ENCLOSURE 2:

Near-term Approach for High Burnup FFRD
(NON-PROPRIETARY)

Contents

List of Figures

Acronyms

Acronym	Definition
AFM	Advanced Fuel Management
BWR	Boiling Water Reactor
CE	Combustion Engineering
DBA	Design Basis Accident
DEG	Double Ended Guillotine
DEGB	Double Ended Guillotine Break
ECCS	Emergency Core Cooling System
FFRD	Fuel Fragmentation, Relocation, and Dispersal
FSAR	Final Safety Analysis Report
GDC	General Design Criterion
HBU	High Burnup
LBE	Licensing Basis Event
LBLOCA	Large Break Loss of Coolant Accident
LOCA	Loss of Coolant Accident
PWR	Pressurized Water Reactor
SSC	Structures, Systems, and Components
TER	Technical Evaluation Report
W	Westinghouse

1.0 INTRODUCTION

Framatome plans to submit a high burnup (HBU) topical report (TR) to extend approval of our suite of analytical models and methods up to a maximum rod average burnup of [ ] GWd/MTU. At these elevated burnups, fuel rods become more susceptible to fuel fragmentation, relocation, and dispersal (FFRD). Postulated dispersal of significant quantities of finely fragmented fuel particles introduces regulatory uncertainty, analytical complexities, and may challenge how licensees have historically demonstrated compliance to many regulatory requirements. The Commission has not provided direction or guidance to address regulatory uncertainty with respect to demonstrating compliance in the presence of dispersed fuel particles. Framatome's near-term approach provides a means for licensees to demonstrate continued safe operation with no undue risk to public health and safety while migrating toward advanced fuel management (AFM) reload cores (e.g., extended reload cycles, increased ^{235}U enrichment) during the interim period while research continues to fill data gaps and the Commission considers potential regulatory infrastructure changes.

The purpose of this white paper is to request written feedback on policy issues associated with our proposed near-term approach for addressing HBU FFRD. This will help reduce regulatory uncertainty, assist Framatome with completion of the HBU topical report, and instill stability and predictability in its review. Specifically, Framatome is requesting written feedback on the following conceptual items of its plan:

1. It is acceptable to employ risk and safety significance to define the appropriate level of reasonable assurance of adequate protection of public health and safety,
2. A level of reasonable assurance commensurate with risk, but lower than the traditional high probability level, is justifiable for extremely low frequency of occurrence large piping breaks, and
3. It is acceptable to employ a supplemental analysis using the level of reasonable assurance defined above to demonstrate that LOCA-related Final Safety Analysis Report (FSAR) safety analyses remain applicable, and plants remain in compliance with existing regulations.

To provide context to this request, Section 2 presents an overview of the near-term approach. Application of the near-term approach is limited to AFM reload cores in Combustion Engineering and Westinghouse nuclear plants with core reload batch quantities of Framatome fuel assemblies. Section 3 provides an overview of the risk-insights and safety significance aspects of the near-term approach and their use justifying the level of reasonable assurance of adequate protection.

2.0 OVERVIEW OF NEAR-TERM APPROACH

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It is important to recognize that there is no single, all-inclusive LOCA analytical simulation which feeds important dynamic parameters into a comprehensive performance demonstration of the overall plant systems' response to a postulated LOCA. While differences in system designs and performance requirements exist between and amongst the Combustion Engineering (CE) and Westinghouse (W) fleet, several common safety-related SSCs are designed to withstand the harsh environmental conditions and perform crucial safety functions during and after a postulated LOCA. Each plant's FSAR documents the performance demonstration for each of these safety-related SSCs which are relied upon to mitigate the consequences and maintain risk to public health and safety to acceptable levels. Several of these performance demonstrations are listed below[1]:

a. Environmental qualification of electrical equipment (FSAR Chapter 3.11)
b. Fuel assembly mechanical design (FSAR Chapter 4.2)
c. Containment systems design (FSAR Chapter 6.2)
d. ECCS system performance (FSAR Chapter 6.3 or 15.6)
e. Radiological consequences (FSAR Chapter 15.6)

Each safety-related SSC performance demonstration identified above is conducted with different analytical methods, assumptions, single failures, initial conditions, uncertainty treatment, performance metrics, acceptance criteria, compliance metrics, etc. As such, HBU FFRD may impact each of the above FSAR performance demonstrations differently. [

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Framatome's approach to employ a supplemental analysis to essentially define a boundary condition to another FSAR performance demonstration is not beyond the existing regulatory construct and current practice. For example, FSAR Chapter 4.2 fuel assembly mechanical design analyses predict the performance of fuel assembly grid cages under applied external loads associated with combined safe shutdown earthquake and LOCA-related core plate motions. These analyses are performed in accordance with the requirements of General Design Criterion 2, *Design bases for protection against natural phenomena*, and 10 CFR 50 Appendix S, *Earthquake Engineering Criteria for Nuclear Power Plants.* If grid cage plastic deformation is

[1] The content and formatting of FSARs vary among the W and CE fleet. For older plants, the accident analyses are documented in Chapter 14 (versus Chapter 15 for newer plants). In addition, ECCS performance demonstrations sometimes reside in Chapter 6 and other times in Chapter 15.

predicted, then this information (i.e., boundary condition) is supplied to a ECCS performance demonstration to ensure that the requirements of 10 CFR 50.46 are satisfied. As stated above, LOCA-related performance demonstrations are not always conducted in a consistent manner. In this example, LOCA-related core plate motions may credit leak-before-break to remove the extremely low probability of rupture in large diameter piping from consideration. Hence, while the ECCS performance demonstration considers all piping breaks up to DEG of the largest diameter piping, the fuel mechanical design analyses which provide the predicted grid deformation (i.e., boundary condition) are limited to core plate motions associated with smaller piping breaks. Framatome's approach to employ a supplemental analysis, informed by risk-insights and safety significance, to essentially define a boundary condition to the existing FSAR LOCA-related compliance demonstrations, is therefore justifiable and not a significant departure from existing regulatory practice.

10 CFR 50 Appendix A, *General Design Criteria for Nuclear Power Plants*, states that principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. [

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3.0 JUSTIFICATION FOR LEVEL OF REASONABLE ASSURANCE

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] Risk-insights are defined in terms of initiating event frequency of occurrence. Safety significance is defined in terms of change in defense-in-depth and change in risk of public exposure to radiation.

As indicated by several past NRC-sponsored projects, large piping breaks are extremely low frequency of occurrence events. One reason is because of strict requirements to satisfy General Design Criteria 14, *Reactor coolant pressure boundary*, shown below.

> *Criterion 14 - Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.*

The results of several studies including NUREG-1829, *Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process* (Reference 1), have provided estimated frequencies as a function of break diameter for both PWRs and BWRs. The NUREG-1829 estimates were based on an expert elicitation process which consolidated operating experience and insights from probabilistic fracture mechanics studies with knowledge of plant design, operation, and material performance. Figure 3-1 illustrates these estimated frequencies.

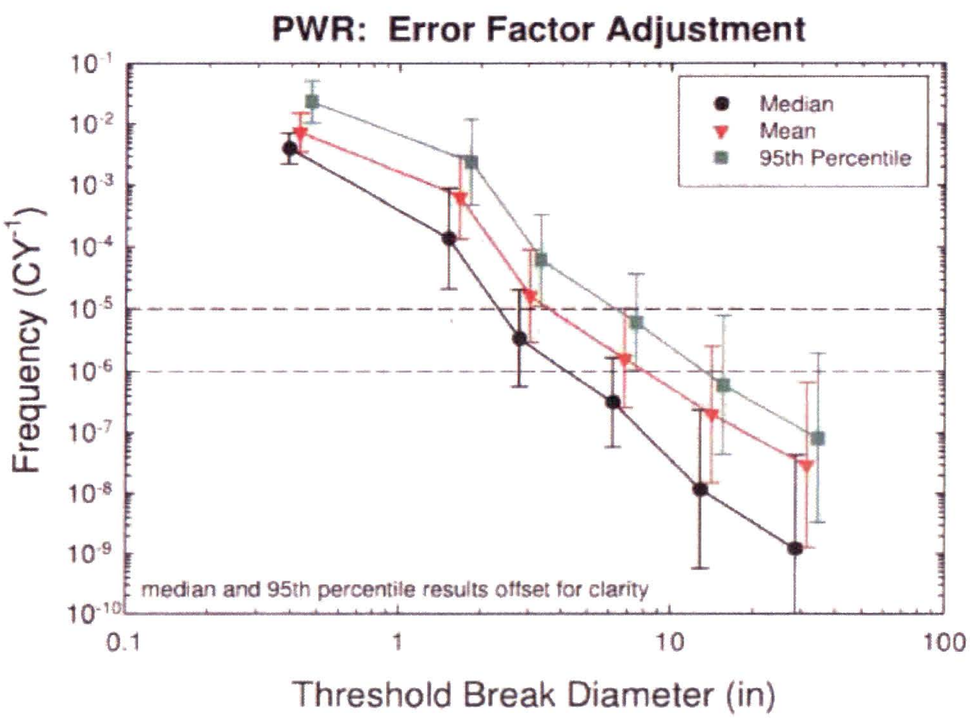
The referenced studies of LOCA estimated frequencies demonstrate that the order of magnitude for large piping breaks are 10^{-6} per year for PWRs. These estimated frequencies are independent of fuel design, ^{235}U enrichment, and burnup. A large piping break is an extremely low probability of occurrence event and no more likely in a future HBU reload core as in currently operating reload cores.

The safety significance is addressed by assessing the change in defense-in-depth and change in risk of public exposure to radiation. The near-term approach does not introduce any proposed changes to existing regulatory requirements with respect to single failure criteria, loss-of-offsite AC power, full piping break spectrum (up to DEG), or conservatisms within existing radiological consequence analyses. Therefore, the existing levels of defense-in-depth are maintained.

For the change in risk of public exposure to radiation, several recent studies will be cited which conclude the HBU fuel fragmentation and dispersal would not increase the radiological source term released into containment or the public exposure to radiation beyond the bounding analysis docketed in each plant's FSAR. Therefore, the risk of public exposure to radiation is not sensitive to the level of reasonable assurance used to preclude fuel dispersal.

Figure 3-1
Error-Factor Adjusted LOCA Frequency Estimates

(Source: NUREG-1829)



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In developing the conceptual approach for introducing risk-insights, Framatome considered several alternatives including existing regulatory guidance such as RG 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.* At this time, during the period while research continues to fill data gaps and the Commission considers potential regulatory infrastructure change, using initiating event frequency as a metric for risk-insights, in lieu of plant-specific quantification of risk attributes (i.e., change in core damage frequency), is appropriate for this application and is consistent with the NRC staff's disposition of in-vessel downstream effects (IVDE) of debris in Generic Safety Issue Number 191 (GSI-191). As discussed in Section 5.5 of Reference 11 (*Integrated Decision Making*), the staff based risk-insights on initiating event frequency with no plant-specific quantification of risk attributed to IVDE. Framatome's near-term approach provides a cost-effective, readily deployable means for licensees to demonstrate reasonable assurance of adequate protection and continued compliance to existing regulatory requirements.

Framatome's near-term approach includes a risk-informed supplemental analysis to provide reasonable assurance that LOCA-related aspects of a plant's UFSAR will not be impacted by FFRD. Inherent in this application is that the composite risk of the event, the phenomena, and its relevance to the safety of the plant and public is extremely low. The Atomic Energy Act of 1954, as amended, establishes "adequate protection" as the standard of safety on which NRC regulation is based. In the context of NRC regulation, safety means avoiding undue risk or stated another way, providing reasonable assurance of adequate protection for the public in connection with the use of source, byproduct, and special nuclear materials. SECY-18-0060, *Achieving Modern Risk-Informed Regulation* (Reference 2), requested Commission approval of several transformation initiatives, including actions to enhance and sustain a culture that embraces transformation at the NRC. One of the overarching themes discussed in the paper is the need for systematic and expanded use of risk and safety insights in decision making, including the need to appropriately scale the scope of staff review and level of detail needed from an applicant for licensing decisions, consistent with NRC regulations and the overall standard of reasonable assurance of adequate protection.

As shown in the examples provided in Section 3.1, the concept of risk-informed regulatory decisions is not new and continues to take on larger roles in recent years.

3.1 *Recent Risk-Informed Approaches*

In 2022, the Commission directed their staff to (1) address FFRD issues relevant to fuels of higher enrichment and burnup levels, and (2) take a risk-informed approach when developing the Increased Enrichment rule and the associated regulatory basis and guidance (Reference 3).

In 2020, the NRC published risk-informed guidance to address the selection of licensing-basis events (LBEs); classification and special treatments of structures, systems, and components; and assessment of defense in depth for advanced, non-LWR designs. The regulatory philosophy of frequency – consequence (F-C) was established in this guidance, RG 1.233, *Guidance for a Technology-Inclusive,* *Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors* (Reference 4). This guidance endorses Nuclear Energy Institute (NEI) 18-04, Revision 1, *Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development* (Reference 5). The selection, classification, and allowable consequences of LBEs are based on the F-C Target shown in Figure 3-2.

In 2019, the Commission directed the staff to review Chapter 15 of the NuScale Design Certification Application without assuming a single active failure of the inadvertent actuation block valve to close (Reference 6). In conclusion, the Commission wrote:

> *In any licensing review or other regulatory decision, the staff should apply risk-informed principles when strict, prescriptive application of deterministic criteria such as the single failure criterion is unnecessary to provide for reasonable assurance of adequate protection of public health and safety.*

In 2019, NRC staff issued staff review guidance for in-vessel downstream effects associated with debris generated under LOCA conditions (Reference 7). This guidance provides a graded approach of review criteria, based on the information, evaluations, and analyses summarized in the Technical Evaluation Report of In-Vessel Debris Effects (TER), to determine the level of plant-specific review activity needed to establish compliance. The guidance represents a shift in regulatory compliance from high probability, plant-specific demonstrations, to reasonable assurance via generic demonstrations. Even for plants deemed well outside the applicability of generic demonstrations, NRC left open the use of risk-informed approaches crediting break size probability.

> *Failure to meet the review criteria specified herein does not necessarily imply regulatory noncompliance. However, if the above review criteria are not satisfied, further plant-specific evaluation may be necessary to demonstrate compliance. For example, if one or more key parameters associated with the AFP analysis in WCAP-17788 is significantly outside the bounding assumptions, alternatives should be considered to demonstrate LTCC adequacy. Two examples of alternative*

> *plant-specific evaluation methods include: (1) a risk-informed request demonstrating that break sizes of the magnitude required to generate such debris loadings are sufficiently unlikely; and…*

In 2013, the Commission requested their staff to include an alternate risk-informed approach for addressing debris effects on long-term core cooling following a LOCA. The draft final 50.46c rule, Emergency Core Cooling System Performance During Loss-of-Coolant Accidents (Reference 8), includes this alternative risk-informed approach. While this draft final rule is still being promulgated (with Commission since 2016), the alternative risk-informed approach was directed by the Commission as a means to resolve debris issues using risk, without the need for an exemption.

In 2010, NRC staff provided the Commission with a risk-informed alternative final rule for approval and publication via SECY-10-0161 (Reference 9). The final rule was summarized as follows:

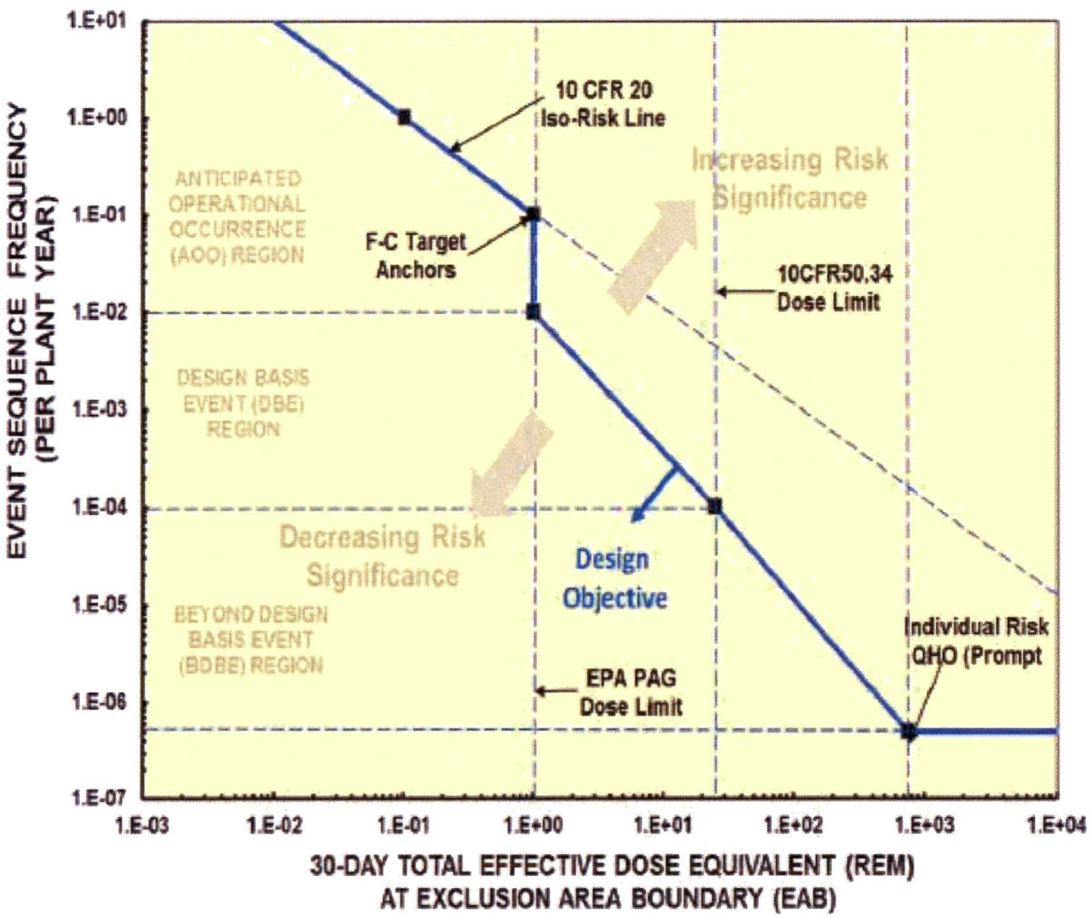
> *The final rule will establish an alternative set of risk-informed ECCS requirements in Title 10 of the Code of Federal Regulations (10 CFR) Part 50.46a with which licensees may choose to comply in lieu of meeting the current requirements in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." The rule divides the current spectrum of LOCA break sizes into two regions. The division between the two regions is delineated by the transition break size (TBS). The first region includes small-size breaks, up to and including the TBS. The second region includes breaks larger than the TBS, up to and including the double-ended guillotine break (DEGB) of the largest reactor coolant system pipe. These larger breaks are considered to have a much lower likelihood than the smaller breaks in the first region. Under the new rule, the ECCS design requirements for pipe breaks less than the TBS are the same as the requirements for all breaks under the current 10 CFR 50.46 ECCS rule. By contrast, under the new rule, the ECCS design requirements for pipe breaks larger than the TBS may be analyzed using less conservative assumptions based on their lower likelihood.*

While the final risk-informed alternative rule was never promulgated, the conceptual treatment of analytical models, inputs, and assumptions based on frequency and the staff's estimated break frequencies supporting the rule are important for this discussion.

In 2003, the Commission (SRM-SECY-02-0057) approved the staff's recommendation for redefining the design basis large-break LOCA in view of the apparent low risk associated with such events and directed the staff to provide a proposed rule change that allows for a risk-informed alternative (Reference 10).

Figure 3-2
Frequency – Consequence Target

(Source: NEI 18-04 Figure 3-1)



4.0 REFERENCES

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2. SECY-18-0060, *Achieving Modern Risk-Informed Regulation*, May 23, 2018.
3. SRM-SECY-21-0109, *Staff Requirements - SECY-21-109 - Rulemaking Plan on Use of Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors*, March 16, 2022.
4. Regulatory Guide 1.233, *Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors*, June 2020.
5. NEI Technical Report 18-04, *Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development, Revision 1*, August 2019.
6. SRM-SECY-19-0036, *Staff Requirements – SECY-19-0036 – Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves*, July 2, 2019.
7. NRC Memo, *U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses*, September 4, 2019.
8. SECY-16-0033, *Draft Final Rule – Performance-Based Emergency Core Cooling System Requirements and Related Fuel Cladding Acceptance Criteria*, March 16, 2016.
9. SECY-10-0161, *Final Rule: Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements (10 CFR 50.46a)*, December 10, 2010.
10. SRM-SECY-02-0057, *Staff Requirements – SECY-02-0057 – Update to SECY-01-0133, Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)*, March 31, 2003.
11. NRC Memo, V. Cusumano to M, Gavrilas, "Technical Evaluation Report of In-Vessel Debris Effects," June 13, 2019.