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DRAFT REGULATORY GUIDE

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(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 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

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Electronic copies of this draft regulatory guide, previous versions of this guide, and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/. The draft regulatory guide is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML16124A200. The regulatory analysis may be found in ADAMS under Accession No. ML16124A198.

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 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.

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.

Public Protection Notification

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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.

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 7). In 2015, the staff evaluated newly published empirical data and analyses and identified further changes to guidance (Ref. 8). Reference 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 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. This guidance is 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. 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. 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.

2. **Analytical Methods and Assumptions**

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 both by comparison with experiment and with more sophisticated spatial kinetics codes. 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.

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 be performed at beginning of cycle (BOC) and intermediate burnup intervals up 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 be performed at intermediate power levels up to hot full power (HFP) conditions. These calculations should 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 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 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 The 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 The anticipated range of fuel thermal properties (e.g., fuel-clad gap thermal conductivity, fuel thermal conductivity) over the fuel rod's lifetime should be investigated to ensure conservative values are chosen, depending upon the transient phenomenon being investigated.

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 account for monitoring uncertainties.

2.2.11 Calculations of the Doppler coefficient of reactivity should be based on and should compare conservatively with available experimental data. Since the Doppler coefficient reflects the change in reactivity as a function of fuel temperature, uncertainties in predicting fuel temperatures at different power levels should be reflected by conservatism in the applied 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. Any loss of available scram reactivity due to allowable rod insertion should be quantified.

2.2.12 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 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 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 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.

2.3.4.3 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. 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).

2.3.5.1 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.

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). 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.

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 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 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).

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

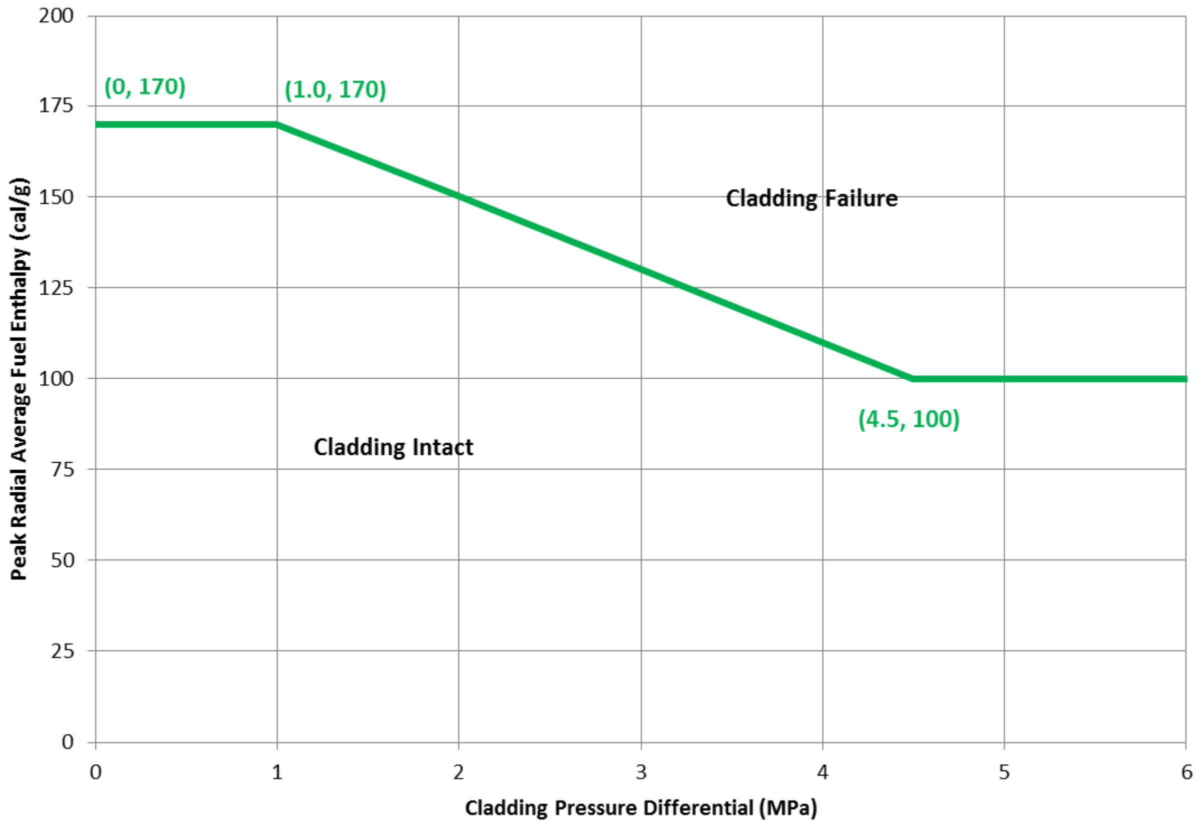


Figure 2: PCMI Cladding Failure Threshold—RXA Cladding at High Temperature Reactor Coolant Conditions

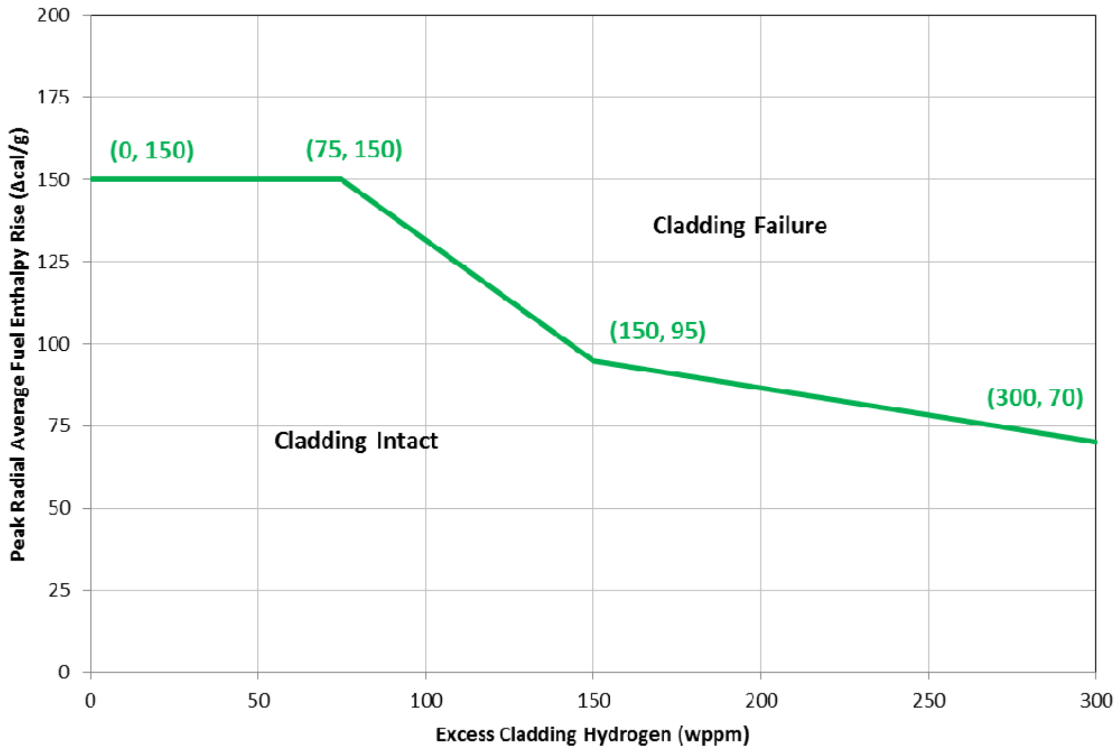


Figure 3: PCMI Cladding Failure Threshold—SRA Cladding at High Temperature Reactor Coolant Conditions

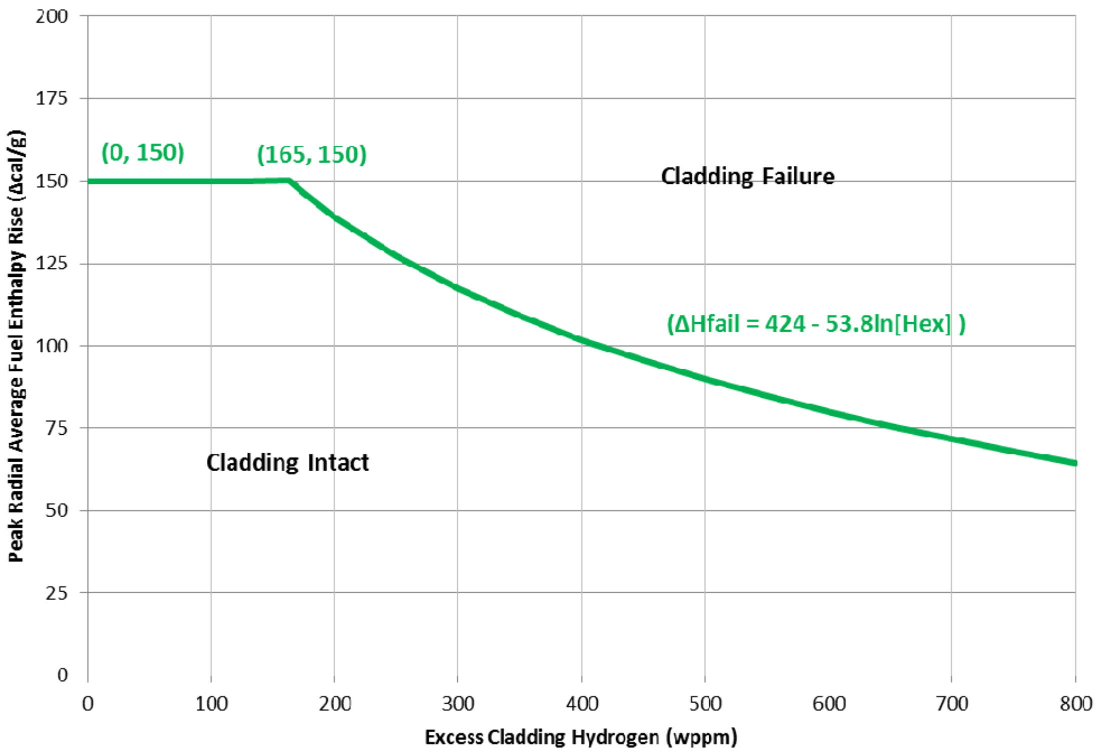


Figure 4: PCMI Cladding Failure Threshold—RXA Cladding at Low Temperature Reactor Coolant Conditions

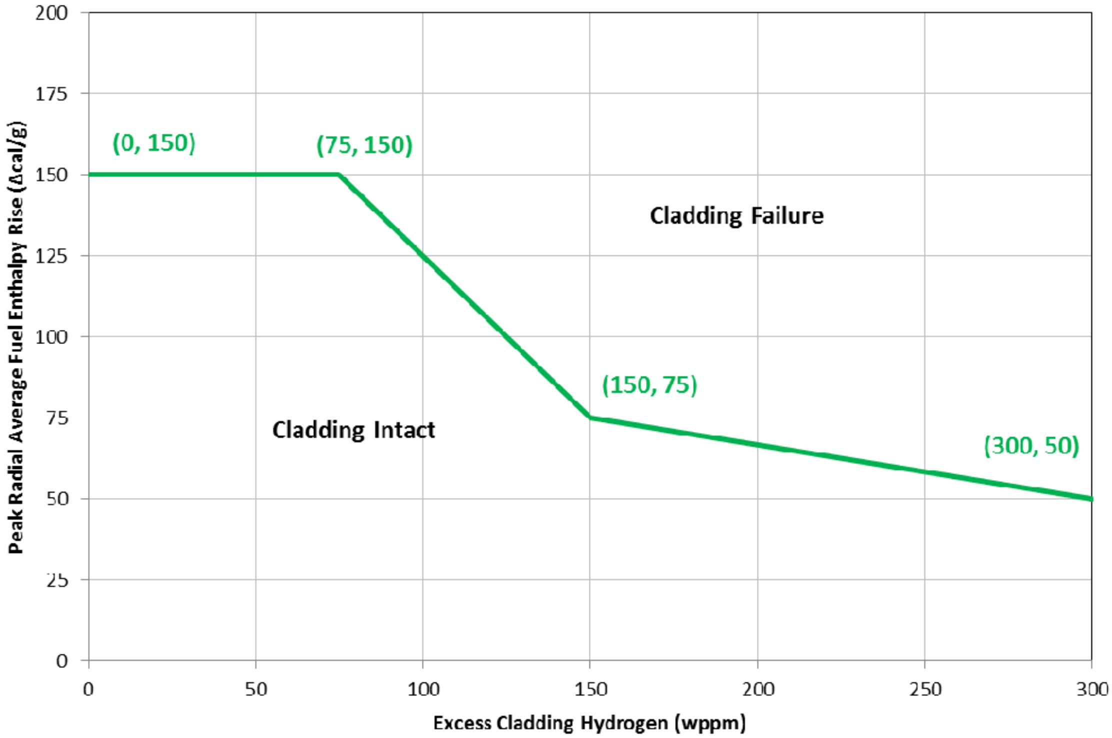
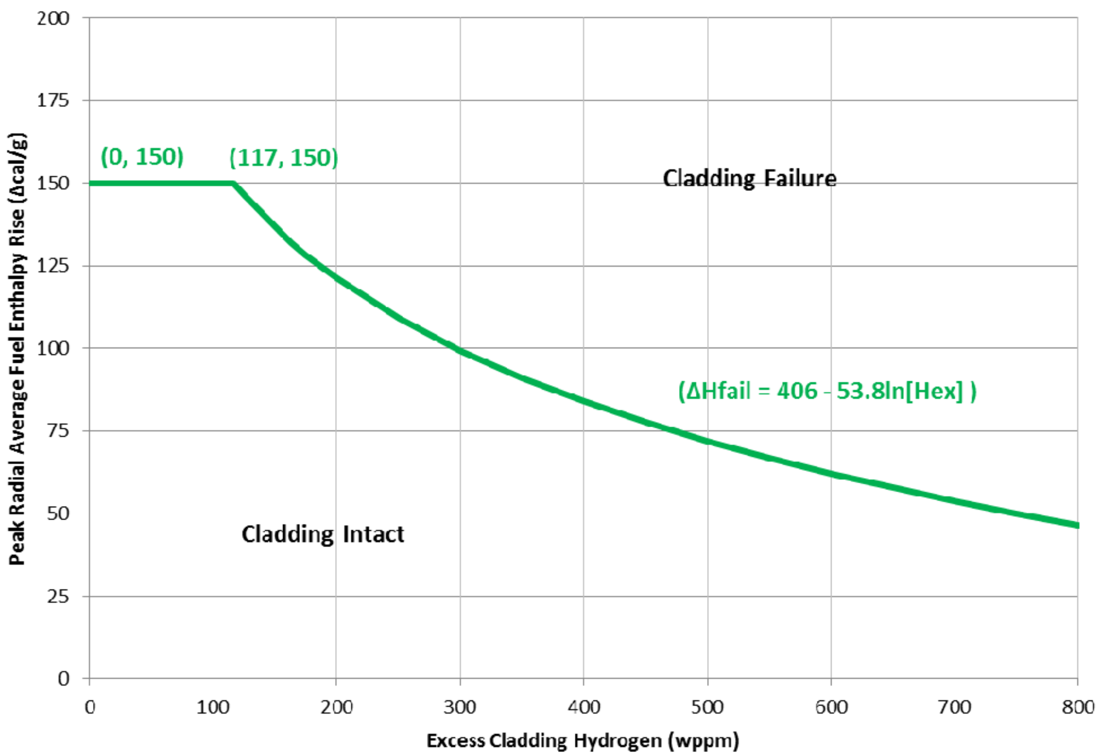


Figure 5: PCMI Cladding Failure Threshold—SRA Cladding at Low Temperature Reactor Coolant Conditions



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.

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.

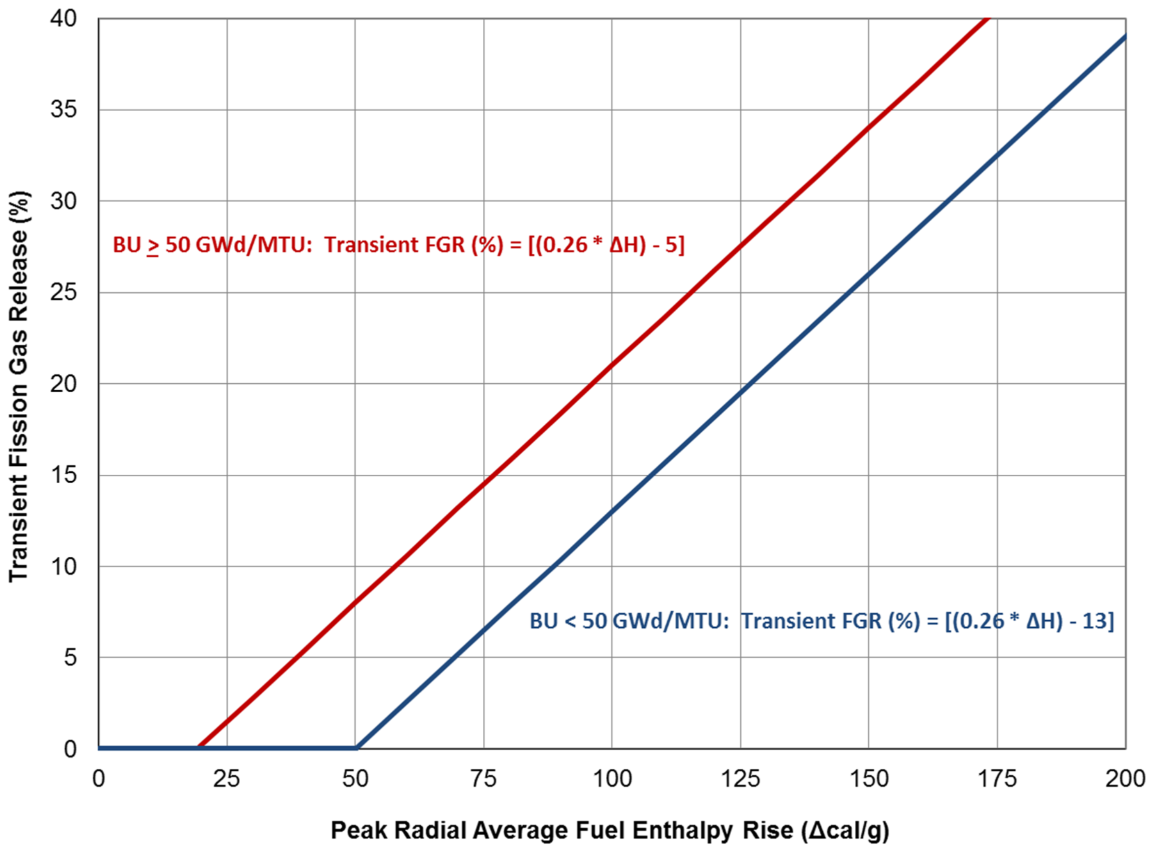
4.1 For stable, long-lived isotopes (e.g., Kr-85), the transient fission product release is equivalent to the burnup-dependent correlations provided in Figure 6.

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 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.

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).

7. Allowable Limits on Damaged Core Coolability

7.1 The limiting peak radial average fuel enthalpy to prevent catastrophic fuel rod failure and avoiding molten fuel-coolant interaction (FCI) is an acceptable metric to demonstrate limited damage to core geometry and that the core remains amenable to cooling.

7.2 The following restrictions should be met:

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

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 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, 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].

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).

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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

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/