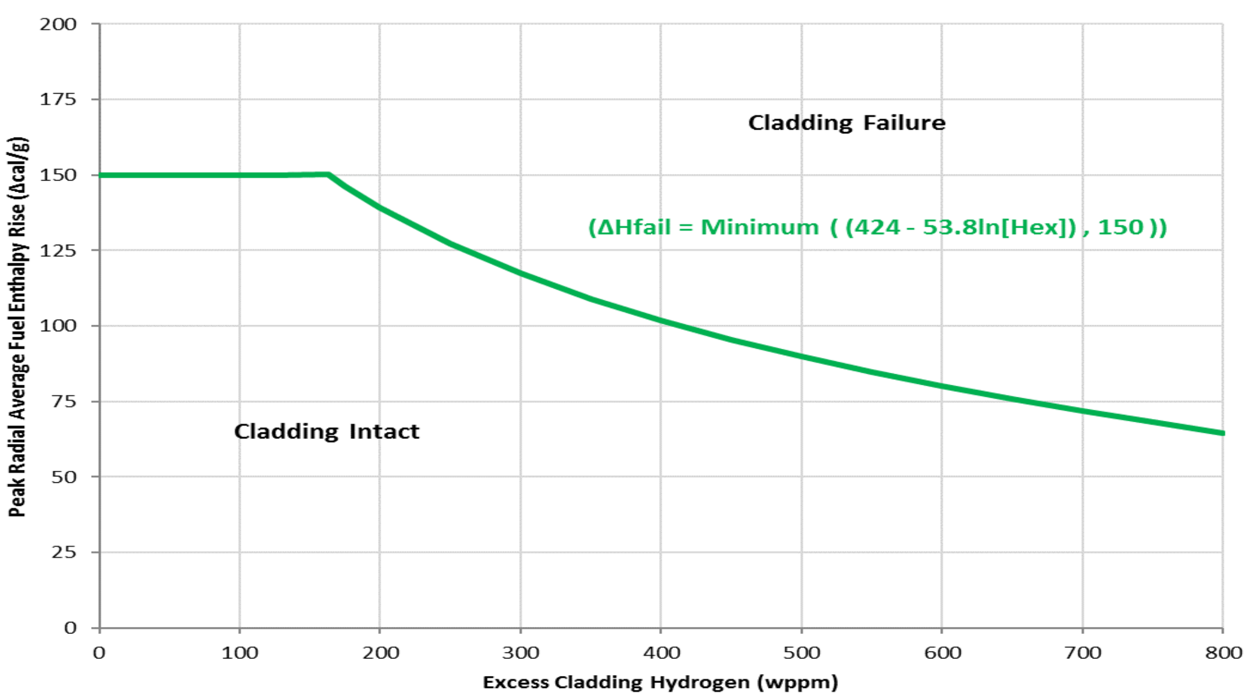

Figure 3. PCMI Cladding Failure Threshold—SRA Cladding at or above 500 Degrees F

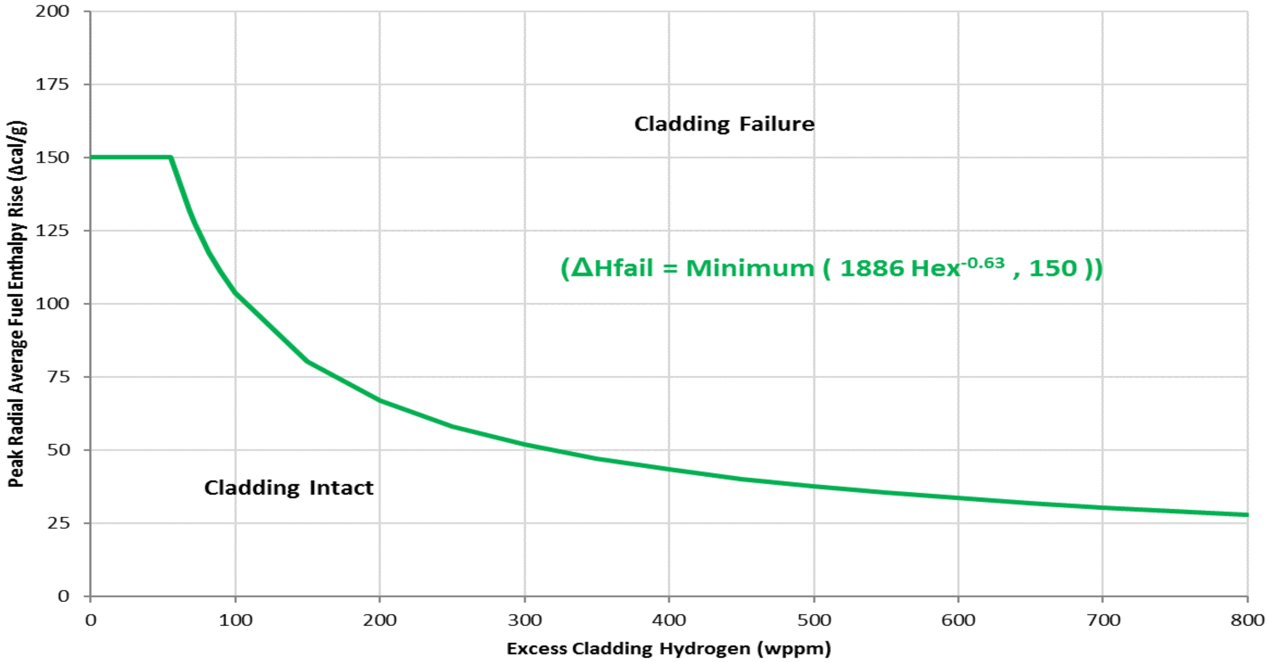

Figure 4. PCMI Cladding Failure Threshold—RXA Cladding below 500 Degrees F

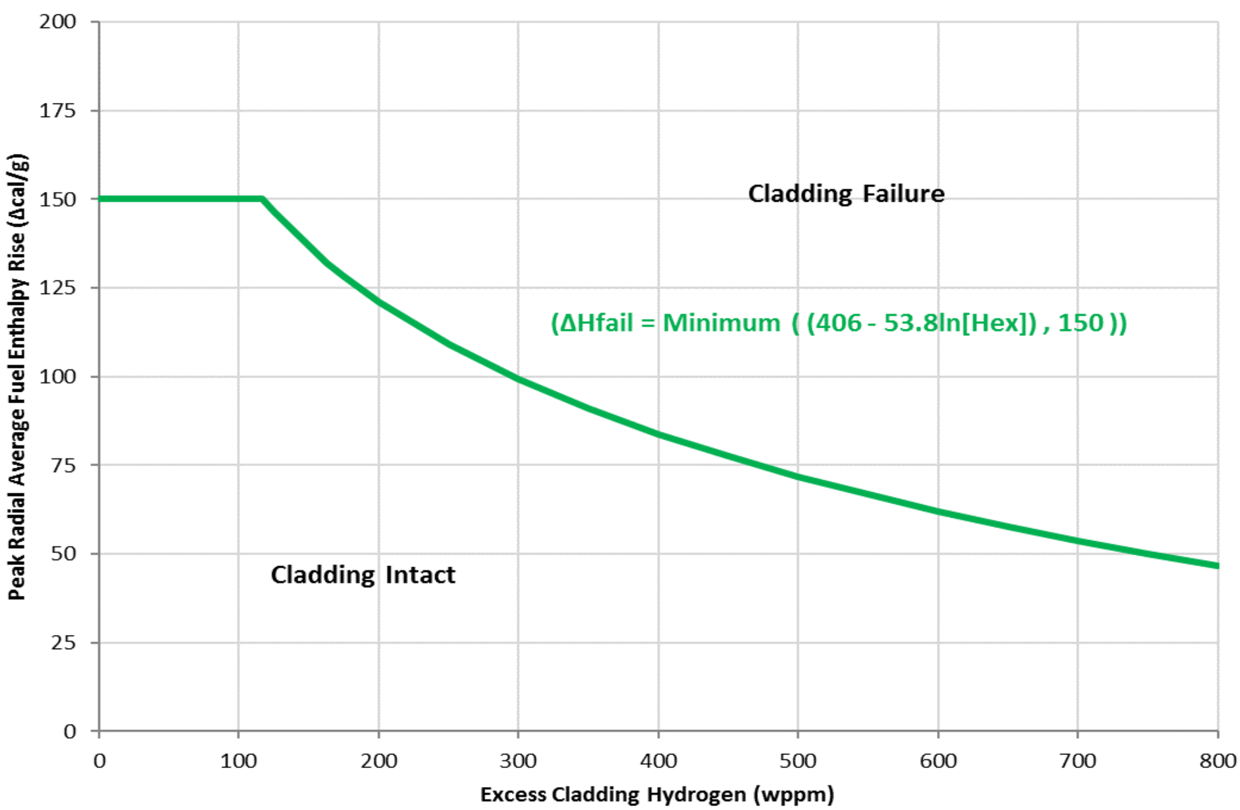

Figure 5. PCMI Cladding Failure Threshold—SRA Cladding below 500 Degrees F

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.

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. 13). 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 centerline 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 purpose of this section is to provide information on how applicants and licensees[1] 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[2] 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 without further backfit consideration. 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 communications, or a rule requiring the use of this RG.

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 license for a facility, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this regulatory guide constitutes a violation.

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

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1 In this section, "licensees" refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52; and the term "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.

2 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.

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. 14), and NUREG-1409, "Backfitting Guidelines" (Ref. 15).

REFERENCES [3]

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, Memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Interim Acceptance Criteria and Guidance," January 19, 2007 (ADAMS Accession No. ML070220400).

8. NRC, Memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1," March 16, 2015 (ADAMS Accession No. ML14188C423).

9. NRC, Memorandum, "DG-1327 Comment Resolution Table," September 6, 2018 (ADAMS Accession No. ML18249A346).

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

11. 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.[5]

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[3] 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.

[4] Elsevier Inc., 360 Park Avenue South, New York, NY 10010, telephone: 212-989-5800.

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

12. Billone, M., T. Burtseva, and Y. Liu, "Baseline Properties and DBTT of High Burnup PWR Cladding Alloys," Proceedings of the 17th International Symposium on the Packaging and Transportation of Radioactive Materials," 2013.[6]

13. 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.[7]

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

15. NRC, NUREG-1409, "Backfitting Guidelines," Washington, DC.

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[6] Institute of Nuclear Materials Management, One Parkview Plaza, Suite 800, Oakbrook Terrace, IL 60181; telephone: 847-686-2236

[7] 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
CEDM	control element drive mechanism
CFR	Code of Federal Regulations
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

FISSION PRODUCT RELEASE FRACTIONS

This appendix provides guidance on steady-state and transient gap fission product inventories available for release following a control rod ejection (CRE) or control rod (blade) drop (CRD) accident. The generic steady-state gap fractions and analytical approach for calculating steady-state gap fractions are acceptable for use in defining the initial radionuclide inventory for all non-loss-of-coolant (non-LOCA) design-basis accidents. The fission product release fraction guidance contained in Appendix B for the CRE and CRD accidents should be used instead of the gap fractions provided in RG 1.183, Revision 0, for a CRE and CRD accident until RG 1.183 is updated.

The total fractions of fission products available for release from a failed fuel rod are equal to the sum of the steady-state fission product gap inventory and 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. For fuel that melts, the combined fission product inventory (steady-state gap plus transient release) is added to the release resulting from fuel melting. RG 1.183 (Ref. B-1) and 1.195 (Ref. B-2) provide additional guidance on fuel melt source term.

The steady-state fission product gap inventories and transient fission gas release fractions are applicable up to a fuel rod average burnup of 65 GWd/MTU.

Steady-State Fission Product Gap Inventory

Table B-1 gives the fractions of the core inventory for various radionuclides assumed to be in the gap for a fuel rod during steady-state operation. The gap fractions from Table B-1 are used in conjunction with the calculated fission product inventory calculated with the maximum core radial peaking factor. The applicability of Table B-1 is limited to uranium dioxide (UO_2) fuel rods with a peak rod average power history below the bounding power envelope depicted in Figure B-1.

Table B-1. Steady-State Fission Product Inventory in Gap

Group	Fraction
I-131	0.08
I-132	0.06
Kr-85	0.36
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.49[B1]

The U.S. Nuclear Regulatory Commission (NRC) memorandum titled "Revised Technical Basis for Fission Product Release Fractions," dated June 4, 2019 (Ref. B-3), documents the derivation of the steady-state gap fractions, including the application of uncertainties. Pacific Northwest National Laboratory (PNNL) Report 18212, Revision 1, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard," issued June 2011 (Ref. B-4), documents an acceptable

[B1] The fraction of Cesium in the gap provided in Table B-1 assumes a diffusion coefficient from the fuel to the gap that is 2 times the coefficient for Krypton (Kr)-85. Chemical changes that accompany extended levels of reactor fuel burnup may retard the transport of fission products such as cesium in the fuel grains to the fuel-cladding gap region. The staff may consider credit for retention of cesium in the fuel-cladding region on a case-by-case basis.

analytical method for calculating steady-state gap fractions. As an alternative to the above gap fractions, a licensee may use this analytical technique, described in the attachment, to calculate steady-state fission product gap inventories based on specific fuel rod designs or more realistic fuel rod power histories.

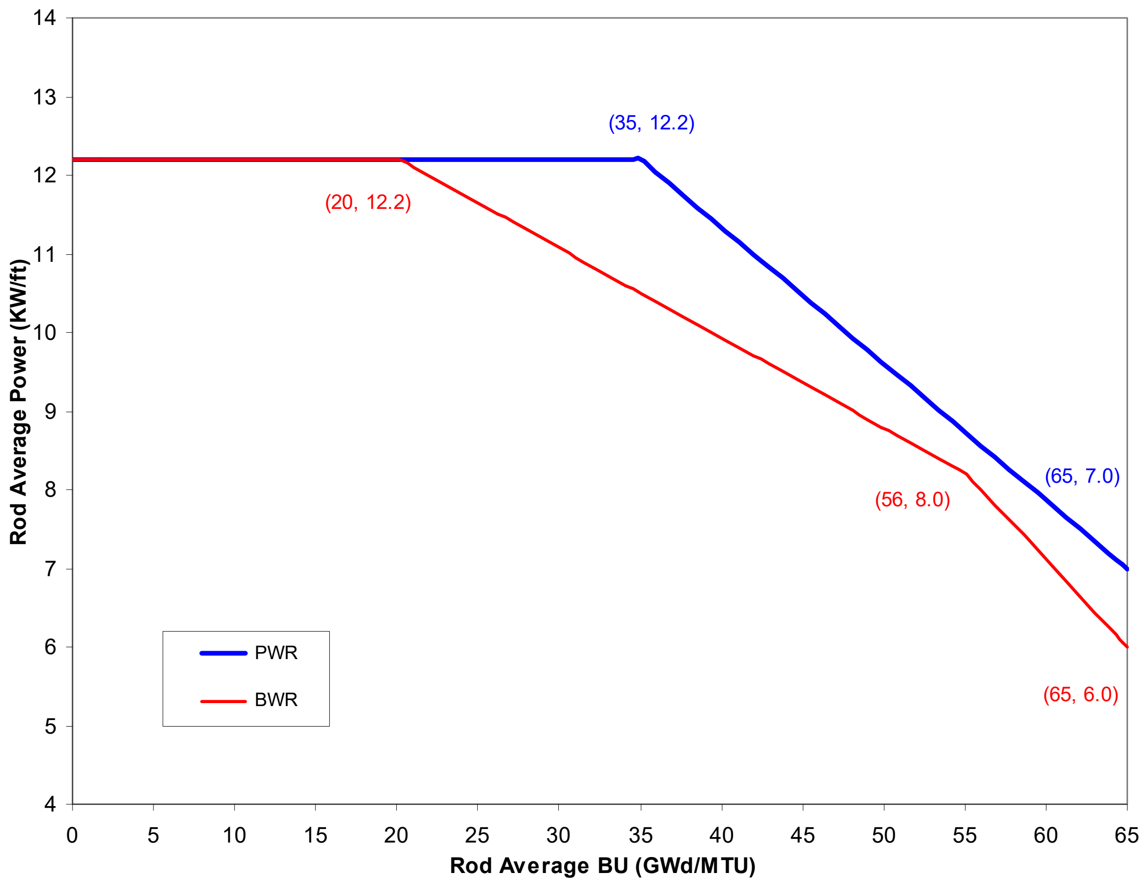

Figure B-1. Maximum Allowable Power Operating Envelope for Non-LOCA Gap Fractions

Transient Fission Gas Release

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-2, 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-5), 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 [$(0.26 * \Delta H) - 13) / 100, 0$]

pellet BU $\geq$ 50 GWd/MTU

transient FGR = maximum [$(0.26 * \Delta H) - 5) / 100, 0$]

where:

FGR = fission gas release, fraction

ΔH = increase in radial average fuel enthalpy, Δcal/g

An investigation into the effect of differences in diffusion coefficients and radioactive decay on fission product transient release (Ref. B-4) concluded that adjustments to the empirically based correlations are needed for different radionuclides.

B.1 For stable, long-lived isotopes (e.g., Kr-85), the transient fission product release is equivalent to the burnup-dependent correlations.

B.2 For alkali metals, such as cesium (Cs)-134 and Cs-137, the transient fission product release correlations should be multiplied by a factor of 1.414.

B.3 For volatile, short-lived radioactive isotopes such as halogens (e.g., iodine (I)-131, I-132, I-133, I-135) and xenon (Xe) and krypton noble gases except Kr-85 (e.g., Xe-133, Xe-135, Kr-85m, Kr-87, Kr-88), the transient fission product release correlations should be multiplied by a factor of 0.333.

<u>Total Fission Product Inventory Available for Release</u>

The transient fission product release fractions should be added to the steady-state fission product gap inventory for each radionuclide (present before the event) to obtain the total fission product inventory available for release upon cladding failure. The sum total of combined fission product inventories from each fuel rod predicted to experience cladding failure (all failure modes) should be used in the dose assessment. If localized fuel melting is predicted during the postulated accident, additional fission product releases will need to be included in the total radiological source term.

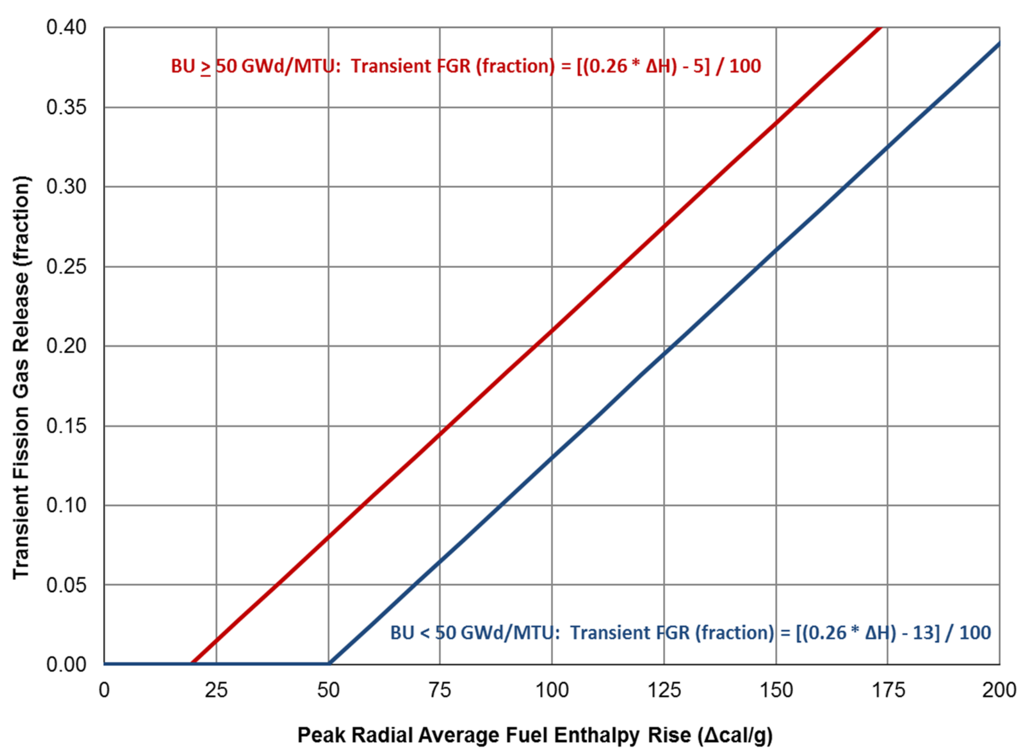

Figure B-2. Transient Fission Gas Release

ANALYTICAL TECHNIQUE FOR CALCULATING STEADY-STATE FISSION PRODUCT GAP INVENTORIES

This section provides an acceptable analytical technique for calculating steady-state fission product gap inventories based on specific fuel rod designs or more realistic fuel rod power histories. Following the analytical technique detailed below, bounding gap inventories were developed based on bounding fuel rod power histories and limiting fuel rod designs. Table B-1 lists these bounding (maximum) gap inventories. Lower gap fractions are achievable using less aggressive rod power histories or less limiting fuel rod designs or both (e.g., 17x17 versus 14x14 fuel rod design). Alternatively, applicants may use the bounding gap inventories provided in Table B-1 if they meet the peak radial average power envelope in Figure B-1.

Steady-state gap inventories represent radioactive fission products generated during normal steady-state operation that have diffused within the fuel pellet, have been released into the fuel rod void space (i.e., rod plenum and pellet-to-cladding gap), and are available for release upon fuel rod cladding failure. Given the continued accumulation of long-lived radioactive isotopes and the inevitable decay of short-lived radioactive isotopes, the most limiting time-in-life (i.e., maximum gap fraction) for a particular radioactive isotope varies with fuel rod exposure and power history. The analytical technique described in this attachment prescribes the use of bounding fuel rod power profiles based on core operating limits or limiting fuel rod power histories. In addition, this analytical technique produces a composite worst time-in-life (i.e., maximum gap fraction for each radioactive isotope). Therefore, the steady-state fission product gap inventories calculated using this analytical approach will be significantly larger than realistic fuel rod or core-average source terms.

The NRC maintains the FAST (formerly FRAPCON / FRAPTRAN) fuel rod thermal-mechanical fuel performance code to perform independent audit calculations for licensing activities. While calibrated and validated against a large empirical database, FAST and its predecessors are not NRC-approved codes and may not be utilized to calculate that plant-, fuel-, or cycle-specific gap inventories are in accordance with the acceptable analytical procedure below without further justification.

The analytical technique used to calculate steady-state gap inventories should include the following attributes:

B-1. For stable, long-lived radioactive isotopes, such as krypton (Kr)-85, cesium (Cs)-134, and Cs-137, an NRC-approved fuel rod thermal-mechanical performance code with established modeling uncertainties should be used to predict the integral FGR. The code should include the effects of thermal conductivity degradation with burnup and should have been verified against measured fuel temperatures and stable FGR data up to the licensed burnup of the particular fuel rod design.

B-1.1 Long-lived radioactive isotopes will continue to accumulate throughout exposure, with insignificant decay because of their long half-lives. For this reason, maximum gap inventories for long-lived isotopes are likely to occur near or at the end of life of the fuel assembly.

B-1.2 As recommended by American Nuclear Society (ANS) Standard ANS-5.4-2011, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuels" (Ref. B-6), the cesium diffusion coefficient should be assumed to be a factor of 2.0 higher than for the noble gas

nuclides. Because the release fraction is approximately proportional to the square root of the diffusion coefficient, the cesium release fraction equals the following:

$$(\text{gap inventory})_{Cs\text{-}134,\ Cs\text{-}137} = (\text{gap inventory})_{Kr\text{-}85} * (2.0)^{0.5}$$

where $(\text{gap inventory})_{Kr\text{-}85}$ is calculated using an approved fuel performance code.

B-2. For volatile, short-lived radioactive isotopes, such as iodine (I) (i.e., I-131, I-132, I-133, and I-135) and xenon (Xe) and Kr noble gases (except for Kr-85) (i.e., Xe-133, Xe-135, Kr-85m, Kr-87, and Kr-88), an NRC-approved release model or an NRC-endorsed ANS-5.4 release model should be used to predict the release-to-birth (R/B) fraction using fuel parameters at several depletion time steps from an NRC-approved fuel rod thermal-mechanical performance code. The fuel parameters necessary for use in the NRC-endorsed ANS-5.4 model calculations of the R/B fraction are local fuel temperature, fission rate, and axial node/pellet burnup. Consistent with Item B-1, the code should include the effects of thermal conductivity degradation with burnup and should have been verified against measured fuel temperatures and stable FGR data up to the licensed burnup of the particular fuel rod design.

Because of their relatively short half-lives, the amount of activity associated with volatile radioactive isotopes depends on their rate of production (i.e., fission rate and cumulative yield), rate of release, and rate of decay. Maximum (R/B) ratios for short-lived isotopes are likely to occur at approximately the maximum exposure at the highest power level (i.e., knee in the power operating envelope).

B-2.1 NUREG/CR-7003 (PNNL-18490), “Background and Derivation of ANS-5.4 Standard Fission Product Release Model,” issued January 2010 (Ref. B-7), provides guidance on using the NRC-endorsed ANS-5.4 release model to calculate short-lived (R/B) factors.

B-2.1.1 For nuclides with half-lives of less than 1 hour, no gap inventories are provided. Because of their rapid decay (relative to the time for diffusion and transport), these nuclides will be bounded by the calculated gap fractions for longer lived nuclides under the headings “Other Noble Gases” and “Other Halogens” as shown in the example below.

For nuclides with half-lives of less than 6 hours, an approved fuel performance code is applied to predict the R/B fraction using Equation 12 in NUREG/CR-7003 and its definitions of terms, as follows:

$$\left(\frac{R}{B}\right)_{i,m} = \left(\frac{S}{V}\right)_{i,m} \sqrt{\frac{\alpha_{nuclide} D_{i,m}}{\lambda_{nuclide}}}$$

For nuclides with half-lives of greater than 6 hours, the R/B fraction is predicted by multiplying the fractal-scaling factor ($F_{nuclide}$) by the predicted Kr-85m (R/B) using Equation 13 of NUREG/CR-7003, as follows:

$$\left(\frac{R}{B}\right)_{i,nuclide} = F_{nuclide} \left(\frac{S}{V}\right)_{i} \sqrt{\frac{\alpha_{Kr-85m} D_{i}}{\lambda_{Kr-85m}}}$$

The R/B fraction for isotope I-132 should be calculated using this fractal equation even though its half-life is less than 6 hours (2.28 hours) because its precursor of tellurium (Te)-132 has a half-life of 3.2 days, which controls the release of I-132.

B-2.1.2 Table B-2 lists the fractal-scaling factors for each nuclide calculated using the following equation from NUREG/CR-7003:

$$F_{nuclide} = \left(\frac{\alpha_{nuclide} \lambda_{Kr-85m}}{\lambda_{nuclide} \alpha_{Kr-85m}} \right)^{0.25}$$

Table B-2. Fractal Scaling Factors for Short-Lived Nuclides

NUCLIDE	NUREG/CR-7003, TABLE 1			FRACTAL SCALING FACTOR
	Half-Life	**Decay Constants**	**Alpha**	
Xe-133	5.243 days	1.53E-06	1.25	2.276
Xe-135	9.10 hours	2.12E-05	1.85	1.301
Xe-135m	15.3 months	7.55E-04	23.50	1.005
Xe-137	3.82 months	3.02E-03	1.07	0.328
Xe-138	14.1 months	8.19E-04	1.00	0.447
Xe-139	39.7 seconds	1.75E-02	1.00	0.208
Kr-85m	4.48 hours	4.30E-05	1.31	1.000
Kr-87	1.27 hours	1.52E-04	1.25	0.721
Kr-88	2.84 hours	6.78E-05	1.03	0.840
Kr-89	3.15 months	3.35E-03	1.21	0.330
Kr-90	32.3 seconds	2.15E-02	1.11	0.203
I-131	8.04 days	9.98E-07	1.00	2.395
I-132	2.28 hours	8.44E-05	137*	2.702
I-133	20.8 hours	9.26E-06	1.21	1.439
I-134	52.6 months	2.20E-04	4.40	0.900
I-135	6.57 hours	2.93E-05	1.00	1.029

* The I-132 alpha term accounts for significant contribution from precursor Te-132.

B-3. Fission product gap inventories are calculated at a 95-percent probability and 95-percent confidence level (95/95).

B-3.1 For short-lived isotopes, the 95/95 upper tolerance gap inventory is based on the empirical database used in the development of the FGR model. For example, the 2011 ANS-5.4 release model standard recommends multiplying the best estimate predictions by a factor of 5.0 to obtain upper tolerance gap inventories.

B-3.2 For long-lived isotopes, the 95/95 upper tolerance ($\mu + k\sigma$) gap inventory is based on the verification and validation database of the fuel thermal-mechanical code. An example of a 95/95 analysis is described in NUREG/CR-6534 (PNNL-11513), Volume 4, "FRAPCON-3 Updates, Including Mixed Oxide Properties" (Ref. B-8). FRAPCON-3.3 predicts that release fractions for long-lived isotopes exhibit a standard deviation of 0.028 (absolute) based on its validation database of measured stable noble gases from 23 fuel rods. With a database of 23 fuel rod measurements, $k = 2.36$, assuming a normal distribution and $23 - 2 = 21$ degrees of freedom. If FRAPCON-3.3 predicted a Kr-85 best estimate integral release fraction of 0.228, then the Kr-85 95/95 upper tolerance gap inventory ($\mu + k\sigma$) would equal 0.294 (0.228 + 2.36*0.028).

B-4. Nominal fuel design specifications (excluding tolerances) should be used.

B-5. Actual in-reactor fuel rod power histories may diverge from reload core depletion calculations because of unplanned shutdowns or power maneuvering. As a result, the rod power history or histories used to predict gap inventories should bound anticipated operation. Rod power histories used in the fuel rod design analysis based on core operating limits report thermal-mechanical operating limits or radial falloff curves should be used. The fuel rod power history used to calculate gap inventories should be verifiable.

B-5.1 The calculation supporting the bounding gap inventories in Table B-1 used a segmented power history for both the boiling-water reactor and pressurized-water reactor limiting designs. Seven different power histories were considered, with each running at 90 percent of the bounding rod average power, with the exception of running at the linear heat generation rate limit for approximately 9 to 10 gigawatt-days per metric ton of uranium (GWd/MTU) burnup (rod average) at seven different burnup intervals.

B-6. A flatter axial power distribution spreads the power and promotes a higher FGR along the fuel stack. A bounding axial power distribution should be used. Any rod axial power profile should be verifiable.

B-7. Each fuel rod design (e.g., UO_2, UO_2-Gd_2O_3, part-length, full-length) should be evaluated.

B-8. The minimum acceptable number of radial and axial nodes as defined in ANS-5.4 should be used, along with the methodology of summing the release for these nodes, to determine the overall release from the fuel to the fuel cladding gap.

The following example calculation illustrates the potential improvement in radiological source term achievable by calculating less bounding gap fractions. In this example, the licensee elects to calculate gap inventories based on cycle-specific rod designs and power profiles. The resulting gap fractions are significantly lower than the generic bounding values in Table B-1.

EXAMPLE CALCULATION
PRESSURIZED-WATER REACTOR GAP INVENTORIES
BASED ON REALISTIC POWER HISTORIES

This example illustrates the potential improvement in the radiological source term from calculating less bounding gap fractions. For this example, the licensee elects to calculate gap inventories based on

EXAMPLE CALCULATION
PRESSURIZED-WATER REACTOR GAP INVENTORIES
BASED ON REALISTIC POWER HISTORIES
(continued)

In this example, the FRAPCON-3.3 code with the ANS-5.4 release model is used to calculate the release fraction for stable nuclide Kr-85 and (R/B) ratios for volatile Kr-85m, Kr-87, and Kr-88 at each depletion time step for the two limiting fuel rod power histories. The licensee followed the analytical guidance in Regulatory Positions B-1, B-2, and B-3 to make adjustments to calculate the remaining nuclides. While this example employs the FRAPCON-3.3 code, licensees should use an NRC-approved fuel performance code with established modeling uncertainties.

<u>Long-Lived Stable Release</u>:

Kr-$85_{95/95}$	= [(Fission gas release)$_{FRAPCON}$ + (kσ)$_{FRAPCON}$]	
Power 1	= [(0.0614) + 2.36*0.028]	= 0.1275
Power 2	= [(0.0474) + 2.36*0.028]	= 0.1135
Cs-$134_{95/95}$	= [(Fission gas release)$_{FRAPCON}$ + (kσ)$_{FRAPCON}$]	
Power 1	= [(0.0614)$(2.0)^{0.5}$ + 2.36*0.028]	= 0.1529
Power 2	= [(0.0474)$(2.0)^{0.5}$ + 2.36*0.028]	= 0.1331

<u>Short-Lived Volatile (R/B) Ratio</u>:

Kr-85$m_{95/95}$	= [(Maximum Kr-85m R/B)$_{FRAPCON}$ * 5.0]	
Power 1	= [(0.0008) * 5.0]	= 0.0040
Power 2	= [(0.0013) * 5.0]	= 0.0065
Kr-$87_{95/95}$	= [(Maximum Kr-87 R/B)$_{FRAPCON}$ * 5.0]	
Power 1	= [(0.0004) * 5.0]	= 0.0020
Power 2	= [(0.0006) * 5.0]	= 0.0030
Kr-$88_{95/95}$	= [(Maximum Kr-88 R/B)$_{FRAPCON}$ * 5.0]	
Power 1	= [(0.0006) * 5.0]	= 0.0030
Power 2	= [(0.0009) * 5.0]	= 0.0045
Xe-$133_{95/95}$	= [(Maximum Kr-85m R/B)$_{FRAPCON}$ * 5.0 * (Fractal Scaling)$_{Xe\text{-}133}$]	
Power 1	= [(0.0008) * 5.0 * 2.276]	= 0.0091
Power 2	= [(0.0013) * 5.0 * 2.276]	= 0.0148
Xe-$135_{95/95}$	= [(Maximum Kr-85m R/B)$_{FRAPCON}$ * 5.0 * (Fractal Scaling)$_{Xe\text{-}135}$]	
Power 1	= [(0.0008) * 5.0 * 1.301]	= 0.0052
Power 2	= [(0.0013) * 5.0 * 1.301]	= 0.0085
I-$131_{95/95}$	= [(Maximum Kr-85m R/B)$_{FRAPCON}$ * 5.0 * (Fractal Scaling)$_{I\text{-}131}$]	
Power 1	= [(0.0008) * 5.0 * 2.395]	= 0.0096
Power 2	= [(0.0013) * 5.0 * 2.395]	= 0.0156

EXAMPLE CALCULATION
PRESSURIZED-WATER REACTOR GAP INVENTORIES
BASED ON REALISTIC POWER HISTORIES
(continued)

$I\text{-}132_{95/95}$ = [(Maximum Kr-85m R/B)$_{FRAPCON}$ * 5.0 * (Fractal Scaling)$_{I\text{-}132}$]
Power 1 = [(0.0008) * 5.0 * 2.702] = 0.0108
Power 2 = [(0.0013) * 5.0 * 2.702] = 0.0176

$I\text{-}133_{95/95}$ = [(Maximum Kr-85m R/B)$_{FRAPCON}$ * 5.0 * (Fractal Scaling)$_{I\text{-}133}$]
Power 1 = [(0.0008) * 5.0 * 1.439] = 0.0058
Power 2 = [(0.0013) * 5.0 * 1.439] = 0.0094

$I\text{-}134_{95/95}$ = [(Maximum Kr-85m R/B)$_{FRAPCON}$ * 5.0 * (Fractal Scaling)$_{I\text{-}134}$]
Power 1 = [(0.0008) * 5.0 * 0.900] = 0.0036
Power 2 = [(0.0013) * 5.0 * 0.900] = 0.0059

$I\text{-}135_{95/95}$ = [(Maximum Kr-85m R/B)$_{FRAPCON}$ * 5.0 * (Fractal Scaling)$_{I\text{-}135}$]
Power 1 = [(0.0008) * 5.0 * 1.029] = 0.0041
Power 2 = [(0.0013) * 5.0 * 1.029] = 0.0067

The cycle-specific fuel rod design and power history gap inventories are listed below, along with the generic bounding values from Table B-1 of this regulatory guide.

GROUP	**GAP INVENTORY**	
	Bounding	**Cycle-Specific**
I-131	0.08	0.02
I-132	0.06	0.02
Kr-85	0.36	0.13
Other Noble Gases	0.05	0.02
Other Halogens	0.05	0.01
Alkali Metals	0.49	0.16

REFERENCES[1]

B-1 U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Washington, DC.

B-2 NRC, RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," Washington, DC.

B-3 NRC, Memorandum, "Revised Technical Basis for Fission Product Release Fractions," June 4, 2019 (ADAMS Accession No. ML19154A226).

B-4 Pacific Northwest National Laboratory (PNNL), Report 18212, Revision 1, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard," June 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112070118).

B-5 NRC, Memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1," March 16, 2015 (ADAMS Accession No. ML14188C423).

B-6 American Nuclear Society, ANSI/ANS-5.4-2011, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel," LaGrange Park, IL, May 19, 2011.[2]

B-7 NRC, NUREG/CR-7003 (PNNL-18490), "Background and Derivation of ANS-5.4 Standard Fission Product Release Model," Washington, DC, January 2010 (ADAMS Accession No. ML100130186).

B-8 NRC, NUREG/CR-6534 (PNNL-11513), "FRAPCON-3 Updates, Including Mixed-Oxide Properties," Volume 4, Washington, DC, May 2005 (ADAMS Accession No. ML051440720).

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 ANSI/ANS standards may be purchased from the ANS Web site (http://www.new.ans.org/store/), or by writing to American Nuclear Society, 555 North Kensington Avenue, La Grange Park, IL 60526 (telephone: 800-323-3044).

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 the current commercial zirconium alloys to aid in the implementation of threshold curves for hydrogen-dependent, pellet-clad mechanical interaction cladding failure. These models also are 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) 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") 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 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.