DG-1327 Clarification

Reactivity Initiated Accident Guidance

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January 25, 2017 • White Flint, Rockville MD

Overview

- Categories for Discussion
  - Methods & Assumptions
  - Failure Thresholds
  - Release Fractions
  - Miscellaneous

PWR vs BWR Perspective

- **While Rod Eject / Blade Drop are Reactivity Initiated Accidents…**
  - These different events don't share an identical topology
    - Analytical space is different
  - Every assumption isn't automatically meaningful to both PWR's and BWR's
  - Example from item 2.2.5
    - (a) is PWR speak
    - (b) is BWR speak
  - Example item 2.2.10
    - Muddy regarding BWR

Methods & Assumptions

- **Approved Models**
  - What does "account for calculational uncertainties mean"?
  - Realistic / Risk Informed methods to be allowed?
  - Expecting a full RG 1.203 process?

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

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Methods & Assumptions

- **5% power DNB/CPR threshold?**
  - Value is below TS monitoring power level.
    - Correlation range of applicability may not extend that low
  - DNB/CPR may not be appropriate metric relative to very fast transient condition

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Methods & Assumptions

- ## Misc. Assumptions
  - Are "sensitivity" studies going to be plant and cycle specific?
  - What is NOT a "major reactivity feedback?"
    - Direct Moderator Heating
      - Non-Eq. T-H
  - What is meant by "manufacturing tolerances?"
    - Plant , fuel type, and/or cycle specific.
      - As-built vs bounding tolerance
  - Accounting for something vs. sensitivity/parametric evaluation.

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

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Methods & Assumptions

- **Misc. Assumptions**
  - What is meant by “wider operating conditions”?
  - Effectively, you’re saying the determination of limiting conditions is non-linear.
    - When does the search stop? To “survey a larger population” implies a realistic assessment.
  - What is “sufficient parametric study”?

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.

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Methods & Assumptions

- **Misc. Assumptions**
  - Why do advanced methods need to implement artificial conservatism to compare against failure criteria?
  - Extensive focus on bounding assumptions
    - Seems incompatible with implications of 2.2.4 (limiting scenario tied to non-linear effects, not artificial conservatisms)

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.

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Methods & Assumptions

- **Misc. Assumptions**
  - Approved hydrogen pickup model is explicit
  - Need for an approved hydride orientation model is not as obvious. Need to validate the failure curve utilized.
  - Is the use of RG 1.224 account for hydride orientation issue?

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.

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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).

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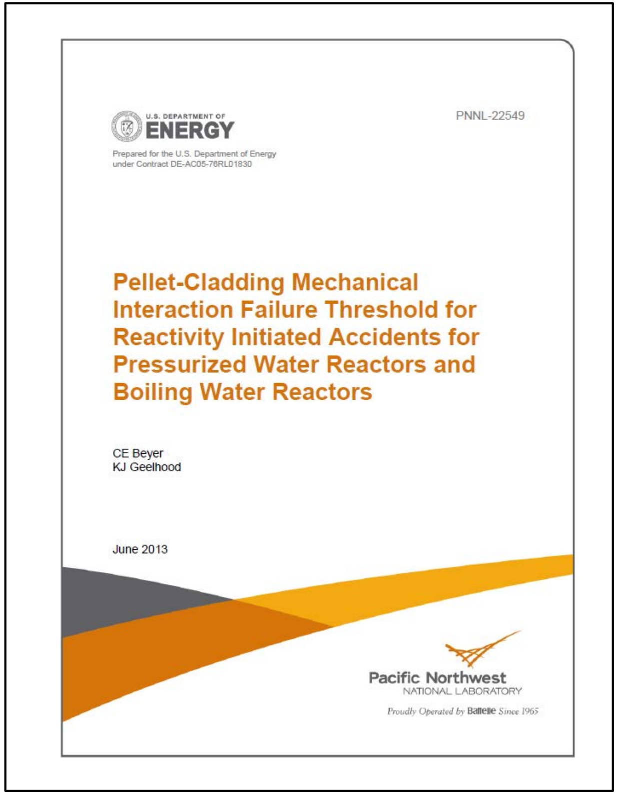
Failure Thresholds

- **EPRI Test Program**
  - Why Revision 1
    - MBT Data / NSRR corrections
  - Temperature Effects
  - Pulse Effects
  - Power History Effects
  - Hydrogen > 300ppm
  - Elongation
  - Failure Limits

http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000003002005540

Failure Thresholds

- **Best Estimate?**
  - Yes, in the sense that curve fits are relative to nominal data.
  - No, in the sense that the shape of curve fits displays negative impact in areas without failures
    - Low Hydrogen region
  - No, in the sense that correlation coefficients are in some cases substantially less than 1



U.S. DEPARTMENT OF ENERGY

PNNL-22549

Prepared for the U.S. Department of Energy
under Contract DE-AC05-76RL01830

Pellet-Cladding Mechanical Interaction Failure Threshold for Reactivity Initiated Accidents for Pressurized Water Reactors and Boiling Water Reactors

CE Beyer
KJ Geelhood

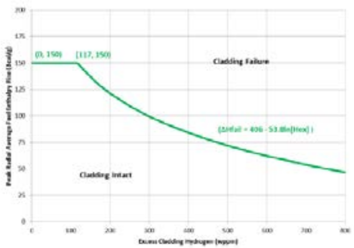
June 2013

Pacific Northwest
NATIONAL LABORATORY
Proudly Operated by Battelle Since 1965

Release Fractions

- **Appropriate Location**
  - Keep information in one place; remove from DG-1327
  - Locate to 1.183, 1.195, etc. (sign of a bigger problem)
- **Example: Changing dose method constitutes an AST/TID backfit?**

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.

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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.

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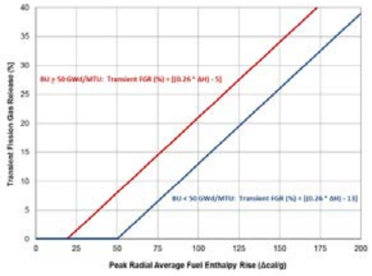
Release Fractions

- **Transient Fission Gas Release**
  - Database doesn't represent low burnup

Figure 6: Transient Fission Gas Release



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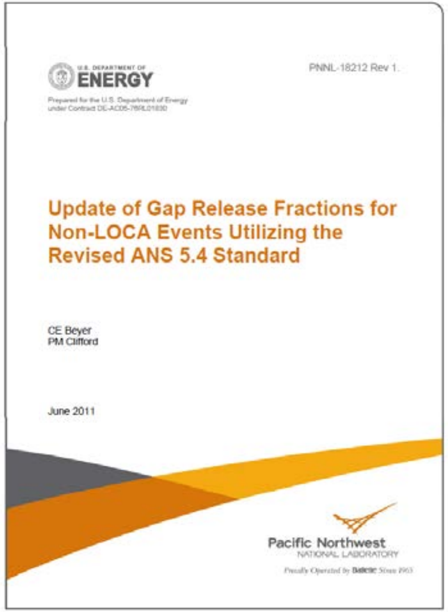

Miscellaneous

- **Logistical Issue**
  - Approved Analytical Methods must Exist
    - Method reviews in a timely manner?
- **Potential New Method Elements**
  - Transient Fission Gas Release / Mechanical aspect
  - Corrosion/Hydrogen Uptake/Crud
  - Hydride Characterization
  - FCI Impact if Centerline Melt Allowed
  - Contribution of Fission Gas Release to Pressure Surge

Questions?? / Discussion