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DRAFT REGULATORY GUIDE

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DRAFT REGULATORY GUIDE DG-1327

(Proposed New Regulatory Guide)

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, "Domestic Licensing of Production and Utilization Facilities," provides for the licensing of production and utilization facilities.
  - Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," contains general design criteria (GDC) for nuclear power plants. Criterion 28 (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

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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 include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

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 review of safety analysis reports submitted as part of license applications for nuclear power plants.
    - SRP Section 15.4.8 provides guidance to the NRC staff for reviewing PWR CRE accidents.
    - SRP Section 15.4.9 provides guidance to the NRC staff for reviewing BWR CRD accidents.
    - SRP Section 4.2 provides guidance to the NRC staff for reviewing reactor fuel designs.
    - SRP Section 4.2, Appendix B provides guidance to the NRC staff in 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 PWR CRE.
- Preliminary draft RG 1.224, "Establishing Analytical Limits for Zirconium-Alloy Cladding Material," (Ref. 7) includes guidance for estimating cladding hydrogen content.

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Purpose of Regulatory Guides

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

Paperwork Reduction Act

This Draft Regulatory Guide contains information collection requirements 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) control numbers 3150-0011 and 3150-0151.

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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. This guide replaces the PWR CRE analytical guidance provided RG 1.77.

Background

The NRC staff initially provided guidance for PWR CRE in RG 1.77 in 1974 (Ref. 6). 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 evaluated the effect of newly discovered burnup-related and cladding corrosion-related phenomena on fuel rod performance and issued interim acceptance criteria and guidance (Ref. 3 and 8~~7~~). In 2015, the staff evaluated newly published empirical data and analyses and identified further changes to guidance (Ref. 9~~8~~). Reference 9~~8~~ documents the empirical database as well as the technical and regulatory bases for this guide. That information is captured in this guide to reflect the latest state-of-knowledge.

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, the occurrence of 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 at a later moment, 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 results in a localized power excursion. This accident encompasses the consequences of all such reactivity control system excursions through postulating the worst possible combination of rod worth and core conditions.

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 can 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 must be limited to be within applicable regulations.

General Design Criterion (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 impair significantly the capability to cool the core. GDC 28 also requires that these postulated reactivity accidents include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line 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 fuel rod bundle array geometry. 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 (IAEA), the International Organization for Standardization (ISO), and the International Electrotechnical Commission (IEC) and did not identify any standards that provided useful guidance to NRC staff, applicants, or licensees.

C. STAFF REGULATORY GUIDANCE

This guide 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 be directly 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 be directly applicable to advanced LWRs and modular LWRs. Application of this guide beyond PWR CRE and BWR CRD, as well as the range of applicability described below, will be considered 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 are limited as follows:

1.1 ~~1.1~~ Currently approved LWR fuel rod designs comprised of slightly enriched UO_2 ceramic pellets (up to 5.0 wt% ^{235}U) within cylindrical zirconium-based cladding, including designs with or without barrier lined cladding, integral fuel burnable absorber (e.g., gadolinium), or a pellet central annulus.

1.1.1 ~~This guidance is not applicable to mixed oxide (MOX) fuel rod designs.~~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) will be addressed on a case-by-case basis.

1.1.2 Not applicable to mixed oxide (MOX) fuel rod designs.

1.2 ~~The high temperature cladding failure threshold described in Section 3.1 is applicable to reactor startup, zero power, and low power operations (i.e., < 5% rated power) and covers the entire initial reactor coolant temperature range (i.e., room temperature to operating temperatures). For all other operating conditions up to full power (i.e., Mode 1), fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., departure from nucleate boiling and critical power ratios).~~

~~1.3~~ As described in Section 3.2, separate PCMI cladding failure thresholds are provided for different initial reactor coolant temperatures and different cladding thermal annealing treatments.

1.2.1 The high temperature PCMI cladding failure threshold curves are applicable to reactor coolant temperatures at or above 500 ˚F. Below 500 ˚F, the low temperature PCMI cladding failure threshold curves are applicable.

1.2.2 The recrystallization annealed (RXA) PCMI cladding failure threshold curves are applicable to cladding which has undergone final thermal treatment that produces RXA metallurgical state, while the stress relief annealed (SRA) PCMI cladding failure threshold curves are applicable to cladding which has undergone final thermal treatment that produces SRA metallurgical state. For any other metallurgical condition, the applicant

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should provide justification for similarity with either SRA or RXA metallurgical condition~~The fully recrystallized annealed (RXA) PCMI cladding failure threshold curves are applicable to cladding which has undergone thermal treatment to remove all residual stresses and is in an RXA state. For all other stages of thermal treatments, the stress relief annealed (SRA) PCMI cladding failure threshold curves are applicable~~.

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1.2.3 Due to the dominant role liner fuel test results played in the development of the RXA PCMI cladding failure threshold curves and the influence of the natural of low alloy liner on the initial hydride distribution, the applicability of these failure threshold curves for non-liner cladding designs is limited to cladding with less than 70 wppm excess hydrogen.

2. Analytical Methods and Assumptions

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The following analytical inputs, assumptions, and methods are considered 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~~ that account for calculational uncertainties~~. The analytical models and computer codes used should be documented and justified, and the conservatism of the models and codes should be evaluated. C~~ both by c~~omparison with experiment and/or with 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, fuel element heat transfer parameters, and other relevant parameters should be included.

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

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

2.1.3 Calculations should be based upon design-specific information~~ accounting for manufacturing tolerances~~.

2.1.4 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) should be accounted for in fuel enthalpy calculations.

2.2 Initial conditions

2.2.1 Accident analyses should consider the full range of cycle operation from ~~be performed at~~ beginning of cycle (BOC) ~~and intermediate burnup intervals up to~~ to end of cycle (EOC).

2.2.2 Accident analyses at cold zero power (CZP) and hot zero power (HZP) conditions should encompass both (1) BOC following core reload and (2) re-start following recent power operation.

2.2.3 Accident analyses should consider the full range of power operation ~~be performed at intermediate power levels~~ up to hot full power (HFP) conditions. These calculations should consider~~confirm~~ power-dependent core operating limits (e.g., control rod insertion limits, rod power peaking limits, axial and azimuthal power distribution limits). At lower-power conditions where certain core operating limits do not apply, the analysis must consider the potential for wider operating conditions due to xenon oscillations or plant maneuvering.

2.2.4 The maximum uncontrolled rod worth (the worth of an ejected rod in a PWR or a dropped blade in a BWR) should be calculated based on the following conditions: (a) the range of control rod positions allowed at a given power level and (b) additional 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.~~Because of burnup-dependent and corrosion-dependent factors that tend to reduce cladding failure thresholds and allowable limits on damaged core coolability during fuel rod lifetime, the limiting initial conditions may involve the uncontrolled movement of lower-worth control rods or partially inserted control rods (e.g., uncontrolled rod motion at a core location adjacent to higher burnup fuel assemblies). As such, 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 and PWR ejected rod core locations and exposure points to identify the limiting scenarios.~~

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 maximum uncontrolled rod worth defined in section 2.2.4 (e.g., uncontrolled rod motion at a core location adjacent to higher burnup fuel assemblies). As such, 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 and 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 reduce the number of cases analyzed.

~~The maximum rod worth (or differential worth) should be calculated based on the following conditions: (a) all control rods at positions corresponding to values for maximum allowable insertions at a given power level and (b) additional 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 in each rod group for different control rod configurations, both expected and unexpected. The value of rod worths should be increased, if necessary, to account for calculational uncertainties in parameters (e.g., neutron cross sections) and power asymmetries due to xenon oscillations.~~

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

2.2.7 For CRE, 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. For CRD, credit may be taken for the velocity limiter when determining the rate of withdrawal due to gravitational forces.

2.2.8 For at-power scenarios, t~~T~~he initial reactor coolant pressure, core inlet temperature, and flow rate used in the analysis should be conservatively chosen, depending upon the transient phenomenon being investigated. Range of values should encompass the allowable operating range and monitoring uncertainties.

2.2.9 F~~The anticipated range of f~~uel 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 ~~should be investigated to ensure conservative values are chosen, depending upon~~ the transient phenomenon being investigated. Time-in-life specific fuel properties may be used for a given burnup-specific statepoint analysis.

2.2.10 The moderator reactivity coefficients due to 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 ~~account for monitoring~~ uncertainties.

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

2.2.12 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 due to allowable rod insertion should be quantified.

2.2.13~~2~~ 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: (a) time required for instrument channel to produce a signal, (b)

time for the trip breaker to open, (c) time for the control rod motion to initiate, and (d) time required before control rods enter the core if the tips lie outside the core. Allowances for inoperable or out-of-service components and single failures should be included in the response of the reactor protection system.

2.3 Predicting the total number of fuel rod failures

2.3.1 At each initial state point, the total number of failed rods that must be considered in the radiological assessment is equal to the sum of all of the fuel rods failing each of the cladding failure thresholds described in Section C.3, "Fuel Rod Cladding Failure Thresholds," 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. When applying Figure 1, the cladding differential pressure must include both the initial, pre-transient 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, pre-transient 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 provided in Figure 6.

2.3.3 Due to 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 pre-transient gas inventory, within the calculation of total rod internal pressure.

2.3.4 When applying the PCMI cladding failure thresholds, an NRC-approved alloy-specific cladding corrosion and hydrogen uptake model must be used to predict the initial, pre-transient cladding hydrogen content. 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 should be accounted for in these approved models either directly or implicitly through the supporting database.~~When applying the PCMI cladding failure thresholds, an approved alloy-specific cladding corrosion and hydrogen uptake model must be used to predict the initial, pre-transient cladding hydrogen content. 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 must be accounted for in these approved models.~~

2.3.4.1 As an alternative, acceptable alloy-specific hydrogen uptake models to estimate pre-transient cladding hydrogen content are provided in Appendix C~~Alloy-specific hydrogen uptake models in RG 1.224, "Establishing Analytical Limits for Zirconium-Based Cladding," (Ref. 9) may be used to estimate the pre-transient 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 are based on total

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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 their own approved alloy-specific hydrogen model which separates out hydrogen in the oxide layer, then these curves would no longer be applicable.

2.3.4.3~~2~~ The mid-wall cladding ~~average (e.g., mid-wall)~~ 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~~3~~ Due to 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 thermo-mechanical 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, a majority of zirconium hydride platelets will precipitate in the circumferentially 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). As described in References 11 and 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. Each applicant should address the possibility of hydride reorientation because of power maneuvering or reactor shutdown consistent with the requirements in NUREG-0800 Section 4.2, Fuel System Design, Section II.1.A.vi, page 4.2-7, Revision 3, March 2007.~~Because of the thermo-mechanical treatment of the cladding material under fabrication and its effect on the final cladding microstructure, zirconium hydride platelets will precipitate in a preferential orientation. Usually, SRA cladding exhibits circumferentially orientated zirconium hydride platelets, 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). As described in References 11 and 12, hydride reorientation from the circumferential direction to the radial direction is possible when the fuel rod is heated and subsequently cooled under an applied tensile load (e.g., high rod internal pressure).~~

2.3.6 Fuel cladding failure may occur almost instantaneously during the prompt fuel enthalpy rise (due to PCMI) or may occur as total fuel enthalpy (prompt + 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 cladding temperature failures, the total radial average fuel enthalpy (prompt + delayed) should be used.

~~2.3.5.1 The RXA PCMI failure curves in Figures 2 and 4 should be applied to any zirconium alloy cladding material that exhibits more than 10 percent~~

~~of the zirconium hydrides aligned in the radial direction. Otherwise, the SRA PCMI failure curves in Figures 4 and 5 should be applied.~~

~~2.3.5.2 Each applicant should address the possibility of hydride reorientation because of power maneuvering or reactor shutdown.~~

2.4 Fission product release fractions

2.4.1 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 pre-transient, steady-state inventories, to obtain the total radiological source term for dose calculations. See Appendix B for further information.

2.5 Reactor coolant system peak pressure

2.5.1 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, taking into account 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 amount and rate of reactivity insertion, fuel rods may experience several degradation mechanisms and failure modes. During a prompt critical reactivity insertion (i.e., $\Delta\rho/\beta_{eff} > 1.0$), fuel temperatures may approach melting temperatures, and rapid fuel pellet thermal expansion may promote PCMI cladding failure. During more benign power excursions, local heat flux may exceed critical heat flux conditions, prompting fuel cladding temperatures to rise. Fuel cladding may fail because of oxygen-induced embrittlement (i.e., brittle failure) or fuel rod ballooning and rupture (i.e., ductile failure).~~

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 which encompass each degradation mechanism and failure mode. To ensure a conservative assessment of onsite and offsite radiological consequences, each of these failure modes must be quantified, and the sum total number of failed fuel rods must not be underestimated.

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

The empirically based high temperature cladding failure threshold is shown in Figure 1. 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~~lower operating modes (e.g., Mode 2, less than 5 percent reactor power)~~. 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 ~~total~~ peak radial average fuel enthalpy (cal/g) versus fuel cladding differential pressure (MPa).

For at-power ~~all other~~ operating conditions (i.e., above 5 percent reactor power~~up to full power (i.e., Mode 1), fuel~~), 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

The empirically based PCMI cladding failure thresholds are shown in Figures 2 through 5. Because fuel cladding ductility is sensitive to initial temperature, hydrogen content, and zirconium hydride orientation, separate PCMI failure curves are provided for RXA and SRA cladding types at both low temperature reactor coolant conditions (e.g., BWR cold startup) and high temperature reactor coolant conditions (e.g., PWR hot zero power). The PCMI cladding failure threshold is expressed in peak radial average fuel enthalpy rise (Δcal/g) versus excess cladding hydrogen content (weight parts per million [wppm]). Excess cladding hydrogen content means the portion of total hydrogen content in the form of zirconium hydrides (i.e., 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

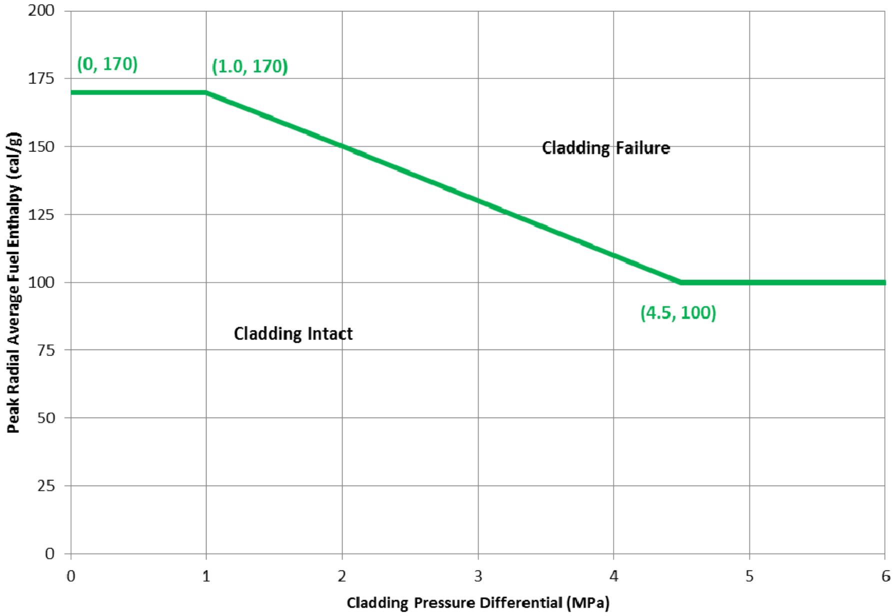
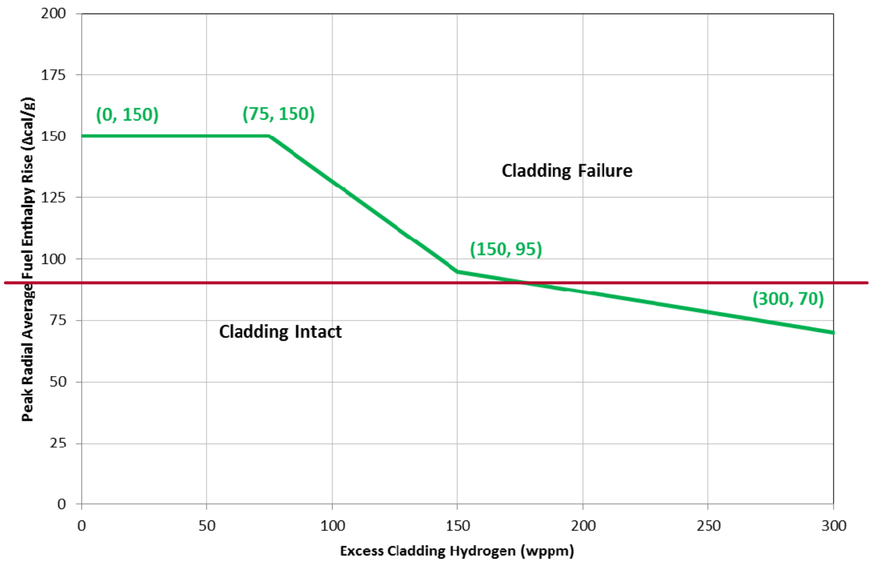


Figure 2: PCMI Cladding Failure Threshold—RXA Cladding at High Temperature Reactor Coolant Conditions



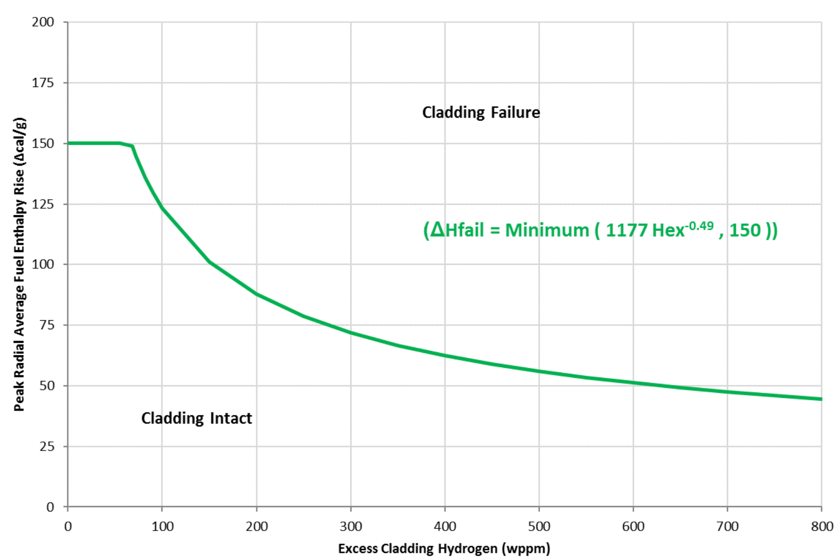
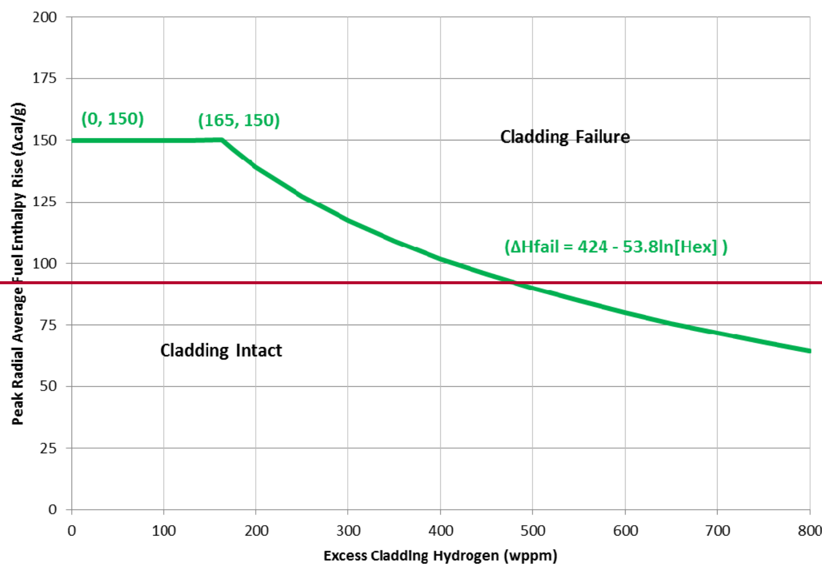


Figure 3: PCMI Cladding Failure Threshold—SRA Cladding at High Temperature Reactor Coolant Conditions


(0, 150)
(165, 150)
Cladding Failure
(ΔHfail = 424 - 53.8ln[Hex])
Cladding Intact
Peak Radial Average Fuel Enthalpy Rise (Δcal/g)
Excess Cladding Hydrogen (wppm)

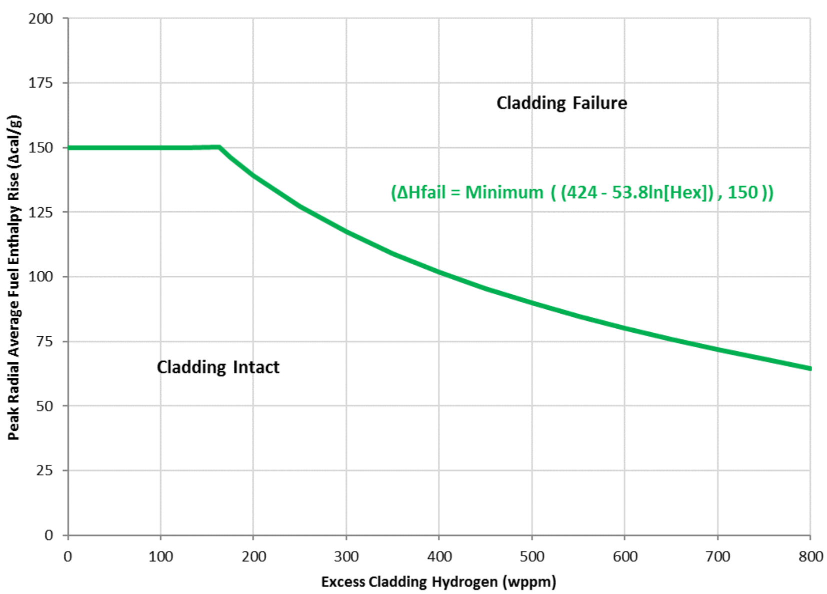
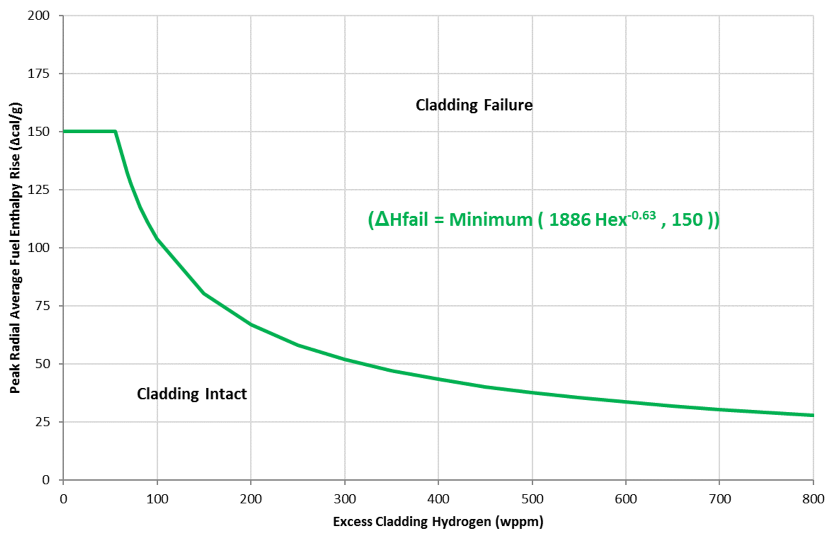

Cladding Failure
(ΔHfail = Minimum ((424 - 53.8ln[Hex]) , 150))
Cladding Intact
Peak Radial Average Fuel Enthalpy Rise (Δcal/g)
Excess Cladding Hydrogen (wppm)

Figure 4: PCMI Cladding Failure Threshold—RXA Cladding at Low Temperature Reactor Coolant Conditions



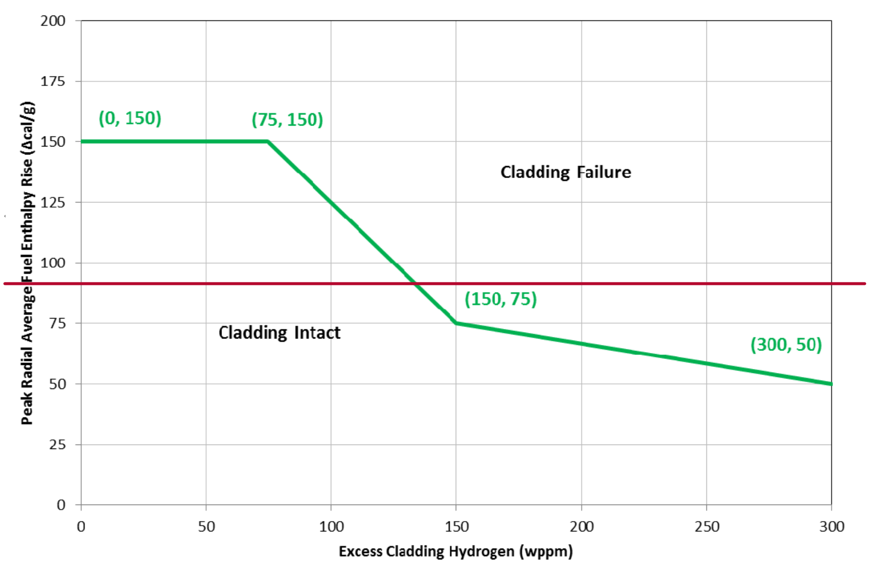
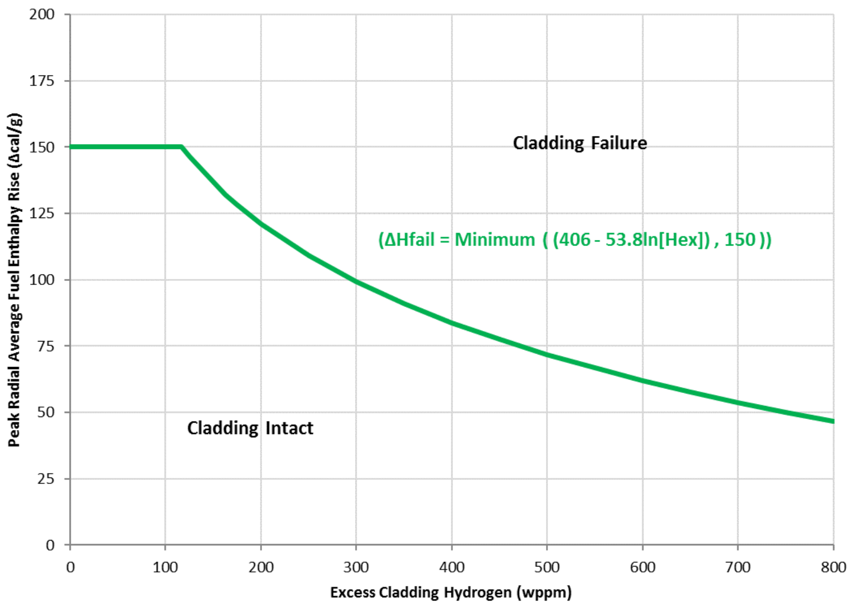
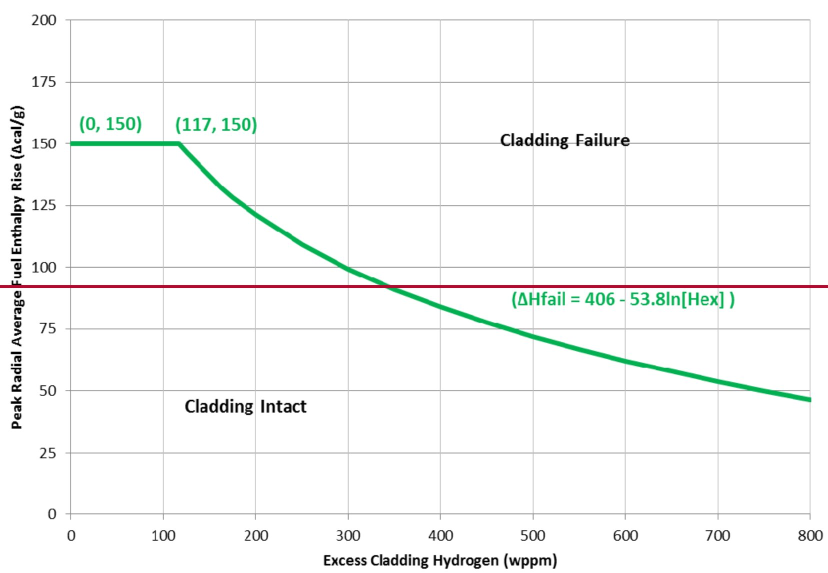


Figure 5: PCMI Cladding Failure Threshold—SRA Cladding at Low Temperature Reactor Coolant Conditions



Cladding Failure
(ΔHfail = Minimum ((406 - 53.8ln[Hex]) , 150))
Cladding Intact
Peak Radial Average Fuel Enthalpy Rise (Δcal/g)
Excess Cladding Hydrogen (wppm)



(0, 150)
(117, 150)
Cladding Failure
(ΔHfail = 406 - 53.8ln[Hex])
Cladding Intact
Peak Radial Average Fuel Enthalpy Rise (Δcal/g)
Excess Cladding Hydrogen (wppm)

~~4. Fission Product Release Fractions~~

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~~The total fission product fraction available for release following any event would include the steady-state fission product gap inventory (present before the event) plus any fission gas released during the transient. Whereas FGR (into the rod plenum) during normal operation is governed by diffusion, pellet fracturing and grain boundary separation are the primary mechanisms for FGR during the transient.~~

~~The empirically based transient FGR correlation is shown in Figure 6. 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 are provided as a function of peak radial average fuel enthalpy rise (Δcal/g).~~

~~An investigation into the effect of differences in diffusion coefficients and radioactive decay on fission product transient release (Ref. 13) concluded that adjustments to the empirically based correlations are needed for different radionuclides.~~

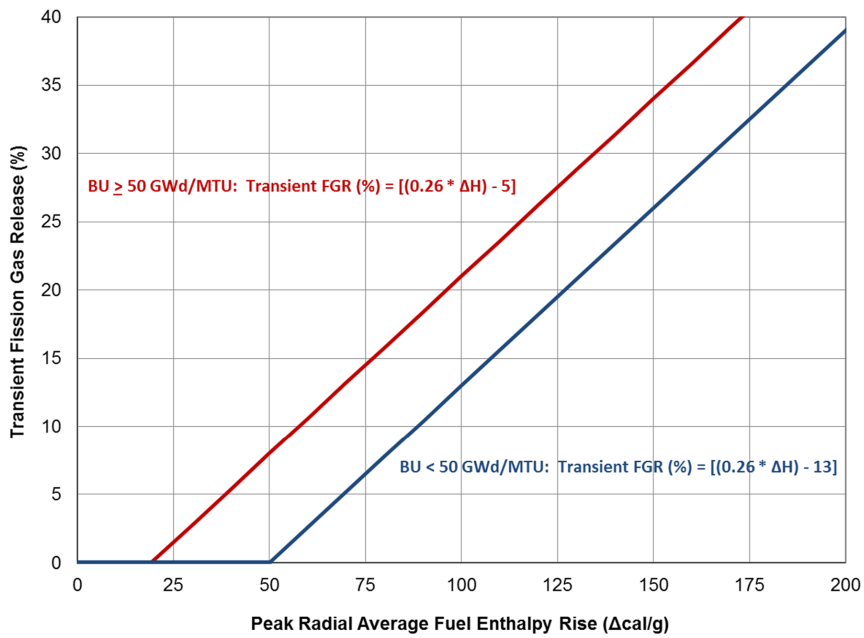
~~4.1 For stable, long-lived isotopes (e.g., Kr-85), the transient fission product release is equivalent to the burnup-dependent correlations provided in Figure 6.~~

~~4.2 For Cs-134 and Cs-137, the transient fission product release correlations provided in Figure 6 should be multiplied by a factor of 1.414.~~

~~4.3 For volatile, short-lived radioactive isotopes such as iodine (i.e., I-131, I-132, I-133, I-135) and xenon and krypton noble gases except Kr-85 (i.e., Xe-133, Xe-135, Kr-85m, Kr-87, Kr-88), the transient fission product release correlations provided in Figure 6 should be multiplied by a factor of 0.333.~~

~~4.4 The transient fission product release fractions must be added to the steady-state fission product gap inventory for each radionuclide (present before the event) to obtain the total radiological source term for dose calculations. Additional fission product releases from fuel melting may need to be included in total radiological source term. See RG 1.183 for steady-state fission product gap inventories and further guidance.~~

~~**Figure 6: Transient Fission Gas Release**~~



~~5~~. Allowable Limits on Radiological Consequences

The accident dose radiological consequences criteria for CRD and CRE accidents are provided in Regulatory Guide 1.183 (Ref. 4) and Regulatory Guide 1.195 (Ref. 5).

~~The offsite radiological consequences should be limited to "well within" the guidelines in 10 CFR Part 100, "Reactor Site Criteria," except for plants that adopt the alternate source term, which will be limited to "well within" the guidelines in 10 CFR Part 50.67. The term "well within" equates to 25 percent of allowable limits. For example, the allowable radiation dose for an individual located on the boundary of the exclusion area for any 2-hour period would be 6.25 rem total effective dose equivalent (TEDE) (equivalent to 25 percent of 25 rem TEDE prescribed in 10 CFR 50.67(b)(2)(i)). See RG 1.183 for further guidance.~~

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5~~6~~. Allowable Limits on Reactor Coolant System Pressure

~~The maximum reactor coolant system pressure should be limited to the value that will cause stresses to not exceed Emergency Condition (Service Level C), as defined in Section III of the ASME Boiler and Pressure Vessel code (Ref. 14).~~For new license applications, the maximum reactor coolant system pressure should be limited to the value that will cause stresses to not exceed Emergency Condition (Service Level C), as defined in Section III of the ASME Boiler and Pressure Vessel code (Ref. 14). For existing plants, allowable limits for the reactor pressure boundary as specified in the plant's UFSAR should be maintained.

6~~7~~. Allowable Limits on Damaged Core Coolability

L~~7.1 The l~~imiting peak radial average fuel enthalpy to prevent catastrophic fuel rod failure and avoiding molten fuel-c~~e~~oolant interaction (FCI) is an acceptable metric to demonstrate limited damage to core geometry and that the core remains amenable to cooling.

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~~7.2 T~~The following restrictions should be met:

a) ~~7.2.1~~ Peak radial average fuel enthalpy must remain below 230 cal/g.

A limited amount of fuel melting is acceptable provided it is restricted to less than 10 percent of fuel volume. The peak fuel temperature in the outer 90 percent of the fuel volume must remain below incipient fuel melting conditions

b) ~~7.2.2 A limited amount of fuel melting is acceptable provided it is restricted to the fuel centerline region and is less than 10 percent of pellet volume. The peak fuel temperature in the outer 90 percent of the pellet's volume must remain below incipient fuel melting conditions.~~

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For fresh and low-burnup fuel rods, the peak radial average fuel enthalpy restriction will likely be more limiting than the limited fuel ~~centerline~~ 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 must 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 regulatory guide. 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 regulatory guide 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 regulatory guide 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 regulatory guide 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 regulatory guide. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this regulatory guide 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 regulatory guide 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 regulatory guide. Examples of such unplanned NRC regulatory actions include issuance of an order requiring the use of the regulatory guide, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this regulatory guide, generic communication, or promulgation of a rule requiring the use of this regulatory guide without further backfit consideration.

During regulatory discussions on plant-specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this regulatory guide, 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 regulatory guide are part of the licensing basis of the facility. However, unless this regulatory guide is part of the licensing basis for a facility, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this regulatory guide constitutes a violation.

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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.

If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this new or revised regulatory guide, and (2) the specific subject matter of this regulatory guide is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this regulatory guide or provide an equivalent alternative process that demonstrates compliance with the underlying NRC regulatory requirements. This is not considered backfitting as defined in 10 CFR 50.109(a)(1) or a violation of any of the issue finality provisions in 10 CFR Part 52.

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 regulatory guide or requesting or requiring the licensee to implement the methods or processes in this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 15) and NUREG-1409, "Backfitting Guidelines" (Ref. 16).

E. 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," Washington, DC.

4. NRC, 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, Regulatory Guide (RG) 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors." Washington, DC.

7. NRC, RG 1.224, "Establishing Analytical Limits for Zirconium-Based Cladding," Washington, DC (Preliminary Draft).

~~7.~~8. NRC memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Interim Acceptance Criteria and Guidance," January 19, 2007 (ADAMS Accession No. ML070220400).

~~8.~~9. 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, RG 1.224, "Establishing Analytical Limits for Zirconium-Based Cladding," Washington, DC.~~

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].

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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, Tel: 1 212 989-5800

11. Organisation 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 (ADAMS Accession No. ML101460362).

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 PATRAM, 2013. [5]

13. Pacific Northwest National Laboratory Report 18212 Revision 1, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard," June 2011 (ADAMS Accession No. ML112070118).

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

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

16. NRC, NUREG-1409, "Backfitting Guidelines," Washington, DC, July 1990, (ADAMS Accession No. ML032230247).

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

[6] Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Two Park Avenue, New York, New York 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 Definitions

ADAMS	Agencywide documents access and management system
AOO	Anticipated operational occurrence
ASME	American Society for Mechanical Engineers
BIGR	Fast Pulse Graphite Reactor
BOC	Beginning of cycle
BWR	Boiling water reactor
CEDM	Control element drive mechanism
CFR	Code of Federal Regulations
CRD	Control rod (blade) drop
CRE	Control rod ejection
CZP	Cold zero power
Doppler or FTC	Fuel temperature coefficient, change in reactivity/change in fuel temperature
EOC	End of cycle
FCI	Fuel-coolant interaction
FGR	Fission gas release
GDC	General design criteria
HFP	Hot full power
HZP	Hot zero power
IAEA	International Atomic Energy Agency
IEC	International Electrotechnical Commission
IGR	Impulse Graphite Reactor
ISO	International Organization for Standardization
l*	Average neutron lifetime
MOX	Mixed uranium-plutonium oxide
NRC	Nuclear Regulatory Commission
NSRR	Nuclear Safety Research Reactor
OMB	Office of Management and Budget
PBF	Power Burst Facility
PCMI	Pellet-clad mechanical interaction
PWR	Pressurized water reactor
RG	Regulatory guide
RXA	Recrystallized annealed
SPERT	Special Power Excursion Test Reactor
SRA	Stress relief annealed
SRP	Standard review plan (NUREG-0800)
TEDE	Total effective dose equivalent
TREAT	Transient Reactor Test Facility
wppm	Weight parts per million
βeff	Effective delayed neutron fraction
Δρ	Change in reactivity, δk/keff

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Appendix B

Fission Product Release Fractions

This appendix provides guidance on steady-state and transient gap fission product inventories available for release following a CRE or CRD accident. The generic steady-state gap fractions and analytical approach for calculating steady-state gap fractions are acceptable for use to define the initial radionuclide inventory for all Non-LOCA design basis accidents. This appendix guidance for steady state and transient gap fission product inventories should be used until such time when it is transferred to a future revision of RG 1.183 (Ref. 4) and RG 1.195 (Ref. 5). Please note that for fuel that is predicted to melt (fuel that reaches or exceed the initiation temperature for melting) as a result of the Non-LOCA design basis accidents, the steady state and transient gap fission products are added to the melt activity as discussed in RG 1.183 and RG 1.195.

The total fractions of fission products available for releases for the CRE and CRD accidents are equal to the sum of the steady-state fission product gap inventory and transient fission gas release and the estimate of the number of fuel rods breached. Whereas, the FGR (into the rod plenum) during normal operations is governed by diffusion, pellet fracturing and grain boundary separation are the primary mechanisms for FGR during the transient. For fuel that melts (fuel that reaches or exceeds the initiation temperature for fuel melting) the combined fission product inventory (steady-state gap plus transient release) is added to the release due to fuel melting. Guidance regarding source term for fuel melting contained in RG 1.183 and 1.195 remains unchanged.

~~4. Fission Product Release Fractions~~

~~The total fission product fraction available for release following any event would include the steady-state fission product gap inventory (present before the event) plus any fission gas released during the transient. Whereas FGR (into the rod plenum) during normal operation is governed by diffusion, pellet fracturing and grain boundary separation are the primary mechanisms for FGR during the transient.~~

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 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	0.09
Kr-85	0.38
Other Noble Gases	0.09
Other Halogens	0.05
Alkali Metals	0.50[7]

[7] 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 Kr-85. Chemical changes that accompany extended levels of reactor fuel burnup may retard

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The derivation of the steady-state fission product gap inventories, including the analytical methods and application of uncertainties, is documented in Reference B-1. As an alternative, a licensee may use the 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.

Transient Fission Gas Release

~~The empirically based transient FGR correlation is shown in Figure 6.~~ 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 (Δcal/g). The derivation of the transient fission gas release correlations, including the application of uncertainties, is documented in Reference B-2.

Pellet BU < 50 GWd/MTU

$$\text{Transient FGR} = \text{Maximum}\ [\ (((0.26 * \Delta H) - 13) / 100),\ 0\ ]$$

Pellet BU $\geq$ 50 GWd/MTU

$$\text{Transient FGR} = \text{Maximum}\ [\ (((0.26 * \Delta H) - 5) / 100),\ 0\ ]$$

Where:

FGR = Fission gas release, fraction

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

~~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 are provided as a function of peak radial average fuel enthalpy rise (Δcal/g).~~

An investigation into the effect of differences in diffusion coefficients and radioactive decay on fission product transient release (Ref. B-1~~13~~) concluded that adjustments to the empirically based correlations are needed for different radionuclides.

B4.1 For stable, long-lived isotopes (e.g., Kr-85), the transient fission product release is equivalent to the burnup-dependent correlations ~~provided in Figure 6~~.

B4.2 For alkali metals, such as Cs-134 and Cs-137, the transient fission product release correlations ~~provided in Figure 6~~ should be multiplied by a factor of 1.414.

B4.3 For volatile, short-lived radioactive isotopes such as halogens~~iodine~~ (e.g.~~i.e.~~, I-131, I-132, I-133, I-135) and xenon and krypton noble gases except Kr-85 (e.g.~~i.e.~~, Xe-133, Xe-135, Kr-85m, Kr-87, Kr-88), the transient fission product release correlations ~~provided in Figure 6~~ should be multiplied by a factor of 0.333.

Total Fission Product Inventory Available for Release

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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.

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~~4.4~~ The transient fission product release fractions must 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~~radiological source term for dose calculations.~~ 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, a~~A~~dditional fission product releases ~~from fuel melting~~ may need to be included in total radiological source term. ~~See RG 1.183 for steady-state fission product gap inventories and further guidance.~~

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

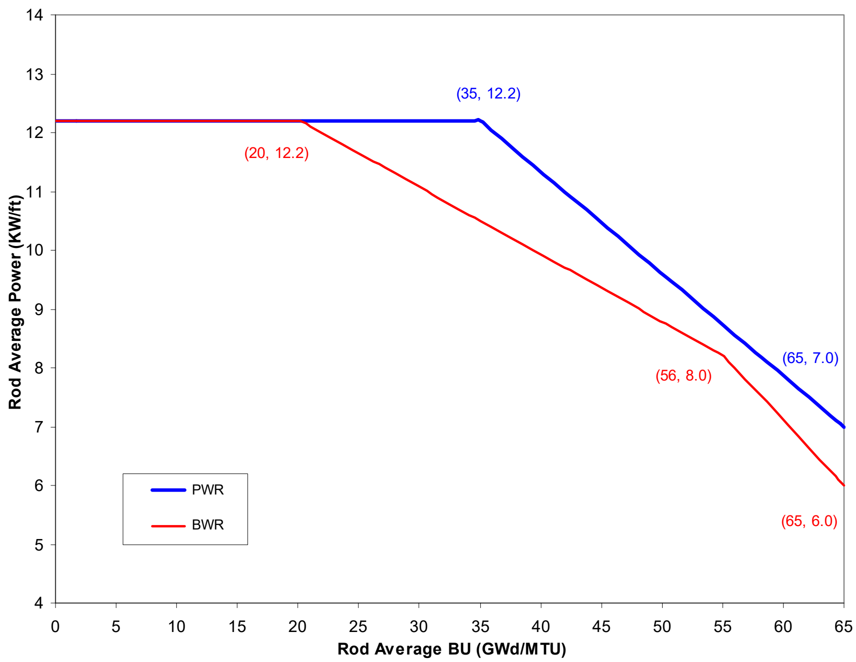

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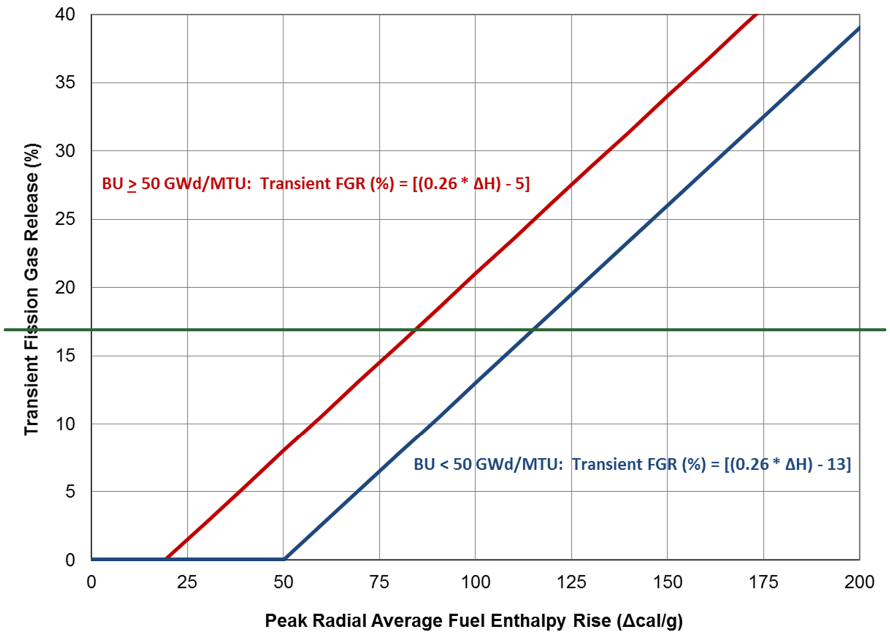
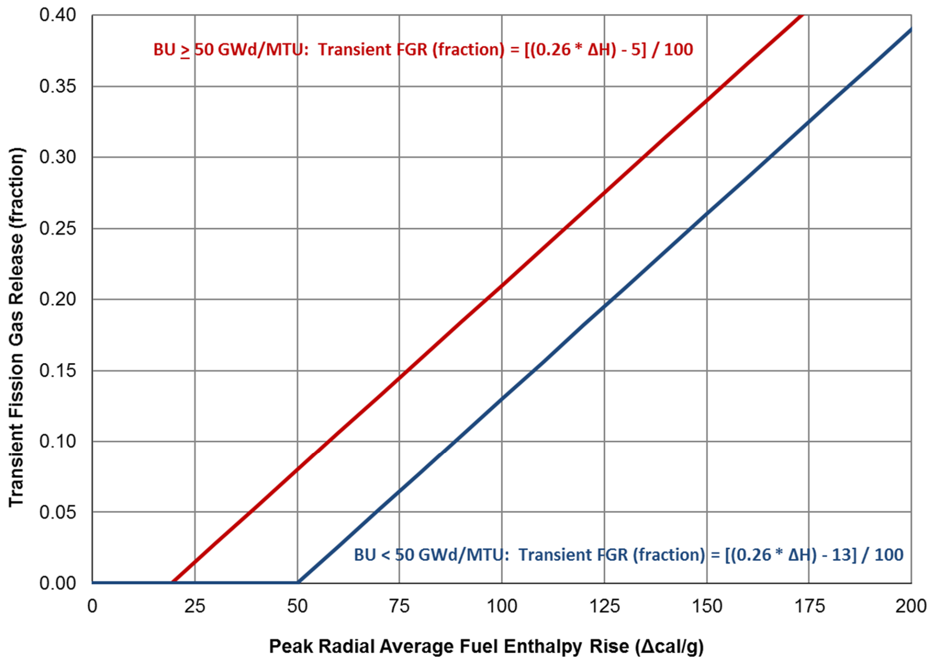
Figure B-2~~6~~: Transient Fission Gas Release



BU ≥ 50 GWd/MTU: Transient FGR (%) = [(0.26 * ΔH) - 5]
BU < 50 GWd/MTU: Transient FGR (%) = [(0.26 * ΔH) - 13]
Transient Fission Gas Release (%)
Peak Radial Average Fuel Enthalpy Rise (Δcal/g)
0
5
10
15
20
25
30
35
40
0
25
50
75
100
125
150
175
200



BU > 50 GWd/MTU: Transient FGR (fraction) = [(0.26 * ΔH) - 5] / 100
BU < 50 GWd/MTU: Transient FGR (fraction) = [(0.26 * ΔH) - 13] / 100
Transient Fission Gas Release (fraction)
0.40
0.35
0.30
0.25
0.20
0.15
0.10
0.05
0.00
0
25
50
75
100
125
150
175
200
Peak Radial Average Fuel Enthalpy Rise (Δcal/g)

Attachment:

ANALYTICAL TECHNIQUE FOR CALCULATING STEADY-STATE FISSION PRODUCT GAP INVENTORIES

This attachment 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 B-1 as long as 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 isotopes). As such, 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 U.S. Nuclear Regulatory Commission (NRC) maintains the FRAPCON fuel rod thermal-mechanical fuel performance code to perform independent audit calculations for licensing activities. As such, licensees may not use FRAPCON to justify plant-, fuel-, or cycle-specific gap inventories for non-LOCA accidents.

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 shall be used to predict the integral fission gas release. The code should include the effects of thermal conductivity degradation with burnup and should have been verified against measured fuel temperatures and stable fission gas release 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 amounts of decay because of their long half-lives. As such, 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, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuels" (Ref. B-3), 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.

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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 (e.g., Ref. B-3) shall 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 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 fission gas release 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-4), provides guidance on using the 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."

B-2.1.2 For nuclides with half-lives of less than 6 hours, an approved fuel performance code is used to predict the R/B fraction using Equation 12 of, and the terms defined in, NUREG/CR-7003, as follows:

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

B-2.1.3 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-2.1.4 Table B-2 lists the fractal-scaling factors for each nuclide calculated using the following equation from NUREG/CR-7003:

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

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 fission gas release model. For example, the 2011 ANS-5.4 release model standard (Ref. B-3) 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 (μ + kσ) gap inventory is based on the verification and validation database of the fuel thermal-mechanical code. For example, FRAPCON-3.3 (Ref. B-5) predicted 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 (μ + kσ) would equal 0.294 (0.228 + 2.36*0.028).

B-4. Nominal fuel design specifications (excluding tolerances) must 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 must 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. Any rod power history must be verifiable.

B-5.1 The calculation supporting the bounding gap inventories in Table B-1 used a segmented power history for both the BWR and PWR limiting designs, whereby 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 GWd/MTU burnup (rod average) at seven different burnup intervals.

B-6. A flatter axial power distribution (e.g., low value of Fz) spreads the power and promotes a higher fission gas release along the fuel stack. A bounding axial power distribution should be used. Any rod axial power profile must be verifiable.

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

B-8. Use the minimum acceptable number of radial and axial nodes as defined in ANS-5.4 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 example calculation below 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 upon cycle-specific rod designs and power profiles. The resulting gap fractions are significantly lower than the generic, bounding values in Table B-1.

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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

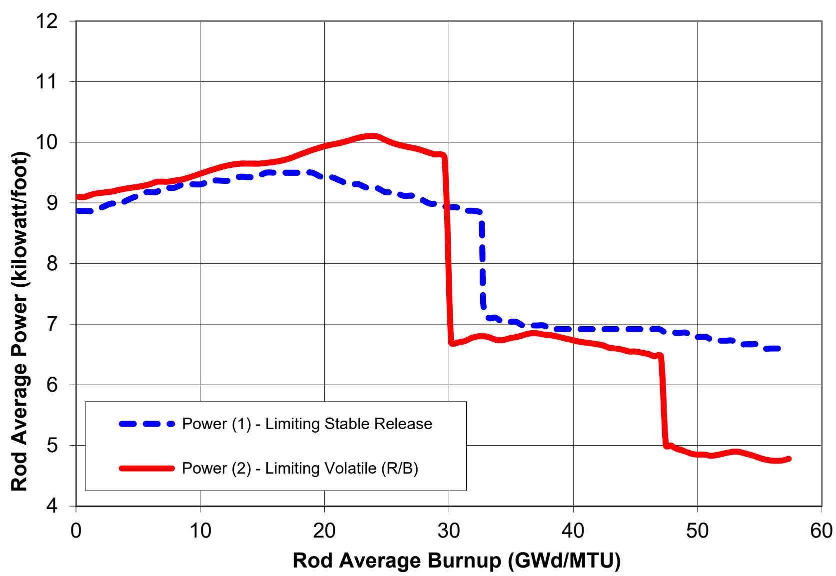
* The I-132 alpha term accounts for significant contribution from precursor Te-132.

EXAMPLE CALCULATION
PWR 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 cycle-specific rod designs and power profiles.

For this cycle, the licensee surveys the reload depletion and identifies the limiting fuel rod power histories for long-lived stable isotopes and short-lived volatile isotopes. The licensee then makes adjustments to account for power uncertainties and plant maneuvering. The figure below illustrates the limiting fuel rod power history for the calculation of stable releases and volatile (R/B) ratios. The licensee has verified that the full-length UO_2 fuel rod design is the most limiting.

As discussed in the analytical procedure, maximum stable releases occur at the end of life in fuel rods with a relatively high power during their second cycle of operation. Maximum volatile (R/B) ratios occur near the highest rod power at low to middle burnup. A high probability exists that the limiting fuel rod design and power history identified for the fuel rod thermal-mechanical rod internal pressure analysis will coincide with that for maximum stable releases. Similarly, the limiting fuel rod design and power history identified for the fuel rod thermal-mechanical anticipated operational occurrence fuel centerline melt analysis will coincide with that for maximum (R/B) ratios.



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EXAMPLE CALCULATION
PWR GAP INVENTORIES BASED ON REALISTIC POWER HISTORIES
(Continued)

In this example, the ~~licensee used the~~ FRAPCON-3.3 code with the ASN-5.4 release model is employed 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~~J~~-1, B~~J~~-2, and B~~J~~-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.

Long-Lived Stable Release:
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

Short-Lived Volatile (R/B) Ratio:
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)$_{Xe-133}$]
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)$_{Xe-135}$]
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)$_{I-131}$]
Power 1 = [(0.0008) * 5.0 * 2.395] = 0.0096
Power 2 = [(0.0013) * 5.0 * 2.395] = 0.0156

EXAMPLE CALCULATION
PWR 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~~3~~ of this regulatory guide.

GROUP	GAP INVENTORY	
	Bounding	**Cycle-Specific**
I-131	0.08	0.02
I-132	0.09	0.02
Kr-85	0.38	0.13
Other Noble Gases	0.09~~8~~	0.02
Other Halogens	0.05	0.01
Alkali Metals	0.50	0.16

REFERENCES

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

B-2 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-3 ANSI/ANS-5.4-2011, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel," American Nuclear Society, May 19, 2011.

B-4 NUREG/CR-7003 (PNNL-18490), "Background and Derivation of ANS-5.4 Standard Fission Product Release Model," A.J. Turnbull and C.E. Beyer, U.S. Nuclear Regulatory Commission, Washington, DC, January 2010.

B-5 NUREG/CR-6534 (PNNL-11513), "FRAPCON-3 Updates, Including Mixed Oxide Properties," Volume 4, D.D. Lanning, C.E. Beyer, and K.J. Geelhood, U.S. Nuclear Regulatory Commission, Washington DC, 2005.

Appendix C

Fuel Rod Cladding Hydrogen Uptake Models

The purpose of this appendix is to provide acceptable fuel rod cladding hydrogen uptake models for the current commercial zirconium alloys to aid in the implementation of the hydrogen-dependent PCMI cladding failure threshold curves. These models also are acceptable for implementing other hydrogen-dependent fuel performance requirements, e.g., ECCS analytical limits on peak cladding temperature and integral time-at-temperature (expressed as equivalent cladding reacted calculated using the Cathcart-Pawell correlation (CP-ECR)) as a function of pre-transient cladding hydrogen content. The models provided in this appendix were originally documented in draft RG 1.224 (Ref. 7).

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PWR Zirconium Cladding Alloys

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

As described in Appendix A of Reference 7, the staff compiled a database of measured cladding hydrogen content for the current commercial PWR zirconium alloys. The PWR empirical database does not exhibit the same breakaway hydrogen uptake at higher fluence levels as observed in BWR Zircaloy-2 data. 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.

BWR Zircaloy-2 Cladding

For BWR conditions, a constant hydrogen pickup fraction does not fit the observed cladding hydrogen data. As described in Appendix A of Reference 7, the staff compiled a database of measured cladding hydrogen content for legacy and modern commercial BWR Zry-2 cladding alloys. Given the allowable range in composition within the Zircaloy-2 ASTM specification (ASTM B351/B351M, *Standard Specification for Hot-Rolled and Cold-Finished Zirconium and Zirconium Alloy Bars, Rod, and Wire for Nuclear Application*) and the degree of flexibility and variability in manufacturing procedures, the staff elected 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, wppm
BU = local axial burnup, GWd/MTU

Applicability

The hydrogen models are applicable to currently approved commercial alloys up to their respective limits on fuel rod burnup, corrosion, and residence time. The hydrogen models are not applicable to fuel rods that experience oxide spallation.

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