ELECTRIC POWER RESEARCH INSTITUTE

Alternative Licensing Approaches for Higher Burnup Fuel

Scoping Study on Deterministic and Risk-Informed Alternatives Supporting Fuel Discharge Burnup Extension

2020 TECHNICAL REPORT

Alternative Licensing Approaches for Higher Burnup Fuel

A Scoping Study on Deterministic and Risk-Informed Alternatives Supporting Fuel Discharge Burnup Extension

All or a portion of the requirements of the EPRI Nuclear Quality Assurance Program apply to this product.

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Abstract

The safety and economic benefits of accident-tolerant fuel with higher burnup limits have been previously developed. One challenge to obtaining higher burnup fuel designs is the potential release of fine fuel particles (fuel fragmentation, relocation, and dispersal) under loss-of-coolant-accident conditions. A conventional strategy to address this challenge, based on fuel testing, modeling, and licensing analysis activities, imposes scheduling risks for the deployment of higher burnup fuel. An alternative licensing approach has been developed for the project that is the subject of this report based on input from industry subject matter experts (SMEs) in key stakeholder organizations (that is, utilities, fuel vendors, and independent organizations). Using a consensus process, these SMEs developed the most desirable option that is applicable for all licensees—to facilitate Nuclear Regulatory Commission review to minimize industry costs and accelerate implementation schedules. This approach limits the need for new test data and models and allows licensees to produce license submittals with minimal additional effort.

The report applies the proposed strategy in a proof of concept evaluation, which supports a generic approach that plants will be able to reference in their regulatory submittals.

Additionally, a roadmap of future actions and specific gaps for implementing the strategy are identified.

Keywords
Burnup
Dispersal
Fuel fragmentation, relocation and dispersal (FFRD)
Fuel rod burst and rupture
High burnup
Loss-of -coolant accident (LOCA)
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PRIMARY AUDIENCE: Utility fuel managers and engineers, safety analysis engineers

SECONDARY AUDIENCE: Fuel vendors, regulators

KEY RESEARCH QUESTION

This report identifies risk-informed licensing strategy options to address fuel fragmentation, relocation, and dispersal under loss-of-coolant-accident (LOCA) conditions for high burnup (HBU) fuel (62–75 GWd/MTU peak rod average burnup) for pressurized water reactors (PWRs). These options are also applicable to boiling water reactors.

RESEARCH OVERVIEW

The process used to develop the generic licensing approach covered in this report engaged key industry stakeholders' subject matter experts (SMEs)—that is, utilities, fuel vendors, and independent organizations—to reach consensus about the options that were most desirable—that is, applicable for all licensees to facilitate Nuclear Regulatory Commission (NRC) review to minimize industry costs and accelerate implementation schedules. This approach limits the need for new data and models and allows licensees to produce license submittals with minimal additional effort.

The SME panel initially evaluated five potential approaches. Of the five, three approaches (that is, options) were identified as viable based on a ranking methodology developed through SME discussions. Then, the group identified the key elements of the proposed approach (for example, analysis, calculations, and data/demonstrations) and determined how to best integrate them. Finally, a feasibility demonstration and gap analysis were performed to design a roadmap and prioritize the actions for follow-on work. This LOCA-focused scoping activity is part of a broader effort to assess HBU fuel issues.

KEY FINDINGS

Through a series of collaborative discussions during the first half of 2020, the SME panel concluded the following:

- The desired licensing basis change for the U.S. fleet should be specifically defined as follows: the fuel fragmentation, relocation, and dispersal of fuel with peak rod average burnups in the range of 62–75 GWd/MTU caused by LOCA events is of sufficiently low risk that it does not need to be included in the design-basis analyses of LOCA consequences, as currently performed to satisfy 10CFR50.46 (Title 10 Part 50.46 of Code of Federal Regulations).
- The risk-informed Regulatory Guide 1.174 process could be used to estimate the change in core damage frequency and large early release frequency with increasing burnup limits to determine whether the risk of FFRD (that is, probability and potential consequences) for HBU fuel is sufficiently small that FFRD need not be considered in deterministic LOCA analysis.

- For large-break LOCAs with application to the U.S. fleet, the approach could be supported by the application of the extremely low probability of rupture leak-before-break analysis tool, developed jointly by the Electric Power Research Institute (EPRI) and the U.S. NRC Office of Nuclear Regulatory Research (Version 2.1 of the code was released in June 2020).
- For small-break LOCAs in PWRs, the initial approach should be to demonstrate that fuel rods will not rupture or burst under realistic conditions, thereby precluding FFRD.
- The proof of concept provided in Section 4 shows that the goals for limiting change in core damage frequency and large early release frequency associated with subjecting HBU fuel to LOCA conditions may be achievable without use of plant-specific information, making it a generic approach that plants will be able to reference in their regulatory submittals (Generic Guidance Document).

WHY THIS MATTERS

The generic, risk-informed licensing approach described in this report provides an alternative strategy to address FFRD under LOCA conditions for HBU fuel (up to 75 GWd/MTU peak rod burnup) that is intended to minimize the need for new data and models.

HOW TO APPLY RESULTS

The information in this report provides the basic principles for a generic, risk-informed approach for stakeholders to expedite licensing of HBU fuel by showing that the increased risk (core damage frequency/large early release frequency) for LOCA-induced FFRD is acceptable.

LEARNING AND ENGAGEMENT OPPORTUNITIES

- The findings of this scoping study are supplemented by the experimental and modeling research conducted by EPRI's Fuel Reliability and Nuclear Fuel Industry Research programs.
- Additional work within EPRI's Risk and Safety Management program on probabilistic risk/safety assessments and its Materials Reliability Program (MRP) on extremely low probability of rupture are highly related to this scoping study and follow-on work.
- The following agencies and organizations might be interested in the results of this project:
  - U.S. NRC
  - U.S. Department of Energy
  - Domestic and international nuclear fuel vendors and utilities
  - International Atomic Energy Agency
  - Organization for Economic Co-operation and Development—Nuclear Energy Agency
  - Domestic and international nuclear research organizations

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Acronyms

ACRS	Advisory Committee on Reactor Safeguards
ATF	Accident-tolerant (advanced technology) fuel
BE	best estimate
BWR	boiling water reactor
CCF	common cause failure
CDF	core damage frequency
ΔCDF	change in core damage frequency
CFR	Code of Federal Regulations
CRAFT	Collaborative Research on Advanced Fuel Technologies for LWRs
DBA	design-basis accident
DEGB	double-ended guillotine break
DG	Draft [Regulatory] Guide
DiD	defense in depth
ECCS	emergency core cooling system
EQ	environmental qualification
EOC	end of cycle
EPRI	Electric Power Research Institute
FFRD	fuel fragmentation, relocation, and dispersal
GDC	general design criteria
HBU	high burnup

LERF	large early release frequency
ΔLERF	change in large early release frequency
LOCA	loss-of-coolant-accident
IEF	initiating event frequency
INL	Idaho National Laboratory
ISLOCA	interfacing system LOCA
LB	large break
LBB	leak-before-break
LERF	large early release frequency
LOCA	loss-of-coolant accident
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PIE	post irradiation examination
PIRT	phenomena identification and ranking table
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RCS	reactor coolant system
RG	Regulatory Guide
ry	reactor year
SB	small break
SMEs	subject matter experts
SOKC	state-of-knowledge correlation
SRM	Staff Requirements Memo
SRP	Standard Review Plan
SSCs	structures, systems and components

TBS	transition break size
TREAT	transient reactor test
xLPR	extremely low probability of rupture

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Section 1: Introduction

The nuclear industry is performing research and development to inform stakeholders of the acceptability of increasing nuclear fuel burnup limits to 75 GWd/MTU peak rod average burnup[1]. These activities apply to both current fuel designs as well as Accident Tolerant Fuel/Advanced Technology Fuel (ATF) near-term concepts. Near-term ATF cladding concepts include zirconium-based alloys with chromium or other coatings and iron-based alloys. These concepts employ uranium dioxide fuel pellets with possible initial enrichments in the 5% to 8% range and limited dopants. The research includes addressing fuel fragmentation, relocation and dispersal (FFRD) for high burnup[2] (HBU) fuel (up to 75 GWd/MTU) and post-accident fuel performance, as discussed in this report.

The current effort described in this report focuses on the regulation (that is, Code of Federal Regulations (CFR)) and alternative licensing options in the U.S. as the initial step, with follow-on work intended to address generic implications and detailed implemenation guidance.

Purpose and Scope

This report identifies alternatives to conventional fuel licensing approaches to address FFRD concerns for HBU fuel during a loss of coolant accident (LOCA). The proposed approach applies to pressurized water reactors (PWRs) and is designed as a generic, risk-informed analysis. In order to minimize the need for plant-specific analysis, the approach applies bounding input where possible to allow excluding consideration of FFRD in design basis LOCA analysis. Non-LOCA scenarios (for example, reactivity insertion accidents) which could lead to FFRD and normal operations are presumed to be addressed by other analysis. This approach can also be applied to boiling water reactors (BWRs), although the current strategy for BWRs relies on demonstrating preclusion of fuel rod burst, and thus FFRD, on a deterministic basis justified by fuel mechanical performance (not covered in this report).

[1] Throughout the report, burnup limits apply to the peak rod average (i.e. the axial average of the maximum rod) in any assembly.

[2] Throughout the report, high burnup means beyond the current regulatory limit of 62 GWd/MTU for PWRs and equivalent value for BWRs.

The identified approach reduces reliance on developing and obtaining U.S. Nuclear Regulatory Commission (NRC) acceptance of FFRD models based on existing empirical data and planned LOCA testing in the Transient Reactor Test (TREAT) Facility at Idaho National Laboratory (INL).

This work identifies and characterizes the highest priority analytical and R&D gaps (roadmap) for informing acceptability of current fuel and ATF concepts to burnups up to 75 GWd/MTU through a risk-informed approach.

Background

The Nuclear Energy Institute (NEI) ATF Working Group in the U.S. is composed of industry stakeholders that guide the industry regarding policy, research, and matters related to ATF. As part of this mission, the group is dedicated to obtaining NRC approval to extend the limit on peak rod average burnup from 62 to 75 MWd/MTU. Regulatory acceptance is needed to deploy reloads of current fuel designs and for ATF near-term concepts to these higher burnups as early as 2026.

This project supports the industry's objectives of enhancing economic competitiveness, safety, performance and reliability. EPRI has been engaged with other industry stakeholders in this effort during the past several years through publication of key reports, such as *Near-Term Accident Tolerant Fuel Gap Analysis* [4], *Accident Tolerant Fuel Technical Update: Valuation 1.0, Gap Analysis, and Valuation 2.0* [5], and *Accident-Tolerant Fuel Valuation: Safety and Economic Benefits (Revision 1)* [6]. In addition, EPRI also organizes periodic workshops and conferences on the topics of ATF and higher burnup/higher enrichment fuels to promote collaboration between all stakeholders, including U.S. and international fuel vendors, research organizations, and regulators.

Regulatory Framework

The NRC has been evaluating the significance of FFRD on accident progression for a number of years. In SECY 15-0148 [7], the staff determined that "inclusion of requirements related to FFRD in the draft final § 50.46c rule is not practicable, nor is it appropriate." SECY 15-0148 also notes that "experimental results have continued to support the hypothesis that FFRD phenomena are primarily a high burnup fuel issue and that the current licensing limits in the U.S. are adequate to prevent dispersal of large quantities of fine fuel fragments."

NRC currently limits PWR codes and analysis methods to a rod average burnup of 62 GWd/MTU with a similar limit for BWRs. This limit is frequently included in the NRC's Safety Evaluation Reports for fuel vendor mechanical design analysis methods. The 62 GWd/MTU rod pin burnup limit is described in Footnote 10 of the NRC Regulatory Guide (RG) 1.183 [8] for all light water reactors (LWRs) regarding LOCA accident release fractions. Ongoing efforts by the NRC to allow a maximum fuel rod average burnup limit of 68 GWd/MTU are described in the draft RG 1.236 [9] presented to the NRC Advisory Committee on Reactor Safeguards (ACRS) in June 2020 and in the NRC memo in Reference [10].

To permit operation of existing fuel to burnups up to 75 GWd/MTU, absence of safety-significant FFRD consequences would need to be demonstrated.

Burnup extension to 75 GWd/MTU requires consideration of effects that cross the boundaries of many technical review areas. Assessment of acceptability is needed within a few years to support industry fuel management goals, and it could require considerable effort on the part of each fuel vendor and plant licensee. To minimize regulatory and schedule risk, one consideration was to determine a viable path for addressing the FFRD consequences of a LOCA that does not require formal NRC rulemaking and minimizes deviations from NRC guidance (for example, Regulatory Guides). In 2019, the U.S. NRC drafted a project plan for preparing to review requests to allow operating to higher burnup (up to 75 GWd/MTU) and increased enrichment (up to about 8%) [11] in which it outlined the activities associated with preparing the agency to conduct its reviews of licensing submittals. The project plan states the NRC's intent to communicate with stakeholders clearly and early.

Process for Developing Alternative Approach

The process used to develop the generic licensing approach discussed in this report was designed to engage key industry stakeholders (that is, utilities, fuel vendors, and independent technical organizations) to reach consensus about the option to address the following criteria:

- Be applicable for all licensees (that is, address FFRD under LOCA conditions for all plant designs).
- Be achievable in a short NRC review time in order to minimize industry cost and implementation delay.
- Consider use of risk-informed methodology to limit the need for new data and models.
- Allow licensees to submit necessary license revisions with minimal additional effort.

The process used to develop the proposed approach is depicted in Figure 1-1.

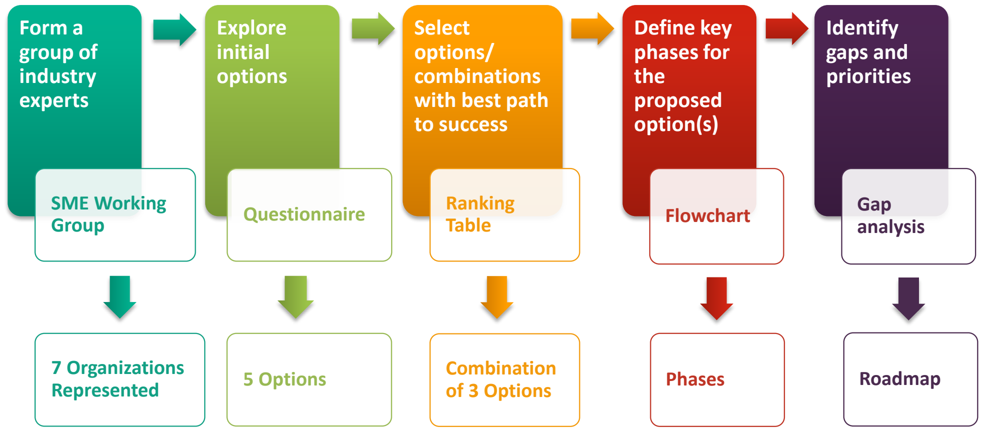

Figure 1-1
Process for Developing Alternative Approach

The first step was to form a small group of subject matter experts (SMEs), composed of industry stakeholders (that is, fuel vendors: Framatome, Westinghouse, GEH /GNF, and utilities: Southern and Exelon) and two independent organizations (EPRI and MPR Associates). The SMEs were designated by their respective organization because of their knowledge in one or more of the following areas: fuel and cladding performance, LOCA (including piping failure mechanisms and plant thermal-hydraulic response), risk analysis, and U.S. and international regulatory practices and expectations.

The following step was gathering initial inputs from the SMEs using a questionnaire to identify initial regulatory options to be evaluated. The responses to the questionnaire led to the selection of the five potential licensing approaches described in Section 3.

Then, the SME panel members independently rated each of the five options based on likelihood of success, discussed their pros and cons, and subsequently agreed on proposed approach.

Next, the SME panel identified the key elements of the proposed approach (for example, analysis, calculations, data/demonstrations). Certain aspects of the lower ranked options were identified as potentially useful, leading to discussions of how to best integrate them.

Finally, a feasibility demonstration and a gap analysis were performed to design a roadmap and prioritize the actions. The roadmap is presented in Section 2 of the report and the gaps are discussed in detail at the end of Section 4.

Section 2: Conclusions and Recommendations

The current plan for addressing LOCA-induced FFRD is based on a traditional fuel testing and modeling approach. This report describes an alternative that incorporates risk-informed insights into LOCA consequences while reducing the dependence on new testing programs and formal methods development. It is one part of parallel efforts to assess acceptability of HBU fuel.

The approach was developed by a panel of subject matter experts with the objective to facilitate timely NRC approval of higher fuel burnup limits (75 GWd/MTU). The panel, comprised of members from two utilities (Southern Nuclear and Exelon), three fuel vendors (Westinghouse, GEH/GNF, and Framatome), EPRI, and MPR Associates, evaluated five options using criteria such as meeting industry timing for going to higher burnup, reducing regulatory risk, ability to provide a satisfactory technical basis, minimizing the need for additional restrictions on operation and limiting implementation cost. The panel reached consensus on a single approach, "Risk-Informed Analysis for LOCA-Induced FFRD," to be further developed, while considering some features from the other approaches described in Section 3.

Risk-Informed Analysis for LOCA-Induced FFRD Summary

Applicability

The approach summarized in this section and described in more detail in Section 4 applies primarily to PWRs. It could also be applied to BWRs if the current approach relying on demonstrating preclusion of FFRD on a deterministic basis justified by fuel mechanical performance is not sufficient.

Licensing Basis Change

The desired licensing basis change is narrowly defined as:

> *Fuel fragmentation, relocation, and dispersal of fuel with peak rod average burnups in the range 62 to 75 GWd/MTU caused by LOCA events is of sufficiently low risk that it does not need to be included in the design-basis analyses of LOCA consequences, as currently performed to satisfy 10 CFR 50.46.*

The safety case is that the existing 50.46 design basis analysis approach/criteria remain adequate for establishing core design limits and emergency core cooling system capability because the risk-informed analysis shows the additional risk from the FFRD phenomenon is acceptably low. The potential impact on defense-in-depth and design margins is addressed as part of presenting an integrated safety case.

Methodology

If adopted for further development, the methodology is crafted to be a generic analysis and, thereby, intended to reduce duplicative regulatory activities.

To demonstrate that FFRD consequences are of sufficiently low risk that they do not need to be considered in deterministic LOCA analysis, the risk-informed RG 1.174 process [1] could be used to generically inform the relative change in overall risk. The principle of RG 1.174 is that a plant modification may be acceptable if the resulting increase in CDF and LERF is shown to be small. This methodology relies on identifying and then determining probability of events leading to FFRD to calculate the change in CDF and LERF (ΔCDF and ΔLERF), and on the demonstration that the defense-in-depth and the safety margins are maintained.

The approach would be supported by the application of the Extremely Low Probability of Rupture (xLPR) leak-before-break (LBB) analysis tool, developed jointly by EPRI and the NRC Office of Nuclear Regulatory Research [2] for Large Break- (LB) LOCAs. This tool could be used within this generic methodology to inform the probability of LB-LOCAs and that LB-LOCAs may be detected in sufficient time to allow for reactor shutdown before a reactor coolant system (RCS) piping rupture occurs. This approach facilitates the demonstration of no fuel rod burst (that is, no FFRD) and estimates the changes in CDF. If showing no rod burst for LB-LOCAs is not successful, then the risks related to the possible FFRD would be assessed (that is, determination of ΔCDF and ΔLERF). This methodology would determine the minimum break size for which an at-power LOCA causing rod burst would be plausible. Targeted LOCA calculations to quantify the expected consequences associated with FFRD would be performed for the range of plausible break sizes.

For small-break (SB) LOCAs in PWRs, it is likely demonstrable that fuel rods will not rupture under realistic conditions, thus precluding FFRD (that is, very small changes in CDF and LERF). Should SB-LOCA performance unexpectedly result in rod burst, the use of additional best estimate LOCA assumptions or an assessment of FFRD consequences could be used to demonstrate acceptable performance.

A high-level illustration of the Risk-Informed Analysis for LOCA-Induced FFRD approach is presented in Figure 2-1. A detailed description is provided in Section 4 and a summary is presented in Figure 2-2.

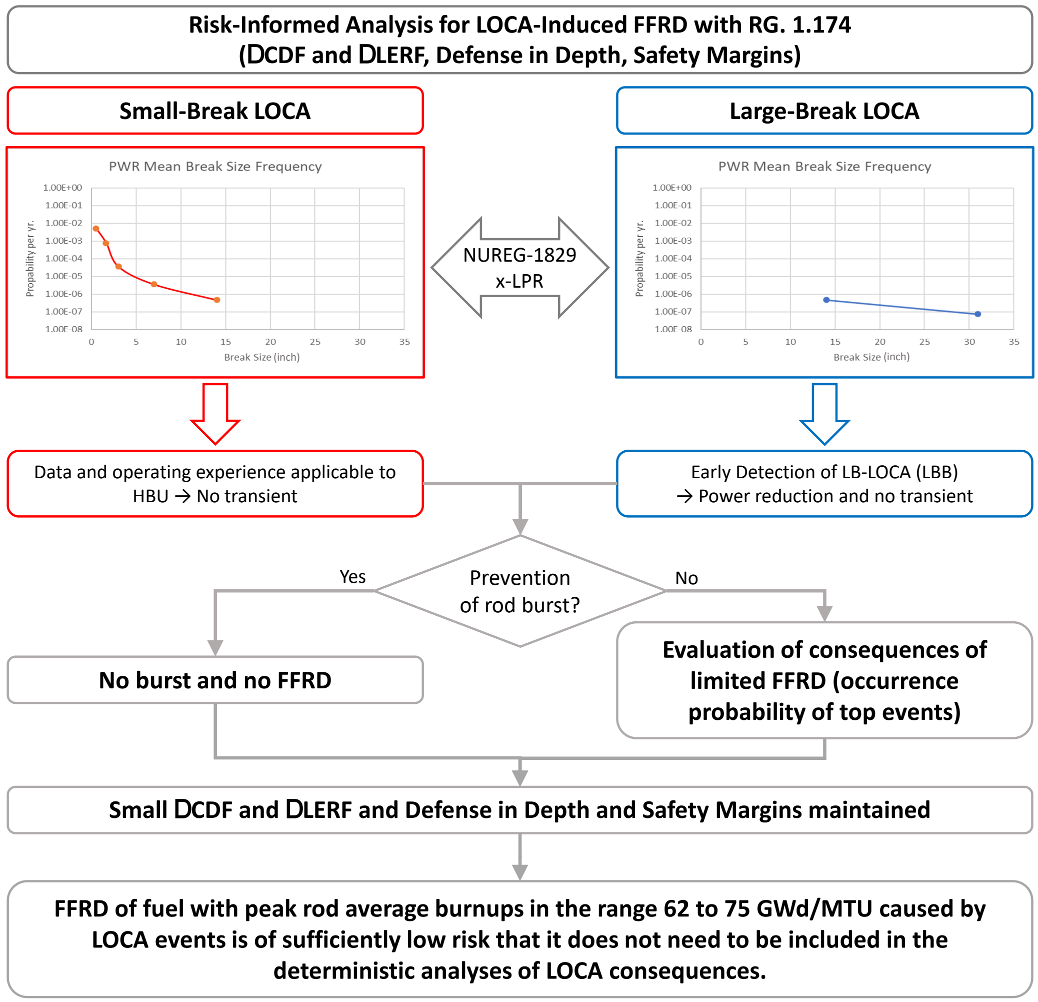

Figure 2-1
Concept – Risk-Informed Analysis for LOCA-Induced FFRD

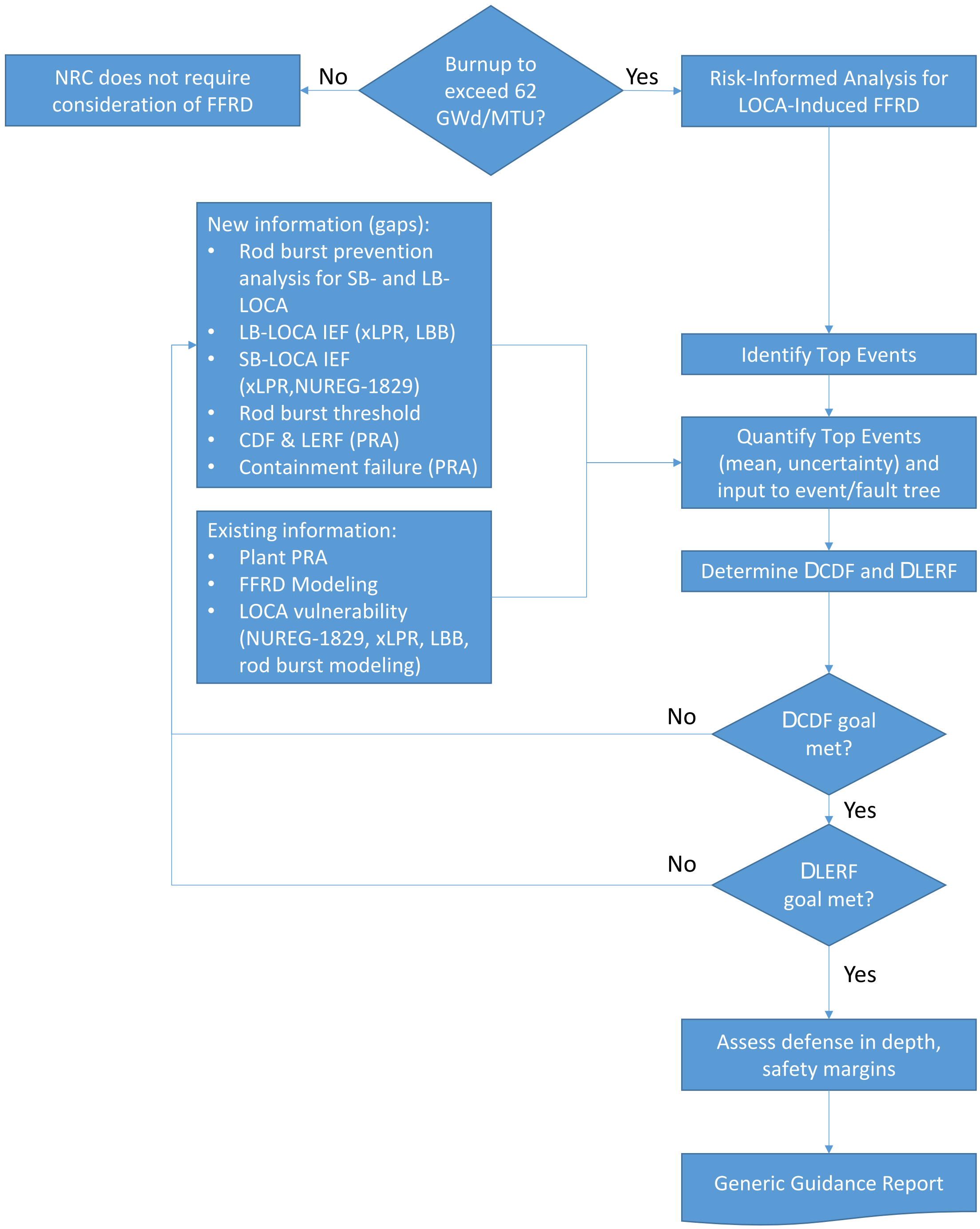

Figure 2-2
Flowchart – Risk-Informed Analysis Methodology for LOCA-Induced FFRD

Application of RG 1.174 [1] in the context of FFRD requires considering both CDF and LERF. According to NUREG-2201 [12]: "In a risk-informed regulatory context, 'core damage' is when fuel is damaged to an extent that radioactive material released from the fuel, should it escape to the environment, could significantly affect public health and safety."[3] Fuel fragmentation and relocation are precursor phenomena. Cladding burst leading to dispersal is necessary for "core damage." Therefore, prevention of rod burst is a conservative criterion for CDF and is recommended as the preferred metric. However, to provide flexibility, the recommended licensing approach considers limited FFRD. If FFRD progresses to the point of dispersal, then only LERF is relevant. Hence, on a risk-informed basis, this approach will demonstrate whether the change in CDF and LERF can be shown to be acceptably small. To be acceptable, rod burst in HBU fuel should be very unlikely (that is, negligible change in CDF) or the extent of FFRD (that is, change in LERF) should be shown to meet acceptance criteria.

To ensure that safety margins remain adequate and the risk-informed principles are consistently applied, the requirements, methodology and acceptability of a risk-informed approach are detailed in RG 1.174. Therefore, the proposed approach should allow each licensee to demonstrate that:

1. The seven NRC requirements for the defense-in-depth are met (see Table 4-2 for a list of the requirements).
2. The safety margins are maintained.
3. The ΔCDF and ΔLERF resulting from FFRD related to fuel operated up to 75 GWd/MTU and subjected to LOCA conditions is less than 10^{-7} per reactor year (ry) and 10^{-8}/ry, respectively, before inclusion of uncertainties.

As part of further development of this risk-informed alternative, generic justifications would be developed to be included in individual licensee regulatory submittals to operate up to 75 GWd/MTU (that is, HBU fuel).

The proof of concept provided in Section 4 shows that demonstrating that ΔCDF and ΔLERF goals associated with subjecting HBU fuel to LOCA conditions may be achievable without use of plant-specific information, making it a generic approach that plants will be able to reference in their submittals (Generic Guidance Document). To provide additional confidence of acceptance when including plant specific data and uncertainty, two criteria were set to serve as an example for informing this methodology:

1. A goal of demonstrating that ΔCDF and ΔLERF are less than 10% of the acceptance criteria in RG-1.174 was used.
2. To reduce potential regulatory delay and to simplify the assessment, the goal should be achievable without relying on modeling of FFRD.

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[3] Also defined as core uncovery with rapid restoration of cooling unlikely, to the point where prolonged clad oxidation and fuel damage is expected.

It is expected that this may be achievable with support of the xLPR tool to lower the predicted initiating event frequency of piping failures leading to rod burst. The methodology allows for best estimate plus uncertainty LOCA analysis in combination with allowing for operator action based on LBB for PWRs to alleviate the consequences of LB-LOCAs. Showing that no burst of HBU fuel rods occurs obviates quantification of FFRD consequences. The xLPR tool was developed jointly by the NRC and EPRI and was released to the public in June 2020 [3] (Version 2.1). The code could be used directly for this application, if NRC accepts its use to establish piping failure frequency as an input to risk analysis. The estimated time needed for this development is approximately 18 months.

The following sections present recommended actions needed to satisfy the approach, the gaps to address, and a timeline consistent with industry objectives.

Roadmap

This section discusses the recommended path forward to satisfy the approach discussed above, as well as the gaps to be addressed, and how to integrate them to support the transition to HBU fuel. The concept is to define a flexible (accommodate fuel and plant design variations) approach in enough detail that it can be consistently and generically implemented across the industry at reasonable cost with little or no regulatory burden and no operational constraints.

Figure 2-3 presents the roadmap for the proposed approach. If the approach is approved for further development, the recommended actions in the roadmap and perhaps others could be performed during a next phase, for example, as part of the Collaborative Research on Advanced Fuel Technologies (CRAFT) for Light Water Reactors program.

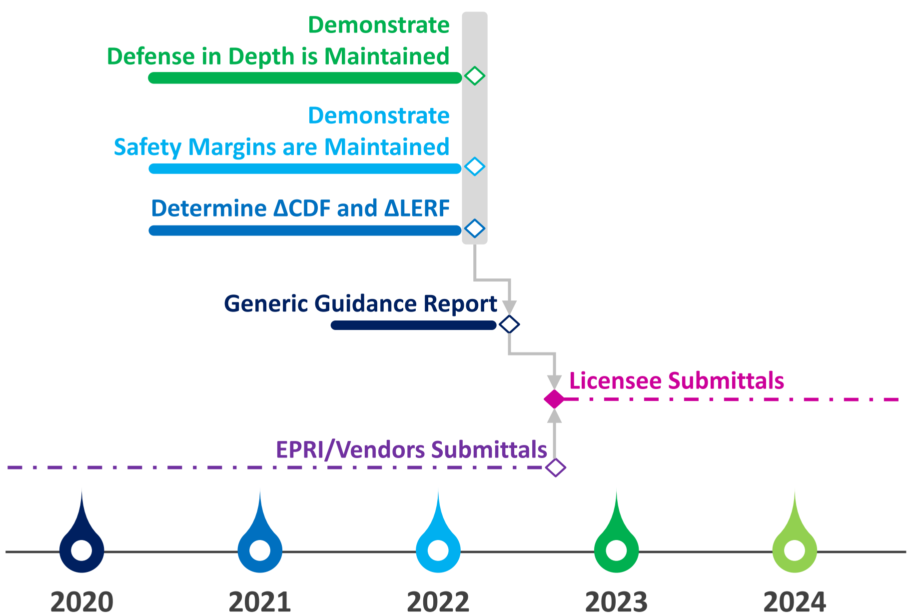

Figure 2-3
Roadmap – Risk-Informed Analysis Methodology for LOCA-Induced FFRD

Develop Generic Guidance for the Plants

The technical basis for the recommended approach described in this report will be discussed further and could be developed within the CRAFT framework as a defined, flexible, comprehensible, efficient, step-by-step methodology. If deemed feasible and useful, obtaining NRC review and approval of this generic methodology may be pursued to allow its use in combination with fuel vendor topical reports as the basis for licensing submittals.

Demonstrate Defense-in-Depth is Maintained

RG 1.174 requires demonstration that the proposed changes do not impact the defense-in-depth by showing that the seven principles and four layers prescribed by the NRC implementation of defense-in-depth are maintained. A discussion of defense-in-depth principles is provided in Section 4.2.2.

Demonstrate Safety Margins are Maintained

RG 1.174 requires demonstration that the proposed changes do not impact the safety margins.

Determine ΔCDF and ΔLERF

Determining ΔCDF and ΔLERF requires the following actions to be conducted in series:

1. Identify the phenomena contributing to rod burst and subsequently to FFRD during a LOCA for PWRs and BWRs.
2. Establish bounding or prototypical probabilistic quantification of the impact for each identified phenomenon for PWRs and BWRs.
3. Probabilistically combine them to determine a failure probability.
4. Assess uncertainty.

Address Gaps in Certain Areas

Table 4-7 summarizes the gaps identified to satisfy this regulatory approach and to strengthen/simplify the justifications. Addressing the gaps noted as “required” is mandatory for the acceptability of the approach. Addressing gaps noted “desirable” is not mandatory, however, addressing some of them would decrease the dependency on site-specific information and help reduce regulatory burden for the relevant stakeholders.

Section 3: Potential Alternative Approaches Initially Considered

This section provides a description of the five alternative approaches initially considered by the SME panel, that is:

1. Redefine LOCA Double-Ended Guillotine Break (DEGB) as Beyond Design Basis
2. Extend LBB
3. Reduce Conservatism for FFRD-Influencing Factors (best estimate plus uncertainties)
4. Risk-informed analysis for LOCA-induced FFRD through RG 1.174
5. Develop FFRD Acceptance Criteria and Analysis Models

These approaches were selected based on discussion amongst the SME panel to the questionnaire answers, provided in Appendix A.

Redefine LOCA Double-Ended Guillotine Break (DEGB) as Beyond Design Basis

LOCA-induced FFRD is primarily a LB-LOCA phenomenon. This option proposes to redefine the LOCA DEGB as beyond design basis. Doing so would allow more realistic assumptions to be used for LB-LOCA transients (for example, decay heat, power peaking, safety-system performance) that could lead to FFRD. Below a specific break size threshold, that is still to be determined, the fuel cladding temperature and the differential pressure between the inside of the rod and the RCS become sufficiently low to avoid rod burst and FFRD for HBU fuel.

This option is a version of the abandoned effort to implement a voluntary alternative to the 10 CFR 50.46 emergency core cooling regulations by identifying a transition break size (TBS), above which the conservatism of the LOCA analysis could be relaxed. Initiated by a March 2003 Staff Requirements Memo (SRM) that directed the NRC staff to estimate LOCA frequencies, 10 CFR 50.46a (“Alternative Acceptance Criteria for Emergency Core Cooling

Systems for Light-Water Nuclear Power Reactors") was to risk-inform LOCA analysis. Under this approach, LOCA sizes would be divided into two regions delineated by the TBS [13]. Smaller break sizes must continue to meet 10 CFR 50.46 criteria for design basis accidents (DBA).

NUREG-1829 ([14] and [15]) documents the results of an expert elicitation process employed to estimate frequency of various size breaks driven by mechanistic causes, and NUREG-1903 [16] evaluated the potential for rare seismic events to cause a large break.

The TBS threshold was set at a break size corresponding with a frequency of 10^{-5}/ry. Table 1 of Volume 2 of NUREG-1829 [15] shows that the initiating event frequency for the largest break is less than 2.1×10^{-7}/ry for both BWRs and PWRs. Hence, the DEGB is already below the level where the NRC would necessarily require a deterministic treatment and already below the quantitative health and safety goals for core damage and large early release frequency. However, the DEGB was implemented as a design basis accident to be a surrogate for multiple possible scenarios to ensure a reactor design has sufficient margins for undercooling transients. As such, the DEGB as the design basis accident is intertwined with many regulations that make it difficult to modify.

The generic TBS thresholds were translated to the largest pipe attached to a PWR reactor coolant loop or to a BWR reactor water recirculation system. The NRC staff required consideration of other LOCA contributors such as rare seismic events (NUREG-1903 [16]), and how to ensure the approach remained valid after implementation at a plant. The NRC developed guidance on implementation published in DG-1216 [17] for comment. In it, the staff noted that NUREG-1829 frequencies were generic, assumed that design, fabrication, inspection, repair, and so on, complied with the licensing basis and applicable codes and standards and that no major modifications were made. Age-related degradation mechanisms also had to be addressed.

DG-1216 required plant specific analysis to verify applicability of NUREG-1829. Figure 3-1 is a flowchart of the verification activity, which should demonstrate that "either the combined effects of all unique plant attributes or the effects of each individual unique plant attribute do not result in increases in the NUREG-1829 generic LOCA frequency estimates." DG-1216 required not only extensive evaluations but also maintaining them on an on-going basis to ensure changes in plant configuration or conditions did not invalidate the characteristics on which the TBS determination was based.

Neither the industry nor other commenters were satisfied with DG-1216. The industry viewed it as too complicated and costly to implement. Enclosure 1 of SECY-16-0009, dated April 13, 2016 ([18]), recommended that work on 10 CFR 50.46a be stopped, noting

> "If issued, 50.46(a) would be a voluntary rule. In a recent public meeting on [risk management regulatory framework], industry representatives at that meeting indicated that the industry would not be interested in implementing 50.46(a) (as presented to the Commission in December 2010). This is consistent with previous industry remarks."

If it were available, the erstwhile alternative ECCS criteria might resolve the FFRD concerns by allowing the largest breaks to be analyzed on a best estimate basis, but still require demonstration that FFRD does not occur during smaller than TBS LOCAs. Also, resurrecting the rule could bring considerable benefits to the industry in other areas, but only if some of the proposed NRC provisions were eliminated as previously noted by the industry. Availability of xLPR as a means to benchmark (or replace) NUREG-1829 frequency estimates might justify relaxing some aspects of the previous DG-1216. Although initial application of the TBS-like concept to FFRD could be done with license exemptions, it is possible that such activity would be interpreted as potentially setting the precedent for a change in the 10 CFR 50.46 regulations. However, the industry and NRC spent over a decade working on 50.46a without the Commission approving the new rule. Considering the need for greater regulatory certainty in future licensing submittals and review/approval process, compatibility with the licensing schedule for HBU fuel is a significant weakness for implementing this approach.

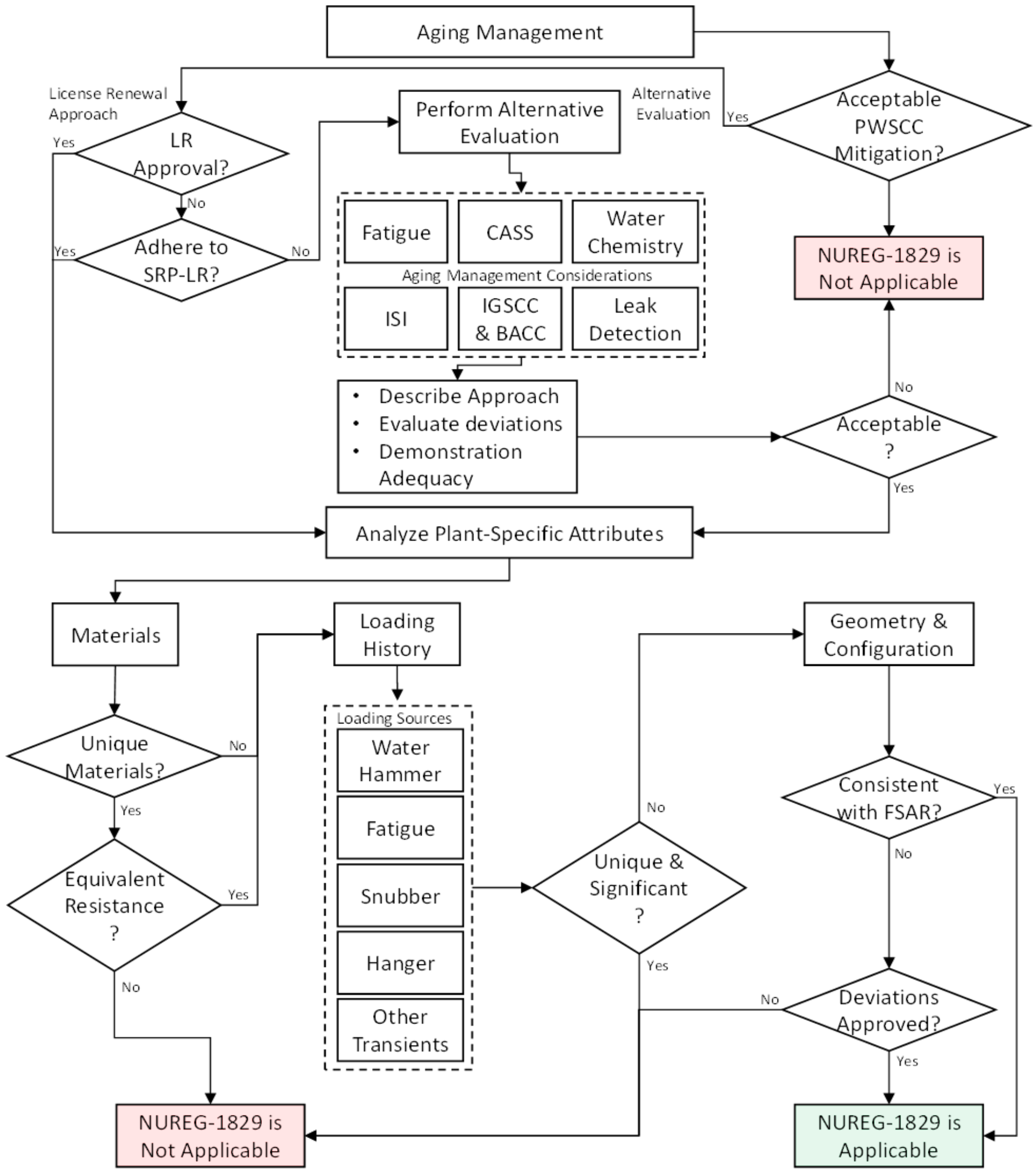

Figure 3-1
Evaluating Plant-Specific Applicability of NUREG-1829 (Reproduced from Reference [17])

Extend Leak Before Break

By applying LBB, fuel rod burst/rupture and, thus, FFRD might be precluded by taking advantage of early leak detection and mitigating actions. The regulatory framework (Standard Review Plan (SRP) Section 3.6.3 [19]) currently permits application of LBB for exclusion of dynamic effects of high energy line breaks, but not for modifying the design basis treatment of fuel degradation. However, detection of a leak while still small by the operators would allow them to implement mitigation strategies (shut down to reduce pressure and decrease decay heat) before a pipe rupture leading to LOCA conditions. Therefore, by crediting LBB to ensure the reactor is in safe shutdown prior to rupture, FFRD would be precluded.

Traditionally, the limit on deterministic LBB line size was 8 to-10 in. (about 20 to 25 cm) in diameter due to the margins required on detectable leak size and a typical plant leak detection capability of 1 gpm (about 3.8 liters per minute). However, plants have been able to apply LBB to smaller lines, at times requiring reduction of leak detection limits below 1 gpm (about 3.8 liters per minute). As illustrated in Figure 3-2 below, using this approach as a standalone would still require that break sizes below the LBB minimum break size do not result in FFRD.

In addition to the application of LBB described above, the xLPR analysis tool developed jointly by EPRI and the NRC Office of Nuclear Regulatory Research [2] might be used to inform lower LOCA event frequencies than currently assumed. The xLPR analysis tool might also be used to quantify the time between initial leakage detection and pipe/line rupture to support the conclusion that the reactor operators would have sufficient time after the detection of leakage to ensure safe shutdown and other mitigating actions. Note that LBB application to smaller break sizes might also be possible in the future based on on-going development of the xLPR tool for LBB applications.

A summary of the "Extend LBB" approach is presented in Figure 3-2.

Development of the xLPR code was initiated by EPRI and the U.S. NRC to support quantitatively assessing the LBB-approved piping system's compliance with General Design Criterion (GDC) 4 on an interval (time) basis. GDC 4 is described in Title 10 of the *Code of Federal Regulations* Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," [20] as follows:

> "*Criterion 4 – Environmental and dynamic effects design bases*. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

As previously mentioned, the regulatory framework does not currently permit application of LBB for modifying the design basis treatment of fuel degradation. This would be an impediment to using LBB informed by xLPR as a standalone strategy. However, integrated with the risk-informed alternative, xLPR is a tool that could be used to inform initiating event frequency determination over the full range of break sizes of concern. Moreover, LBB can be applied in PWR plants down to the minimum break size satisfying the detection threshold.

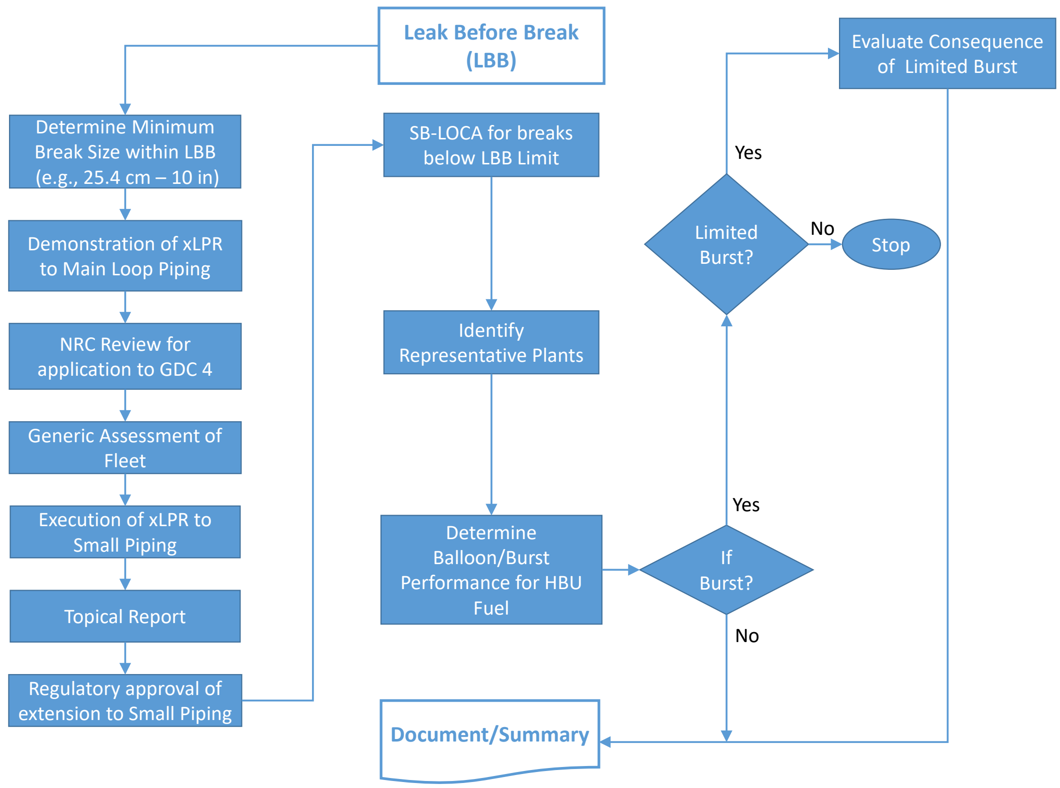

Figure 3-2
Flowchart Summary of the Extend LBB Approach

In the context of this work, this approach was evaluated both as a standalone strategy and as a tool to support other approaches. The assessment presented in Subsection 3.6 of the section showed that the existence of significant risks related to regulatory acceptance, schedule, and approval of first of a kind applications make this approach challenging to apply as a standalone licensing methodology.

Reduce LOCA Analysis Conservatism for Factors Influencing FFRD

The objective of this approach is to identify factors that impact LOCA-induced FFRD and informing the applicability of those factors using a best estimate plus uncertainties methodology. RG 1.157 [21] provides an alternative to the Appendix K analysis requirements and has been used for some reactor designs. For example, many parts of NRC DEGB guidance have their own conservatism or assumptions that might be modified (or departures justified) to sufficiently ameliorate conditions to inform FFRD propensity without need for rulemaking.

RG 1.157 [21] also indicates that best estimate fuel models could be considered to refine the technical basis associated with pellet-cladding mechanical interactions in fuel densification and rod burst phenomena under LOCA conditions.

This approach may require significant changes of analysis and methodologies depending on the scope. The specific set of modified assumptions that would eliminate FFRD concerns would need to be developed and new LOCA analyses performed for the applicable portion of the fleet. While this option could be used in support of a generic approach (for example, to demonstrate that HBU fuel rod burst and thus FFRD is not a likely event and that the consequences are minimal when considering the underlying event more realistically), it is not considered a viable option as a standalone approach.

Risk-Informed Analysis Methodology for LOCA-Induced FFRD

This approach uses the risk-informed process, detailed in RG 1.174 [1], to generically determine the contribution of LOCA-induced FFRD to plant risk for extension of the peak rod average burnup limit from 62 to 75 GWd/MTU. Using the described methodology to demonstrate LOCA-induced FFRD does not contribute to overall plant risk could be the basis for seeking NRC approval to extend current practice for lower burnup fuel of not including FFRD in design-basis LOCA analysis performed to 10 CFR 50.46 criteria. The generic process has sufficient flexibility to address individual plants by accounting for design and plant specific parameters, allowing most development to be done on an industry-wide basis without constraining specialization needed to minimize deviations from an individual plant's design basis. While this report mainly details application of this approach to PWRs, it could also be used for BWRs, if needed.

A detailed description of the risk-informed process provided in RG 1.174 is presented in Section 4.

FFRD Acceptance Criteria and Analysis Models

The NRC regulatory position on FFRD is described in SECY-15-0148 [7] which credits the currently approved fuel burnup limits as the primary barrier for preventing FFRD. This regulatory framework is largely based on LOCA tests performed at the Halden Reactor project and the Studsvik Hot Cell facility and is conservatively based on burnup alone. An overall review of the FFRD phenomena shows that fuel temperature and temperature profiles, fuel pin pressure, fuel pellet dopants, pre-transient power and burnup have some degree of influence on FFRD performance. The U.S. Department of Energy national laboratories and EPRI have been participating in the development of a fundamental model of FFRD behavior, based on the use of advanced modeling and simulation methods to complement these existing experimental datasets.

Additionally, a series of LOCA tests is being planned for the INL TREAT test reactor. These tests will rely fully on internal heating of the fuel pins from nuclear fission-induced heat. This will result in more realistic fuel temperature profiles than were possible in the Halden or Studsvik tests. The Halden and

Studsvik tests relied to some extent on external heating (Halden ~50%, Studsvik 100%), so the pellet temperature distribution was not fully representative of in-reactor LOCA transient conditions. Advanced modeling and simulation activities will be used to help develop the TREAT tests and subsequently model the test results so that fuel suppliers can incorporate these results into their topical reports and LOCA analysis tools and methods. In addition, models of fuel dispersal throughout the RCS and containment are also under development. These models would be used for evaluation of the radiological consequences of limited amounts of HBU fuel rod burst/rupture and resulting fuel fragment dispersal. The TREAT test will be fully instrumented so data will be obtained on fuel dispersal as well as select fission product releases.

The TREAT test facility provides a unique capability. But these tests may not be completed prior to 2024 (especially for ATF fuel segments exposed to higher burnups), and the number of tests will be somewhat limited. This is due in part to the current moratorium and the 2019 Idaho Nuclear Clean-up and Research Agreement between the state of Idaho and the U.S. Department of Energy governing the transportation of irradiated fuel into the state of Idaho. INL expects this moratorium to be lifted in 2021 to support tests of current higher burnup fuel technologies in 2023 and 2024, with testing of higher burnup ATF fuel in 2025 and beyond. Additionally, INL plans to change-out the core internals of the Advanced Test Reactor in 2021. This reactor will be used to establish the pre-test power conditions prior to testing in TREAT. With these restrictions, the tests will not begin until 2023 with completion in 2024 at the earliest.

This approach is the baseline approach that industry is proceeding, following the traditional method of fuel behavior R&D, deterministic analysis-based licensing, and plant-by-plant implementation.

Rating Criteria and Ranking of the Potential Options

This section provides a description of the process used by the SME panel to rate each of the five alternatives based on their ability to satisfy several rating criteria. This section also presents the key steps of the proposed approach (for example, analysis, calculations, data/demonstrations) identified by the group, a proof of concept and a gap analysis.

Following the identification of the five potential licensing alternatives described above, the SME panel discussed differentiators that could be used to assess the desirability of each and then proceeded to rank them:

1. First, the panelists agreed on six criteria considering the needed timing, the level of effort and cost to implement, and ability to provide technical bases satisfactory to the NRC (that is, reduce regulatory risk).
2. Based on perceived relative importance of these criteria, the panel assigned a weight to each of them. Table 3-1 presents the criteria, as well as their definition, meaning of the ratings, and their weight.

3. Then, each organization independently provided a score (0, 1, 2 or 3) and rationale for each option that represented their organization's perspective (that is, there was no restriction on discussing pros and cons with their own organization).
4. Each option received a score corresponding to the sum of the weight times the rating for each criterion, as described in the equation below.

$$Score_{Option} = \sum_{Criteria}[Weight \times Rating] \qquad \text{Eq. 3-1}$$

Finally, the results from each of the seven organizations (that is, three fuel vendors, two utilities, EPRI, and MPR) were totaled, with the best option having the highest score. This selected option then became the focus of the SME panel to refine the licensing approach described in the next sub-section. However, the panel continued to consider the utility of the lower ranked options, subsequently concluding that aspects of some of them would be beneficial as part of the selected approach.

The results of ranking are presented in Table 3-2 and Table 3-3.

- Risk-Informed Analysis Methodology for LOCA-Induced FFRD was ranked highest by five of seven of the SME panel organizations and second by the remaining two. The rationale was that it was least dependent on developing new technical bases and addresses risk factors associated with the TREAT test plans (schedule and regulatory acceptance) while providing a generic but flexible methodology across the fleet. This option also limits the extent of new or revised analysis relative to other options being considered.
- The approach "FFRD Acceptance Criteria and Analysis Models" was ranked second on average. It corresponds to work currently pursued by the industry and research community. If sufficiently developed in time, portions of this work (for example, quantification of fuel dispersal) can be used to strengthen the defense-in-depth aspect of the recommended approach.
- The other three options, "Redefine LOCA DEGB as BDBA," "Extend LBB," and "Reduce Conservatism for FFRD-Influencing Factors," were each rated zero for one or more criteria by one or more of the representatives, showing that they were not considered viable as standalone approaches. The main reasons were related to:
  - Incompatible implementation schedules ("Redefine LOCA DEGB as BDBA" and "Reduce Conservatism for FFRD-Influencing Factors").
  - Regulatory risk ("Redefine LOCA DEGB as BDBA" and "Reduce Conservatism for FFRD-Influencing Factors").
  - Limited plant design or accident scenarios application ("Extend LBB").
  - Prohibitive cost ("Redefine LOCA DEGB as BDBA").

  However, "Extend LBB" and "Reduce Conservatism for FFRD-Influencing Factors" present benefits to the proposed approach such as defense-in-depth, as will be discussed in the following sub-sections.

Table 3-1
Rating Criteria

Criteria	Definition/Instructions	Rating	Weight
Need for extensive regulatory changes	Though individual license changes can be approved as exemptions, they will lead to need for rulemaking (that is, changing 10 CFR 50) which is a lengthy process that would increase risk of not meeting schedule. Deviations from NRC guidance (for example, Regulatory Guides, Interim Staff Guidance) do not require rulemaking but require stronger justification.	*High (3):* Can implement within existing regulatory framework *Medium (2):* Can implement via exemptions *Low (1):* Multiple exemptions/deviations from guidance in multiple NRC branches needed or counter to significant regulatory precedent *0:* Needs for regulatory changes too extensive	9
Addresses pellet fragmentation	In case prevention of burst is not sufficient, the vulnerability to fragmentation may be separately evaluated.	*High (3):* Precludes FFRD *Medium (2):* Moderate FFRD but acceptable consequences *Low (1):* Widespread FFRD *0:* Unacceptable FFRD	8.5
Schedule	The planned schedule to develop the generic alternative and obtain NRC review and acceptance (to support completion of fuel vendor and licensee actions by 2026) is reasonable and achievable.	*High (3):* Significantly early (years) completion beneficial *Medium (2):* Planned schedule has little margin *Low (1):* Planned schedule assumes success *0:* Planned schedule does not meet need	8
Addresses rod burst	Currently available and planned data and modeling/simulation capability is sufficient to justify the regulatory alternative. Even with currently planned testing, the amount of data is limited, and the timing for new data is subject to technical, budgetary, and political factors.	*High (3):* Available data and models sufficient *Medium (2):* Currently planned work will provide sufficient basis *Low (1):* Significant risk that data/model availability will not provide sufficient technical justification *0:* Data/model do not provide justification	7
Breadth of applicability and cost*	In case prevention of burst is not sufficient, the vulnerability to fragmentation may be separately evaluated.	*High (3):* Precludes FFRD *Medium (2):* Moderate FFRD but acceptable consequences *Low (1):* Widespread FFRD *0:* Unacceptable FFRD	6.5
Need for additional data and models	The planned schedule to develop the generic alternative and obtain NRC review and acceptance (to support completion of fuel vendor and licensee actions by 2026) is reasonable and achievable.	*High (3):* Significant (years) early completion is beneficial *Medium (2):* Planned schedule has little margin *Low (1):* Planned schedule assumes success *0:* Planned schedule does not meet need	5

*A sensitivity study was performed regarding the breadth of applicability and cost criteria that showed that attributing a higher weight (that is, 8 or higher instead of 6.5) would not change the overall ranking of the alternatives.

Table 3-2
Ranking Summary by SME Entity and in Average (1 is best)

Options	EPRI	Framatome	GEH – GNF	Westinghouse	Southern	Exelon	MPR	Average	Average wo Outliers
Redefine LOCA DEGB as BDBA	4	5	4	2	4	3	5	4	4
Extend LBB	1	4	5	3	2	4	3	3	3
Reduce Conservatism for FFRD-Influencing Factors	4	1	3	5	5	5	4	5	5
Acceptability of FFRD through RG 1.174	1	2	2	1	1	1	1	1	1
FFRD Acceptance Criteria and Analysis Models	3	3	1	4	3	2	2	2	2

Table 3-3
Option Ranking Average Scores

Options	Average Scores (%)*
Redefine LOCA DEGB as BDA	17
Extend LBB	19
Reduce Conservatism for FFRD-Influencing Factors	14
Acceptability of FFRD through RG 1.174	28
FFRD Acceptance Criteria and Analysis Models	22

*The average score percentage corresponds to the weighted rating distribution attributed by the panel.

Section 4: Risk-Informed Analysis Methodology for LOCA-Induced FFRD

This section presents the key steps of the proposed approach (for example, analysis, calculations, data/demonstrations) identified by the SME panel and provides a proof of concept and gap analysis. Addressing LOCA-induced FFRD is only one part of the overall confirmation required to demonstrate acceptability of operating fuel to higher burnups. Other actions include showing that HBU fuel satisfies other fuel design limits for normal operation and other transients, assessing the need to adjust source term for the larger inventory of long-lived fission products, and demonstrating acceptability of FFRD for other accidents.

Description of the Approach

Installing a core reload with the intent to operate to burnups beyond the current licensing limit is a modification requiring a license amendment. In accordance with RG 1.174 [1], a plant modification may be shown to be acceptable if the changes it causes in CDF and LERF are demonstrated to be small. A licensee may perform a probabilistic assessment of the effect on CDF and LERF of implementing the plant modification. The acceptable values for a change in CDF and LERF depend on the current mean values in comparison to the change (for example, Figure 4-1 below for LERF, which is Figure 5 from RG 1.174). If no rod burst is the success criterion, LERF need not be calculated, but defense-in-depth leads to the need to also consider low probability consequences of FFRD.

To minimize dependence on FFRD modeling and analytical effort, a graded approach is adopted by working to one of two constraints:

- The preference is to avoid the assessment of FFRD consequences altogether by showing that the probability of rod burst of HBU fuel is sufficiently low that FFRD does not need to be considered. Thus, if burst of HBU fuel rods can be shown to not occur (that is, have a very low probability) and defense-in-depth is maintained, no further justification of acceptability of FFRD in HBU fuel during LOCA should be needed. Only fuel having burnups from 62 to 75 GWd/MTU is relevant, because rod burst of lower burnup, high power fuel has already been accounted for in the existing regulatory framework and thus is not a factor in the RG 1.174 analysis.

- If HBU fuel burst cannot be analytically precluded, then justification relies on limiting the extent of FFRD, with acceptability based on a low probability of a LOCA leading to an unacceptable extent of FFRD. The allowable amount of FFRD depends on considerations such as release of radioactivity to the reactor coolant system and containment and would be determined as part of the next phase of development, if the proposed approach is adopted.

Thus, the goal for the risk analysis is:

> *Fuel fragmentation, relocation, and dispersal of fuel with peak rod average burnups in the range 62 to 75 GWd/MTU caused by LOCA events is of sufficiently low risk that it does not need to be included in the design-basis analyses of LOCA consequences, as currently performed to satisfy 10 CFR 50.46.*

This is a justification for the extension of current practice to 10 CFR 50.46 analysis of lower burnup fuel (less than 62 GWd/MTU), which focuses on cladding ductility, coolable geometry, and hydrogen production while not explicitly modeling or establishing limits for FFRD.

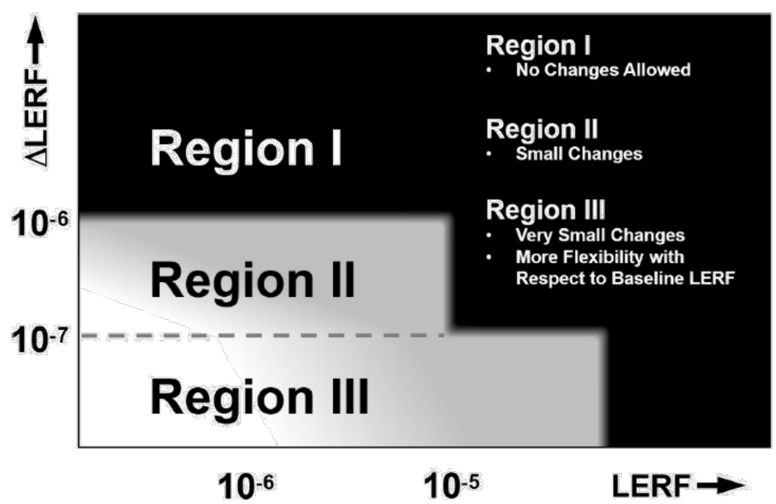

Figure 4-1
Acceptance Guidelines for LERF (Figure 5 from RG 1.174) [1]

Table 4-1
Graded Application of Success Criteria in Determining Risk of FFRD

Success Criteria in Order of Preference
No damage of HBU fuel (that is, no rod burst): FFRD precluded
Limit extent of HBU fuel FFRD: assess consequences of limited FFRD during next phase

For ΔCDF, the goal is to remain below 10^{-7}/ry, which is one-tenth of the threshold below which RG 1.174 states the change is considered regardless of whether there is a calculation of the total CDF.

Figure 4-1 graphically shows the range of LERF values for which the NRC considers the risk-informed RG 1.174 process may be used to demonstrate acceptability based on minimal increment in risk.[4] For example, if the LERF for a given plant were $2x10^{-6}$/ry (roughly the average for US power reactors), then a change resulting in a ΔLERF of up to 10^{-6}/ry would be acceptable. However, a plant with a LERF of $2x10^{-5}$/ry is in Region III and can only accept a ΔLERF of up to 10^{-7}/ry. In developing this generic risk-informed methodology, the goal is to show that a ΔLERF of 10^{-8}/ry is possible without use of plant-specific information. The factor of ten margin should accommodate on-going refinement of this approach, allowance for uncertainties, and assuage concerns for plants that do not have a detailed Level 2 PRA. Figure 4-1 is not intended to have precise boundaries for acceptability and other criteria must be addressed, as described later.

The results of this approach are compared to the results of a full-scope PRA. Therefore, the approach requires the existence of an appropriately performed and reviewed (for example, meets consensus standards, peer-reviewed), plant-specific PRA. For those plants without a detailed Level 2 PRA, the options are: 1) use the no burst success criterion to avoid need to determine ΔLERF, 2) apply an existing containment fault model, or 3) calculate the ΔCDF and then determine ΔLERF using the generic values.

RG 1.174 allows the determination of ΔLERF without necessarily exercising the full PRA model:

- RG 1.174 paragraph 2.3.3 – For applications like component categorization, sensitivity studies on the effects of the proposed licensing basis change may be sufficient.
- RG 1.174 paragraph 2.4 – When an increase in LERF is very small (that is, less than 10^{-7}/ry), "the change is considered regardless of whether there is a calculation of the total LERF."
- RG 1.174 paragraph 2.5.2 – "If the calculated values of ΔCDF and ΔLERF are very small, as defined by Region III in Figures 4 and 5 [of RG 1.174], a detailed quantitative assessment of the base values of CDF and LERF is not necessary."

Thus, either of two techniques can be used to determine ΔCDF and ΔLERF:

1. Use the full plant PRA model by establishing new success criteria and modeling the plant events and conditions affecting their frequency of occurrence.
2. Performing a standalone estimate of ΔCDF and ΔLERF using only the success criteria and events relevant to the burnup limit extension.

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[4] As some plants may not have a detailed Level 2 PRA, provisions must be made for an alternative to LERF, as discussed below.

The full PRA technique provides the opportunity for more integrated understanding of sensitivities and avoids the need to maintain a separate assessment but nonetheless requires considerable effort and complicates the formal plant PRA. The standalone technique is a much lower effort and quickly accomplished. Not referencing a specific plant PRA also opens the possibility of justifying generic, fleet-wide acceptance of a LOCA-induced FFRD exclusion without individual plant analyses and licensing reviews. Because the assessment in this report is generic, the feasibility evaluation uses the standalone technique. Further development should investigate how to perform a generic, standalone determination that could be referenced by individual plants without the need for plant-specific calculations.

The remainder of this section describes the application of the recommended licensing strategy. While this presents a conceptual framework, some detailed technical discussion is provided to assist in understanding the feasibility of the approach, to justify the approach on a fleet-wide basis, and to facilitate actual application to specific plants following further development. The values used are intended to be plausible, but inputs must be formally justified in the future as part of detailed development of the generic risk impact.

Although the licensing approach is risk-informed and requires determination of the impact on CDF and LERF, much of RG 1.174 (and the next few sub-sections of this report) deals with traditional safety topics such as defense-in-depth. This focus on deterministic elements is to ensure that safety margins are not eroded by application of PRA. To evaluate the feasibility of the proposed approach, the following discussion considers the activities and outcomes necessary to support an integrated safety case that is expected to address regulatory guidance for a risk-informed justification (Figure 4-2). The generic risk determination process is shown in Figure 2-2 and can be summarized as:

1. Narrowly define the applicability of the analysis to limit the scope of regulatory issues.
2. Perform calculation of ΔCDF and ΔLERF; the key steps of which are:
   a. Identify parameters that contribute to CDF in a LOCA and the additional parameters that determine LOCA-induced FFRD impact on LERF.
   b. Assess the state of knowledge (that is, ability to justify) for quantifying the parameters in order to select those with the firmest bases. This means that analyses performed for different designs or at different times may use different combinations and different values.[5]
   c. Calculate the ΔCDF and ΔLERF and compare to NRC guidance for allowable change.

[5] An example of this flexibility is use of LBB to justify crediting operator action for PWRs vs. taking credit of the rod burst resistance of BWR fuel.

3. Assess the potential impact on deterministic safety margins and on defense-in-depth by addressing RG 1.174 expectations.
4. Prepare a regulatory submittal to operate up to 75 GWd/MTU that addresses the information specified in RG 1.174. As previously noted, the recommended licensing approach may be pursued on a fleet-wide basis but is suitable and described for individual plant implementation.

Elements of Risk-Informed Process

RG 1.174 identifies expectations for an integrated analysis of considerations in addition to quantifying ΔCDF and ΔLERF, as described next.

Risk-informed, plant-specific decision-making consists of principal elements (Figure 3 of RG 1.174, shown below as Figure 4-2) that must be addressed in an integrated manner as part of the use of RG 1.174.

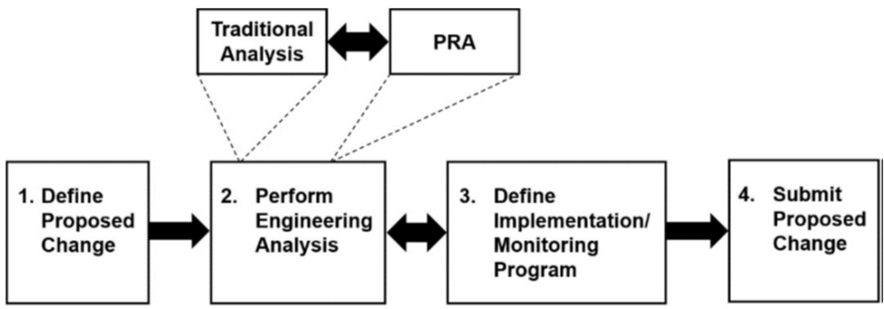

Figure 4-2
Principal Elements of Risk-Informed, Plant-Specific Decision-Making [1]

Element 1: Define the Proposed Change

The licensing bases change is to permit fuel assemblies to be used up to a peak rod average burnup of 75 GWd/MTU. As stated above, the proposed objective of this risk-informed assessment of FFRD is:

> *Fuel fragmentation, relocation, and dispersal of fuel with peak rod average burnups in the range 62 to 75 GWd/MTU caused by LOCA events is of sufficiently low risk that it does not need to be included in the design-basis analyses of LOCA consequences, as currently performed to satisfy 10 CFR 50.46.*

As shown in Table 4-1, the preferred success criterion is that prevention of rod burst (core damage) allows exclusion of FFRD. If rod burst cannot be ruled out, then it must be demonstrated that the extent of FFRD is limited to avoid secondary defense-in-depth issues such as increased source term or adversely affecting long-term operation of safety-related equipment. An objective is to show that the safety categorization of structures, systems, and components (SSCs) per 10 CFR 50.69 does not need to be modified, avoiding the complication that might have on risk determination.

A summary of the key activities to implement the risk-informed analysis methodology for LOCA-induced FFRD is presented in Figure 2-2.

Element 2: Perform Engineering Analysis

The following is a generic evaluation to serve as an example of how to address Element 2 shown in Figure 4-2. It might be developed further to use fleet-wide, or allow each plant to customize it to best match its current design basis, while showing that risk of burst and thus FFRD of HBU fuel is sufficiently low to allow it to be excluded from the design basis. As specified in RG 1.174, a separate evaluation of defense-in-depth and safety margins is performed to demonstrate acceptable impact of the proposed licensing basis change on the functional capability, reliability, and availability of affected equipment. (The organization of the following section follows the RG 1.174 organization.)

Defense-in-Depth

The purpose of this section is to demonstrate that defense-in-depth is maintained. Back up for the risk-informed calculation of CDF and LERF is required by showing availability of other mitigation measures.

Currently, FFRD is not required to be addressed for the design-basis accident analyses based on imposition of peak rod average burnup limits. The desire to extend allowable fuel burnup to 75 GWd/MTU has raised questions from the NRC regarding the increased vulnerability to and magnitude of FFRD. Uncertainties in modeling FFRD may result in several undesirable side effects. A major reason for this evaluation of regulatory alternatives was to determine how to lessen schedule and regulatory risk associated with the current plan discussed in Section 3.5. Evaluating ΔCDF and ΔLERF on the basis that satisfying no rod burst prevents FFRD strengthens the assertion that defense-in-depth is maintained which, in turn, supports the assertion that functionality, reliability, and availability of safety-related SSCs remain acceptable. Therefore, no compensatory or programmatic measures are required. However, as part of an integrated safety case, the risk-informed process requires demonstrating that adequate defense-in-depth remains available, even should some FFRD occur.

Exclusion of FFRD from the design-basis should not reduce the redundancy, independence, or diversity of systems. Showing no effect on common cause failure (CCF) is straightforward if rod burst does not occur. Susceptibility to human error is not affected. Finally, the approach continues to meet the plant's design criteria.

Although no rod burst is the preferred approach, allowing limited rod burst is a possible alternative. A low limit on the extent of rod burst would be established so that reliability of long-term safety-related functions is maintained. As to maintaining multiple fission product barriers where rod burst is not precluded, extension to 75 GWd/MTU may cause some HBU rods to burst but it does not significantly increase the failure probability of any barrier because low to moderate burnup, high power fuel rod burst exceeds that of HBU fuel.

RG 1.174 identifies that defense-in-depth is comprised of four layers:

1. Robust plant design to survive hazards and minimize challenges that could result in an event occurring – this layer is not affected. The approach takes credit for existing measures that minimize the likelihood of the LOCA initiating event. Most of the contribution to the low ΔCDF and ΔLERF comes from the improbability of a LB-LOCA. For simplicity, a "new" initiating event (that is, LOCA leading to FFRD) is evaluated, but actual likelihood of a LOCA is not altered.
2. Prevention of a severe accident if an event occurs – working to a no burst criterion is consistent with this defense. If FFRD of HBU fuel occurs, it is a complication of core damage that occurs after safety system (ECCS and containment isolation) initiation and, therefore, does not affect SSC availability. For longer term safety functions (for example, recirculation), the limit on extent of HBU FFRD effects must be set low enough to meet acceptance criteria (for example, such as dose for equipment environmental qualification (EQ)).
3. Containment of the source term if a severe accident occurs – a LOCA with no rod burst or limited FFRD is not a severe accident.
4. Protection of the public from releases of radioactive material (for example, through siting in low-population areas and the ability to shelter or evacuate people, if necessary) – the amount of FFRD allowed should be set low compared to the release in a severe accident.

The proposed exclusion of FFRD from the LOCA design-basis does not adversely affect the seven defense-in-depth considerations identified by the NRC in RG 1.174 and described below and in Table 4-2. The "Will Meet?" column shows the expectation that, following closure of the required gaps listed in Table 4-7, each of the layers is expected to be satisfied. Note that the first consideration consists in maintaining the balance among the four layers discussed above.

Table 4-2
NRC's Seven Principles for Maintaining Defense-in-Depth

Defense-in-Depth Consideration	Will Meet?	Discussion
Preserve reasonable balance among layers	Yes	• Robust plant design: No change • Prevention of severe accident: a LOCA with no rod burst or limited FFRD is not a severe accident • Containment source term if severe accident occurs: a LOCA with no rod burst or limited FFRD is not a severe accident. • Protection of public: No change
Preserve adequate capability of design features without over reliance on compensatory measures	Yes	• Does not affect capability of design features, so no compensatory measures are needed. • Programmatic activities (for example, quality assurance, testing) are unchanged • Show radiation dose assumptions for deployment of FLEX are not exceeded
Preserve system redundancy, independence, and diversity	Yes	• Change does not degrade safety-related system redundancy, independence, and diversity. o No change if no rod burst o Otherwise, show fission product release to containment is low compared to design basis
Preserve adequate defense against potential CCFs	Yes	• No change if no rod burst; otherwise o FFRD dose needs to be shown within the design-basis accident dose envelope o Amount of material released to RCS or containment is limited and chemically unlikely to clump so potential to adversely affect flow is low
Maintain multiple fission product barriers	Yes	• No change if no rod burst criterion met, otherwise: o Cladding: HBU fuel burst less likely than high power fuel. o RCS: Not applicable – Initiating event is a breach o Containment: Isolation complete before FFRD; longer term effects such as fission product release to containment is bounded by the design-basis accident acceptance criteria.
Provide sufficient defense against human errors	Yes	• No change – operator response in LOCA should be consistent with human reliability assessment best practices to avoid increased vulnerability to human errors.
Continue to meet intent of plant design criteria	Yes	• With exception of extending burnup limit from 62 to 75 GWd/MTU, design limits, technical specifications, and acceptance criteria are not expected to need to be changed to accommodate LOCA-induced FFRD.[6]

[6] Design limits, technical specification, etc. may need to change because of analyses of other conditions associated with HBU fuel.

Safety Margins

No change to LOCA codes, standards or to LOCA safety analysis acceptance criteria are proposed. If the no rod burst criterion is met, then safety margin to fuel damage is maintained. If the extent of burst or if FFRD criteria are applied, limits on the magnitude minimize impact on safety margins. Although some small amount of additional radioactivity may escape the fuel early in the transient, the effects of LOCA-induced FFRD on safety-related SSC performance and off-site consequences will be shown to remain within current analysis.

Element 3: Define Implementation and Monitoring Program

The proposed approach would extend the fuel burnup limit from 62 to 75 GWd/MTU without explicitly including the FFRD phenomenon in the design-basis safety analysis. As such, there would be no new operational monitoring requirements. For reasons other than LOCA-induced FFRD, the current practice of core loading plans, complying with a burnup and heat generation limits, would continue. The objective is to consider those limits in evaluating risk of HBU fuel rod burst and FFRD without imposing any limits specific to FFRD.

A licensee will need to be mindful of implementing other changes that could increase the susceptibility of HBU fuel to rod burst or FFRD.

Element 4: Submit Proposed Change

The change is to allow operation of fuel assemblies to burnups up to 75 GWd/MTU (versus current limit of 62 GWd/MTU) by addressing LOCA-induced FFRD on a risk-informed basis that demonstrates it may continue to be excluded from design-basis analyses.

This evaluation provides an initial framework to support a generic evaluation of LOCA-induced FFRD. If adopted by the industry, the completion of the generic analysis will be performed as a subsequent activity and will include guidance on how individual licensees demonstrate they are consistent with it or how any deviations should be addressed. Individual operating plants must present their justification for the applicability of the generic analysis to their facility in order to adopt higher burnups (for example, confirm applicability of rod burst model to their fuel, justify differences in LOCA initiating event frequencies – IEFs). Licensees should carefully consider and justify deviations from the completed generic analysis.

Determining ΔCDF and ΔLERF

Feasibility of the proposed process is demonstrated in a simple manner to estimate the effect of FFRD of HBU fuel on plant risk. Without having access to a plant-specific PRA, the demonstration should not underestimate the effect on risk. Therefore, estimated ΔCDF and ΔLERF of LOCA-induced FFRD is assumed to be the change in CDF and LERF even though it double-counts a portion of the LOCA contribution to CDF and LERF. Starting with two initiating events (that is, a large and a not so large LOCA) that are already included in the calculation of CDF and LERF, how much new risk is associated with FFRD of HBU fuel during a LOCA is estimated.

Phenomena (Top Events) Leading to FFRD in a LOCA

A PRA uses an event tree to represent the sequence of actions that must succeed (top events) to avoid core damage or containment failure. Probability of each top event failing is determined from a fault tree. The event tree is in essence a horizontal fault tree using only OR gates (that is, each branch has one of two outcomes unless probability of one branch is zero). The branches that lead to core damage or containment failure are summed to determine the CDF or LERF.

Determining the change in CDF and LERF requires identifying the phenomena contributing to FFRD during a LOCA, quantifying them, probabilistically combining them to determine the effect on CDF and LERF, and assessing total uncertainty. Figure 2-2 provides a high-level flowchart of the process. Determination of CDF and LERF involves identifying necessary functions as top events, for each of which fault trees provide estimates of failure probability. For a LOCA, examples are reactor trip, high pressure injection, primary depressurization, low pressure injection, and so on. These events could be supplemented by others pertaining to FFRD. For this evaluation, the tree consists of just those phenomena with importance to preventing or mitigating FFRD.

The first step is to identify parameters/phenomena with the potential to affect occurrence of rod burst and FFRD in HBU fuel during a LOCA. These become top events in the risk assessment. A comprehensive list provides a broader toolbox from which to choose those with the firmest technical basis to credit in the analysis, while assuming all the other contribute to failure to meet the success criterion. Those that may be less certain would be conservatively assumed to occur. Different plant designs will have varying sensitivity to parameters, requiring selection of bounding values for the generic analysis.

Prior Work on Rod Burst and FFRD

Understanding and modeling of rod burst and of FFRD have been the subject of considerable testing and analysis by the nuclear industry and the NRC.

In 2001, an expert panel applied the phenomena identification and ranking table (PIRT) process to HBU fuel subject to a LOCA [22] to improve understanding of performance of HBU fuel under LOCA conditions and if the then-current embrittlement criteria and evaluation models were adequate for HBU fuel or needed modification. This report was focused on the transition to the current 62 GWD/MTU burnup limit. The PIRT methodology can be used to identify the phenomena most important to meeting performance criteria and the state of knowledge of those phenomena in order to determine the need for additional data to better inform analyses. The panel identified about 150 items, many with considerable overlap or redundancy. The list includes parameters dependent on fuel properties (for example, gap size, cladding oxidation), operating conditions (for example, temperature, vapor fraction, water chemistry), and experimental dissimilarity from the actual plant. The report identified rod burst criteria and opening size as high significance and moderate level of knowledge (that is, partially known) and relocation and dispersal as having moderate significance and level of knowledge in regard to fuel response. Each was noted as a candidate for additional consideration but ranked below other relevant parameters.

In April 2006, substantial fuel loss was detected during a LOCA test transient performed at Halden. RIL-0801 [23] states that possibility of rapid cladding embrittlement at the assumed cladding temperature of 2600°F was identified as a reason for renewed concern but concludes "the current NRC burnup limit of 62 GWd/t (average for the peak rod) is probably low enough to prevent significant fuel loss during a LOCA."

In 2012, the NRC staff provided information related to emerging research finding that HBU fuel pellets could fragment, relocate axially and possibly disperse outside of the fuel rod during postulated design-basis accidents such as a LOCA (NUREG-2121 [25]). In March 2012, the staff did not have a sufficient technical basis for concluding whether and in what manner these phenomena should be addressed. The Commission directed the staff to further evaluate FFRD [26]. In 2015, for burnups to 62 GWd/MTU, the staff determined that "inclusion of requirements related to FFRD in the draft final § 50.46c rule is not practicable, nor is it appropriate." SECY-15-0148 [7] states that further evaluation led to the conclusion that:

> "Experimental results indicate that fine fuel fragmentation will be limited to high burnup rods and that fuel relocation will be limited to the region near the fuel rod rupture. Experimental results suggest that fine fragments from high burnup rods can easily disperse from ruptured rods during a LOCA, while larger fragments from lower burnup rods will not easily disperse from ruptured rods. The experimental results have continued to support the hypothesis that FFRD phenomena are primarily a high burnup fuel issue and that the current licensing limits in the U.S. are adequate to prevent dispersal of large quantities of fine fuel fragments."

SECY-15-0148 [7] goes on to say that the NRC staff has improved analytical capabilities for HBU fuels. Calculations showed that fuel rod ruptures, if any, would occur predominantly in the high-power, low-burnup first and second cycle fuel rods. Estimates of the dispersed fuel mass were relatively small but dependent on the assumption that HBU fuel is operated at much lower power than fresher fuel. The low burnup, high power fuel rods may suffer cladding rupture but are not vulnerable to FFRD.

For extension from 62 to 75 MWd/MTU, other and on-going empirical and analytical work were considered to identify events or parameters most relevant to affect CDF and LERF of HBU fuel during a LOCA. These events, should they occur, increase the probability and extent of rod burst or of FFRD. For ΔLERF, frequency of containment failure is also needed. In general, preparing a PRA would start with an event tree including each required safety function as a top event, and each of those top events would be quantified by a detailed fault tree. For purposes of the simple feasibility evaluation in this report, the top events are defined as undesirable outcomes (for example, rod burst occurs instead of rod burst is prevented) and probabilities are estimated for them without use of fault trees. Incorporation of fault trees could be considered if more complex modeling (for example, logical OR inputs, CCFs) for multiple contributing conditions must be assessed and if appropriate basic event frequency data is available.

Two event trees are populated: one for a large LOCA and one for a smaller LOCA. This is done to be able to take advantage of the low IEF and operator action to reduce power per the procedure prior to large breaks (for PWRs), and the lower probability and severity of fuel damage for higher frequency smaller breaks. The phenomena in Table 4-3 are described further below and quantified in Table 4-4. The table identifies currently available sources of data, which would need to be further refined during the next phase involving development of the details of the generic procedure. Note that care must be taken to avoid double counting by including two phenomena that each depend on the same physical parameter and to ensure that important functions are not omitted.

Table 4-3
Phenomena Affecting FFRD Contribution to CDF or LERF

Phenomenon/ Top Event	Description	Current Source of Data
LOCA (initiating event)	IEF of LOCA of a size to lead to rod burst and FFRD	NUREG-1829 [14,15]
Reactor at high power	Probability of being at high power at time of full break	Reactor availability statistics HRA of leak response
ECCS degraded	Probability of less than needed cooling flow (for example, multiple failures)	Reliability and Availability Data System
Rod burst and fuel fragmentation	Probability of burst of rod that is over 62 GWd/MTU any time during an operating cycle	LOCA analysis Core depletion calculations
Relocation and dispersal of large amount of fuel	Probability an unacceptable amount of fuel relocates and is dispersed from ruptured rods	Requires further development
Conditional containment failure	For a given size LOCA, probability containment will have gross leakage	NUREG-1150, Figure 9-1 [27]

- **LOCA occurs** – This is the initiating event. The break must be sizeable because the depressurization of the reactor coolant system must be fast enough to develop a substantial clad circumferential strain while high clad temperature exists. If the LOCA is small, then the conditions for rod burst may not exist.
- **Reactor at high power** – Fraction of the time spent at high core average operating power (local power peaking is considered as part of rod burst). When the reactor is at high power, fuel rod internal pressures are highest, clad strain is higher, and fuel temperature rise is higher should a LOCA occur.

  For PWRs, the alternative use of this parameter is to benefit from operator action during the time before a leak opens to a full rupture in accordance with the principle of LBB.[7] All plants have technical specifications and supporting operating procedures that detail actions for indications of unidentified leakage. Based on LBB, operators at PWR plants have sufficient time to shut down the reactor before the full break opens. However, shutting down the reactor prior to the occurrence of a LOCA also prevents the temperature and pressure transients that can lead to rod burst and FFRD. Considering LBB for purposes of excluding core degradation has not been adopted within regulations for design analysis but is technically justifiable and reasonable to credit for a risk assessment.
- **ECCS degraded** – Probability that ECCS performance is degraded. Several cases might be considered ranging from maximum capability to limited capability with probabilities dependent on the reliability of the equipment needed. An ECCS has redundant trains designed to manage the heat load from the core, including from the highest power fuel despite the design basis single failure coupled with a loss of off-site power. However, some reasonable probability exists that the ECCS is fully functional and can deliver more flow more quickly than assumed in the design basis. The better cooling may be more significant in avoiding rod burst than the lower probability that it will be available (that is, no failures).

  For smaller break sizes or lower power HBU fuel, even with its higher long-term decay heat, degraded ECCS flow may still be sufficient with multiple failures. By showing that less flow is sufficient, more ECCS degradation may be tolerable reducing the probability that sufficient flow will not be available.

  If using a specific plant PRA model, the failure logic already includes fault trees to determine reliability for the various functions needed for core cooling. However, for a standalone risk estimate like this example, failure contribution of ECCS failure to likelihood of rod burst must be included. A best estimate LOCA model could be used to determine the ECCS response

[7] Note that this is not extending LBB to justify changing deterministic criteria for ECCS evaluation, but it is acknowledging in the calculation of ΔCDF that LBB provides sufficient time for operators to detect unidentified leakage, investigate, and shut down and depressurize after a time period identified in the Technical Specifications.

(for example, how much flow how quickly) required to prevent rod burst. The required response defines the success criteria against which ECCS performance is compared. The failure model is exercised to find the probability that the success criteria will not be met.

- **Rod burst and fuel fragmentation –** This parameter addresses the two basic success criteria: likelihood of rod burst given prior parameters for high power and degraded but some ECCS flow. Predictive models for rod burst are the responsibility of the fuel vendors, which will likely work to obtain NRC agreement with their methodologies.

  1. **Burst** – For BWRs, rod burst is generally accepted as low probability in a LOCA because of lower initial pressure, lesser transient temperature rise, and clad material. Therefore, a BWR analysis may justify a low probability of rod burst.

     For PWRs, the two parameters generally accepted to make rod burst more likely are rod internal pressure which is a function of burnup, and cladding local temperature which is a function of initial power and the LOCA transient response. These parameters depend on a number of core design variables including time in cycle, radial and axial power profiles, fuel management strategy, and cladding properties.

     The generic analysis can be used to set criteria for HBU fuel operating parameters that limit burst and FFRD. For implementation at a specific PWR, detailed core power distributions can be used to confirm the loading pattern is within the generic criteria. The analysis can be done with best estimate parameters with margin added at the end to predict which rods are subject to sufficient peaking to be likely to burst. For plant loading patterns that place HBU fuel assemblies in low power locations, many rods with burnups high enough for fine fragmentation should initially be at lower temperatures, giving them more margin to burst.

  2. **FFRD** – The probability of related mechanisms is evaluated to determine the magnitude of FFRD.

     - *Individual rod burnup:* is rod at a burnup for which considerable fine fragmentation is expected to occur. This parameter is quantified by determining the burnup at different times of the operating cycles down to the individual rod or even rod elevation.

     - *Large burst opening:* testing has resulted in a range of rod burst opening sizes, even for similar conditions. If the burst opening is small, a lesser amount of fuel will be dispersed, even if finely fragmented.

     - *Relocation and dispersal from greater than six opening lengths:* post irradiation examination (PIE) has shown that fuel relocation occurs along the ballooned length of rods, defined as the portion of rod length exhibiting circumferential strain above a certain threshold. In most experiments including some with test fuel at considerably higher burnup, this manifests as fragmented fuel relocating axially within the rod from no more than one opening length below and five above the opening. In one test series [24], slow internal depressurization of remote portions of a rod demonstrated a reduced tendency for material to relocate. Spacer grids also appear to impede ballooning and thus axial relocation. Therefore, there is a low probability of greater fuel relocation. Note that there is uncertainty about radial relocation, particularly of the pellet rims which have mechanically bonded to the clad at HBU. In Halden testing, it appeared that this pellet material did not unbond until the fuel was dry and being handled for examination.

- **Containment failure probability** – The largest contributors to containment failure are usually incomplete isolation and an interfacing system LOCA (ISLOCA). Containment isolation occurs within about 30 seconds of the start of a large break LOCA, which is prior to propagation of effects of FFRD [28], so it is reasonable to conclude that any occurrence of FFRD would have no subsequent effect on containment isolation success. Likewise, the maximum possible ISLOCA is too small to cause rod burst so this failure path can also be dispositioned as unaffected by FFRD. By limiting the extent of FFRD, secondary effects such as more severe containment pressure or higher radiation dose to safety-related SSCs are minor and should not affect containment reliability. Therefore, a plant's conditional containment failure probability suitable for the LOCAs being evaluated can be used.

If these parameters are selected as top events for an event tree of a LB-LOCA leading to FFRD, just the one branch where the reactor is at high power with the ECCS degraded and conditions are conducive to rod burst will contribute to occurrence of rod burst (that is, core damage).

Quantification

Per RG 1.174, probabilities should be mean values, with uncertainty then addressed. The lack of integrated FFRD test data currently available and the considerable variation in fuel properties and conditions tested make assigning mean values a challenge. In some cases, a large uncertainty exists in the likelihood of a phenomenon or its effect on FFRD. For those, assuming the occurrence of the detrimental events, a probability of 1.0 may be conservatively assigned and the event tree simplified by omitting them. Those that have the potential to provide substantial strengthening of the low risk of FFRD are identified as gaps for future work. Therefore, for those parameters with the weakest technical justification, the generic analysis will conservatively assume they always fail and contribute to ΔCDF and ΔLERF. In essence, imposing a constraint on use of some of the items in the toolbox is a means to simplify both the assessment and the regulatory review.

The primary constraint will be to try to avoid taking credit for a frequency reduction derived from anything related with post-burst fuel behavior. If this prevents achieving a satisfactory ΔCDF and ΔLERF, the constraint can be relaxed to consider parameters allowing limited rod burst or limited FFRD.

For purposes of gauging the ability to achieve the goals for ΔCDF and ΔLERF, plausible values for top events are identified in Table 4-4. In the table, the Current Source of Data notes the primary source for the value used in the example. Although no formal justification has yet been performed, the following paragraphs give the rationale for the values selected.

Table 4-4
Plausible Values for Top Event Probability

Phenomenon/ Top Event	Probability Mean Value	Current Source of Data
Sizeable LOCA	Large break 18″ (450 mm) – $2.1x10^{-6}$/ry (BWR) 14″ (350 mm) – $4.8x10^{-7}$/ry (PWR) Small break 7″ (180 mm) – $9.4x10^{-6}$/ry (BWR) 7″ (180 mm) – $3.6x10^{-6}$/ry (PWR)	NUREG-1829 Vol. 2, Table 1 [15]
Reactor at high power	0.93 PWR LB: 0.01 if LBB	Plant availability statistics PRA human factors[8]
ECCS degraded	Large break 0.25 Small break 0.1	Very conservative value based on nominal 1% reported unavailability
Rod burst and fuel fragmentation	Large break 0.1 (BWR) 1.0 (PWR) Small break 0.05 (BWR) 0.05 (PWR)	Engineering judgment
Relocation and dispersal of large amount of fuel	1.0	NUREG-2121 [25]
Containment failure	0.01	Plant-specific PRA

[8] Plant specific PRAs will have evaluated LBB probabilities where implemented. LBB is a complicated human reliability assessment topic that involve diagnosing non-alarmed indications, monitoring by multiple operators over an extended period, and eventual plant shut down. The value 0.01/ry was taken from Table 20-6 (item 4 use written operations procedures under abnormal operating conditions) of Reference [29].

- **LOCA probability** – Because the break size below which rod burst is avoided has not been established for various plants for fuel at 75 GWd/MTU, this report assesses ΔCDF and ΔLERF for breaks as small as 7-inch (17.8 cm) diameter. NUREG-1829 ([14] and [15]) is the result of an expert elicitation that conservatively bounds opinions on the probabilities of LOCAs of different sizes. LOCA mean probability values at 40-year plant age are taken from Table 1 of Volume 2 [15]. The break size was arbitrarily divided into two ranges to avoid applying the higher probability of small ruptures to the DEGB. In the future, fracture mechanics analysis using the xLPR software code may be able to inform a lower probability of failure, including breaks caused by seismic events. It may also be useful to deal with aging mechanisms or other issues affecting pipe failure probability if raised by the NRC. Although the size rupture below which rod burst is precluded is not yet established for small break LOCA, the break size selected was considered a reasonable lower bound because panel members expected that rod burst can be excluded for HBU fuel for break sizes that have at least two times the flow area.
- **Reactor at high power** – Plants spend most operating time at close to full power. Postulated LOCA events during maintenance outages or low power operation have insignificant impact on cladding temperatures. Therefore, the utility average plant availability factor was assumed. Alternatively, for PWRs implementing LBB, credit is taken for expectation that precursor leakage is recognized, and the plant shut down prior to full break opening at least 99% of the time (LBB currently does not apply to BWRs).
- **ECCS degraded –** The ECCS is assumed to provide flow sufficient to prevent HBU rod burst for 0.75 (0.90 for small breaks) of the demands, and degraded flow because of component failure for the remaining 0.25. A full PRA model would have more complex logic and allow for various combinations of failed and working components.

Rod burst and fragmentation – BWRs have a low probability of rod burst for LOCA transients because of several factors (for example, clad material, rate of change of pressure differential across cladding, slower cladding temperature rise); values for large and small breaks are selected for this example but would be determined based on thermal-hydraulic analysis in future work. Hotter rods are more likely to burst. HBU assemblies are usually in average or lower power regions of the core in order to satisfy other fuel limits, with fresher assemblies operating at higher powers that control core average conditions. In addition, during part of the cycle, assemblies may have margin to burst and FFRD based on core power and exposure distributions. A number of studies using best estimate plus uncertainty methods have shown few HBU rods burst. Even if a HBU assembly is in a higher power location, not all of its rods will burst. Some uncertainty remains as to the conditions controlling FFRD under actual LOCA transients. There is likely some burnup and cladding temperature threshold below which FFRD is not significant, but there is considerable variation in data on specific conditions. On-going work and, in particular, planned testing at TREAT may inform this probability, but a value of 1.0 will be assumed for now for LB-LOCA, simplifying the assessment and avoiding dependence on FFRD modeling.

For smaller LOCAs, decay heat comes down while the fuel is still being cooled, lessening peak temperature, and system pressures remain high during periods of cladding temperature increases. Probabilities of rod burst for small LOCAs are assumed to be lower than for large breaks because of slower depressurization and fuel rod heat up.

- **Relocation and dispersal of a large amount of fuel** – For simplicity, a value of 1.0 will be assumed, avoiding dependence on FFRD modeling.
- **Containment failure** – If no rod burst occurs, the conditional containment failure probabilities associated with a given size LOCA should be unchanged. If the constraint on rod burst and FFRD are removed, some assessment of their effect on containment reliability is needed. However, as the containment is the final barrier in a severe accident, the limit on the amount of dispersed fuel would be established to be within its design basis and not detrimental to long-term safety-related equipment function. For many plants a failure probability of 0.01 is applicable, and will be used in the proof of concept.

Applying these values to the event tree yields the results shown in Table 4-5 for CDF and Table 4-6 for LERF. Because small and large break LOCAs were analyzed separately, their values must be added together. For both BWRs and PWRs considering PWR operator action based on LBB, the results satisfy the CDF and LERF goals. Recall that these values are inherently conservative in that they double count LOCA initiating events already in the PRA, but they are mean values for which uncertainty must be added. These estimates provide some assurance of ability to achieve satisfactory ΔCDF and ΔLERF values, although these values will change as inputs become better known. Note that the small break results dominate. Application of xLPR to lower the failure probability or support LBB evaluations to confirm that at-power larger breaks will be prevented may be beneficial.

Table 4-5
ΔCDF Results (Goal = $1.0x10^{7}$/ry)

LOCA Type	BWR Mean Value (/ry)	PWR Mean Value (/ry)	PWR Mean Value with LBB applied (/ry)
Large break	$4.9x10^{-8}$	$1.1x10^{-7}$	$1.2x10^{-9}$
Small break	$4.4x10^{-8}$	$1.7x10^{-8}$	$1.7x10^{-8}$
Total	$9.3x10^{-8}$	$1.3x10^{-7}$	$1.8x10^{-8}$

Table 4-6
ΔLERF Results (Goal = $1.0x10^{8}$/ry)

LOCA Type	BWR Mean Value (/ry)	PWR Mean Value (/ry)	PWR Mean Value with LBB applied (/ry)
Large break	$4.9x10^{-10}$	$1.1x10^{-9}$	$1.2x10^{-11}$
Small break	$4.4x10^{-10}$	$1.7x10^{-10}$	$1.7x10^{-10}$
Total	$9.3x10^{-9}$	$1.3x10^{-9}$	$1.8x10^{-10}$

Uncertainty

Based on the acceptance guidelines in RG 1.174, the PRA results use mean values. A formal propagation of uncertainty may not be necessary if the state of knowledge correlation (SOKC) is unimportant to the regulatory decision under consideration. This could be done by showing most of the dominant scenarios do not involve multiple events that rely on the same parameter for their quantification. Section 6 of NUREG-1855 [30] provides guidance on SOKC.

Prediction of progression of events involving FFRD involves use of models. While LOCA models are well pedigreed, models of fuel degradation involving FFRD during a LOCA are less mature. RG 1.174 acknowledges that, in many cases, the industry's state of knowledge is incomplete, which gives rise to model uncertainty. Where the appropriateness of models is accepted, they have become consensus models, which NUREG-1855 [30] defines as having a publicly available published basis and having been peer reviewed and widely adopted by an appropriate stakeholder group. RG 1.174 points out widely accepted PRA practices may be regarded as consensus models. For risk-informed regulatory decisions, the NRC has used or accepted the consensus model approach. For some issues with well-formulated alternative models, PRAs have addressed model uncertainty by using discrete distributions over the alternative models, with the probability associated with a specific model representing the analyst's degree of belief that the model is the most appropriate.

To implement the no burst constraint, every post-burst top event probability is set to 1.0, which means taking no credit for it to reduce CDF or LERF. If this results in an unacceptable ΔCDF or ΔLERF, then the constraint may by relaxed by estimating the risk (that is, probability and consequences) of FFRD. Relying on a less well justified phenomenon requires understanding the impact of a specific assumption or choice of model on the predictions of the PRA because the probabilities, or weights given to different models are subjective. The impact of using alternative assumptions or models may be addressed by performing appropriate sensitivity studies or by using qualitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models. In other words, taking credit for post-burst behavior involves additional evaluation to account for increased uncertainty.

If supported by the industry, follow up activities to implement the proposed strategy will be required. During this subsequent phase, alternative assumptions and models that would drive the ΔLERF toward unacceptability should be identified and sensitivity studies should be performed. It may be possible to define the range of FFRD behavior sufficiently well to perform sensitivity studies (for example, the extent of HBU rod burst is limited to only fuel rods with above core average power) to show that various outcomes are still acceptable. In general, the results of the sensitivity studies should confirm that the guidelines are still met even under the alternative assumptions (that is, change generally remains in the appropriate region).

In consideration of the need to move forward at the current state of knowledge but allow for folding in improved modeling as it becomes available, the approach outlined above imposes the constraint to use only basic, well understood, non-FFRD phenomena to determine ΔCDF and ΔLERF. As these are factors that are already quantified in existing PRAs (for example, LOCA IEF, conditional containment failure probability), uncertainty values should already be available. For the FFRD factors, placeholders are used with conservative values of 1.0. This will permit ready inclusion of improved understanding of FFRD as it develops.

Prediction of Fuel Response to LOCA Conditions

RG 1.174 discusses the need for ensuring that severity of consequences is understood, lest low probability and uncertainty in consequences lead to overlooking a significant phenomenon. There are uncertainties in prediction of burst opening size and more so of FFRD, but there is more confidence in modeling burst occurrence.

The two primary factors in determining if and when a rod bursts in a LOCA are generally accepted to be rod internal pressure and cladding local temperature [31], which are functions of burnup (higher burnup causes higher rod internal pressure) and local power (higher local power causes higher fuel and cladding temperatures following the accident). Cladding temperature is also affected by LOCA break size because of the complicated interaction among rate of blowdown, rate of power drop, loss of heat transfer, and so on. Nevertheless, to reach high enough cladding temperatures to burst, it is likely that the fuel location must be no longer covered by liquid water.

Although HBU fuel assemblies are subject to these same burst-related factors, they are usually placed in lower power locations in order to stay within other fuel limits. At these locations, the LOCA temperature transient is less severe, making rod burst less likely. Unless FFRD is precluded by excluding burst, assuring no safety-related equipment failure from fuel dispersal requires some quantification of the extent of FFRD. However, this is an area that is not well characterized.

A number of experiments have been performed to determine conditions leading to rod burst and fine fragmentation. They have been described in detail in several references ([25], [32]). Varying interpretations of their results have led to a range of predictions for initiation and subsequent severity of FFRD. The experimental data are not necessarily representative of the LOCA event sequence and fuel rod conditions causing FFRD, but a number of estimates have nevertheless been made based on those tests of the amount of fuel that might be dispersed.

Staff from NRC's Office of Nuclear Regulatory Research presented a conference paper [33] of nominal (that is, no added conservatisms, all trains of ECCS operable) predictions of rod ruptures and possible resulting fuel dispersal. Core-wide fuel rod rupture census calculations and the associated fuel dispersal predictions were obtained with the FRAPCON/FRAPTRAN and TRACE codes. Two PWRs and one BWR, five LOCA scenarios, and three distinct time steps during an operating cycle were evaluated. Core average discharge burnups were given for the PWRs as 54.5 (Westinghouse 4-loop) and 57.8 (Combustion Engineering) GWd/MTU, but peak burnups are not given. With the exception of the Westinghouse 4-loop analyses, the paper states "It is believed that these temperatures are too low to result in fuel rod ruptures, meaning no fuel dispersal is possible for this particular transient in this particular reactor." For the Westinghouse 4-loop analysis that did predict ruptures, key observations include:

- Most rod bursts occur at end of cycle (EOC) because of higher internal rod pressures.
- 55% of fuel rods in the core ruptured, all of it new or once burner with the only twice burned fuel assembly with ruptured rods located at the center of the core.
- Ruptures usually occur at the peak power locations.
- Bursts began at about 30 seconds, initially peaked at 75 seconds, tailed off by 100 seconds, and briefly resumed between 210 and 260 seconds. The twice burned fuel assembly rod ruptures occurred from about 65 to 100 seconds.
- Dispersed fuel mass was evaluated for three different thresholds (55, 60, and 65 GWd/MTU) for onset of fine fragmentation (below 1 mm). Assuming any rupture location with a cladding hoop strain above 5% contributed, predicted dispersed fuel mass ranged from 28 to 53 kg at beginning and middle of cycle. At EOC, dispersed mass was highly sensitive to the assumed threshold, ranging from 105 to 622 kg. These results indicate that, on a nominal basis, the amount of dispersed fuel is small except at EOC, which limits the probability of occurrence by requiring a limiting LOCA to occur during a short timeframe.

NUREG-2121 [25] discusses several evaluations of failed fuel. Nissley performed a deterministic calculation of a four-loop Westinghouse plant in which an entire train of emergency core cooling was lost and 15% higher than maximum power peaking was applied. Using an empirical rupture temperature versus hoop stress curve, less than 10% of rods breached. For comparison, a best estimate plus uncertainty analysis was performed, yielding a peak cladding temperature almost 200°C (392°F) lower.

NUREG-2121 [25] evaluated axial fuel relocations and particle migration through a rupture opening. It concluded that some fragmentation appears to almost always occur during a LOCA, regardless of burnup and that axial relocation occurs if there is appreciable cladding diametric strain. Rod ballooning is partly inhibited by grid spacers, which could act as choke points for axial relocation. The amount of fuel dispersal involves the amount of fragmented fuel that relocates, fragment size which decreases with burnup, and rupture opening size; the latter two of which determine the likelihood of the rupture opening being blocked by fragments. No direct correlation between burnup and rupture opening size was noted, but higher rod internal pressure did cause wider openings.

NRC [34] noted results of an unofficial fuel dispersal estimate without identifying burnup or fuel or plant type: "During a postulated LOCA, a large number of fuel rods would likely experience cladding failure – Between 3,000 - 5,000 lbs. of UO_2 would be susceptible to fragmentation and dispersal assuming 20% of the core experienced failure with an average 9 inch balloon region." (9 inches is ~ 23 cm). This estimate is considerably higher than Raynaud's results [33], and no basis is given for the stated assumptions or how the estimate was determined.

FFRD Effects on LERF in PWRs

Although the objective is to calculate ΔCDF on a no burst basis, defense-in-depth evaluation must address the potential effects of FFRD on phenomena that are important to LERF. The following list of potential effects indicates their expected disposition subject to closure of required gaps.

- **Failure of containment isolation** – Containment isolation occurs within about 30 seconds of an isolation signal, which should initiate within a few seconds of large break LOCA blowdown starting. FFRD occurs relatively quickly (but no earlier than about 30 seconds) for large breaks – and much later (if at all) as break size decreases – but this is not nearly enough to affect isolation. Conclusion: no effect.
- **Increase of containment pressurization/leakage rate** – The leak rate depends on containment pressure, which is calculated on a conservative basis (for example, rapid transfer of stored energy in fuel to coolant, high mass flow from break), and on parameters not affected by FFRD (for example, stored energy of the coolant, component individual leak rates). Because of the assumed rapid heat transfer, rapid heating of coolant resulting from dispersal of fine particles from a burst rod should be bounded. For burst and subsequent FFRD to occur, the fuel rod likely must be uncovered (channel voided) at the location of the burst. Compared to assuming rapid heat transfer on a core wide basis, depositing fine fuel fragments into a steam atmosphere results in small additional energy deposition rate because of the small amount of heat contained in the dispersed fuel fragments. Small heat input to the steam will have little effect on pressure. For beyond burst, the potential for double sided oxidation needs to be considered. Conclusion: no adverse effect.
- **Increase of early source term inside containment** – Fragmentation may free more trapped fission product gases early in an accident, potentially increasing the quantity during the Gap Release Phase. Fine particles, though, would not be carried to containment leakage points and could not pass through the tight leak paths. Note that most gaseous and halogen fission products (excepting krypton-85) have short half-lives, so their inventory available for release from HBU fuel is not greater. Conclusion: may increase gaseous release.
- **Likelihood of failure of neighboring rods from increases in heat flux if fuel relocates into a shorter, ballooned rod segment** – Channel blockage must already be accounted for in design basis methods for purpose of cladding temperature predictions. Conclusion: need not be addressed as part of LOCA-induced FFRD.

- **Effect of redistribution of fuel particles on radiation exposure to safety-related equipment EQ** – If the no rod burst success criterion is applied, no fuel is released. For occurrence of FFRD to affect EQ dose, the fuel particles must relocate outside the reactor pressure vessel and primary coolant system. Models are still being developed to quantify transport. However, the equipment EQ profile is based on conservative dose estimates for generally at least 30 days. The limit on the FFRD contribution, if any, would be set to be within the EQ assumptions. Conclusion: no decrement.
- **Effect of redistribution of fuel particles on long-term core cooling (for example, flow blockage)** – GSI-191 evaluations demonstrate that a substantial amount of material with a tendency to clump must collect in specific locations to cause a long-term core cooling problem. The amount of fuel material dispersed is small and unlikely to clump (otherwise it would not disperse). Conclusion: no decrement.
- **Accumulation of a potentially critical mass outside the core** – The dispersed fuel particles would be from HBU rods that have depleted U-235 which includes neutron absorbing fission products. In addition, a large mass of U-235 would have to implausibly arrange in an optimum geometry. Conclusion: not credible.

Quality Assurance

PRA evaluations and, hence, this approach are not required to be performed under 10 CFR 50 Appendix B. However, plant-specific evaluations must be performed under a quality control process including analysis by qualified personnel, use of procedures, version control, record keeping, and so on.

Benefits

The benefits from extension of fuel burnup to 75 GWd/MTU are economic improvements with respect to fuel utilization, high level waste reduction, and risk reduction. Significant focus in industry has been on the economic benefits (for example, see [35]), but for purposes of RG 1.174 the risk reduction is of more relevance. By virtue of being able to use a large number of fuel assemblies for a third cycle or even fourth cycle, the following nuclear risk reductions accrue:

- Fewer new fuel assemblies will be required to produce the same energy. This reduces the mining, milling, enrichment, fuel fabrication and the associated transportation of nuclear fuel, dose consequences from these activities and the risk of industrial accidents associated with these activities.
- Fewer spent fuel assemblies will be generated, reducing the amount of spent fuel interim storage, spent fuel shipments, and eventual disposal. This lowers the risks of accidents associated with these activities and the associated radiological consequences of these activities.

- Worker radiation exposure will be reduced because of the smaller number of maintenance outages (with longer cycle lengths), fewer spent fuel assemblies in storage, and fewer number of spent fuel assemblies that must be loaded into dry storage casks. Additionally, the risk from fuel handling events is reduced and with fewer plant startups the risks from these maneuvers is also positively impacted.

Gaps

A major objective of this evaluation is to identify gaps needing action to inform the technical bases for a LOCA to cause FFRD of fuel at burnups of up to 75 GWd/MTU.

Table 4-7 summarizes these gaps. The table indicates if closure of a gap is *required* for this regulatory approach to succeed or is *desirable* to strengthen/ simplify the technical bases. Also, the reason for these gaps may be to support of any of the following areas:

- For generic proof of concept calculation of ΔCDF (indicated in table by “CDF” in Area column) and/or ΔLERF (LERF). These can be closed with generic estimates with moderate uncertainty.
- Provide guidance for plant-specific implementation (Plant). These can be closed by providing criteria to set plant-specific estimates with strong technical basis.
- Assist in demonstrating that the seven layers of defense-in-depth (DiD) and that safety margins are maintained. These may be closed on a qualitative or approximate quantitative basis with a sound technical basis.
- Expected to be necessary to lower top event probability to achieve probability goals (Goal). Gap closure requires a strong, defensible basis for a value with a defined uncertainty.

A single gap may have multiple purposes with different needs. For example, improved modeling of the extent of FFRD required for defense-in-depth can be sufficient on an approximate numerical (for example, bounding) basis for defense-in-depth but also be desirable on a sound quantitative basis as part of the determination of plant-specific ΔLERF.

Table 4-7
Gaps to Address

Gap	Need	Area	Explanation
Conditional containment failure probability	Required	LERF DiD Plant	Determine best estimate value for containment failure for a given size LOCA under conditions of no burst and limited FFRD.
Smallest break size for which rod burst occurs	Required	CDF LERF Plant	Establish LOCA sizes for which IEF must be included as an input to the ΔCDF and ΔLERF estimates.
Effect of FFRD on coolable geometry or ability to prevent progression leading to severe accident	Required	DiD Plant	Demonstrate acceptability of limited FFRD consequences on prevention of severe accident
Safety margins	Required	Safety margins	Demonstrate that the proposed changes do not impact safety margins
LOCA IEF out to 80 years	Required	Goal CDF LERF Plant	NUREG-1829 IEF estimates are about 20 years old. During deliberations on 50.46a, the NRC expressed concern regarding effect of aging on the values from NUREG-1829, which shows 40-year IEFs as about 30% and 130% higher than the 25-year values for BWRs and PWRs, respectively. Also, the NRC will expect that beyond design basis earthquake risk (NUREG-1903) be included. For plants that do not screen out of seismic PRA, updated IEFs to address both of these aspects may be obtained from xLPR. Alternatively, plant IEF PRA values, if suitable, or the methodology described in Reference [36] could be used.
Application of LBB (PWR)	Desirable	Goal CDF LERF DiD Plant	Credit for detecting leakage prior to break opening (that is, LBB response in accordance with Technical Specifications) is a key risk mitigation for PWRs. Probability of operators recognizing a leak and successfully completing LBB actions to shut down prior to full break opening needs to be justified if not already done so in the plant PRA.
Impact of axial fuel relocation on heat flux	Desirable	DiD	Determine the impact of fuel relocation on in-core heat flux. NRC concern that clad ballooning followed by axial relocation will cause a widespread failure needs to be evaluated by modeling fuel relocation without excess conservatism.

Table 4-7 (continued)
Gaps to Address

Gap	Need	Area	Explanation
Vulnerability to human errors	Desirable	DiD	The methodology should be evaluated for vulnerability to human errors that could increase probability of failures related to FFRD.
Acceptable model of fuel relocation and release from a fuel rod and subsequent dispersal outside RCS	Desirable*	DiD Plant	Accurate modeling of dispersal is needed to verify that the potential for adverse effects on safety-related systems is within the EQ envelope and meets the accident source term acceptance criteria.
TREAT transient test data	Desirable	DiD, LERF	The planned TREAT testing is intended to be more prototypical than previous testing to address uncertainty as to the conditions for FFRD and when during a transient different FFRD phenomena occur. While assessment can be done without this data, having it would reduce uncertainty and improve ability to justify closure of the fuel relocation gap noted above.

* This gap could be requalified as "required" as a backup to no rod burst.

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Appendix A: Regulatory Alternatives for High Burnup Fuel – Questionnaire for SMEs

Questionnaire

This section provides a list of questions to gather information to perform the assessment. At this stage, answers need not be detailed, and emphasis should be placed on the succinct description of concepts and useful references. If you consider a question outside the area where you feel you have input to be considered, then just skip it.

For purposes of this assessment, cladding is current zirconium-based alloys with or without chromium coating, or iron-based alloy, holding uranium dioxide fuel pellets with possible initial enrichments slightly exceeding 5% and limited possible dopants.

Fuel Performance

1. Compare susceptibility to clad failure and subsequent FFRD above 68 GWd/MTU for the near-term ATF concepts[1] and the fuel currently used[9].
2. Do you see a problem with justifying adequacy of performance of HBU near-term ATF concepts for any other conditions (beside LOCA), such as a reactivity induced transient?
3. What gaps in fuel/clad data will exist after currently planned testing (for example, in TREAT)?
4. Will these gaps prevent regulator acceptance of fuel performance modeling?
5. In going from current fuel[2] depleted to 68 GWd/MTU to near-term ATF concepts depleted to 75 GWd/MTU, is the change in FFRD likelihood and severity an actual concern?
6. Could limitations on HBU fuel core locations be used to offset concerns with going to 75 GWd/MTU?

[9] That is, zirconium-based cladding with UO_2-only pellets.

7. Is the concern for FFRD a result of the different clad, the higher initial enrichment, the higher burnup, the longer time in-pile, a combination, and/or other? Parameters?
8. Do variations in different suppliers' near-term ATF concepts (details are proprietary) require each to be addressed separately?
9. Does increasing enrichment from current limits of <5% to possibly about 7% have any effect on FFRD performance?

Licensing Alternatives Evaluation

Reply to the following questions for all alternatives presented above (Section **Error! Reference source not found.**) and for any others you would like to propose to the group for discussion.

1. Rank alternatives by which you think has the highest to lowest (or no) potential for success, where success is obtaining NRC agreement with modest industry effort by 2026.
2. What data is most critical to success?
3. What actions must be performed by fuel vendors and plant licensees to enable success?
4. Does the alternative require changes to NRC regulations (that is rule-making)? List the impacted regulations.
5. Does the alternative require change to NRC regulatory guidance? List the most significant Regulatory Guide and SRP items and challenges to developing modified guidance.
6. What do you see as the single greatest impediment or objection to NRC acceptance?
7. What peripheral or otherwise indirectly related issues must industry be ready for the NRC to potentially raise?
8. Is FFRD a LOCA only issue or must reactivity insertion transients also be addressed?
9. What concerns do you have with implementing an alternative licensing approach?

LOCA

1. Given existing LOCA evaluation guidance in the US and related history (for example, Appendix K, RG 1.157, transition break size), is it conceivable to make the DEGB beyond design basis, or is that a non-starter (for current fuel and near-term ATF concepts)?
2. What justification might be used to move the DEGB from design basis to beyond design basis (for current fuel and near-term ATF concepts)? For example, could LBB be justified for core cooling and containment design in addition to its current applicability for pipe whip and jet impingement? Would this require any adaptation of the xLPR computational tool (NUREG-2110, https://www.nrc.gov/docs/ML1214/ML12145A470.pdf)?

3. Where do you see the potential to remove conservatism from the NRC LOCA evaluation methodology that might be useful to address the occurrence of FFRD phenomena (for current fuel and near-term ATF concepts)?
4. NUREG/CR-6744 (https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6744/) is a LOCA PIRT for HBU fuel performed in 2001. Is it applicable for the near-term concepts ATF concepts? If so, is it a useful input for our evaluation, and how should it be used?
5. What fuel/clad response to LOCA conditions would be significantly different for near-term ATF concepts at HBU? Minor differences (for example, slightly slower tail off of decay heat) do not need to be considered for now.

Risk-Informed Approach

1. Is there some suitable blend or alternative to the current three methods of LOCA analysis identified by the NRC (1 of the first 2 plus the third are needed):
   a. Design basis using Appendix K.
   b. Best estimate with uncertainty using RG 1.157.
   c. PRA determination of core damage frequency which considers the low likelihood of DEGBs and uses best estimate core cooling methodology.

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