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Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors

Comment On: NRC-2020-0034-0005
Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors

Document: NRC-2020-0034-DRAFT-0011
Comment on FR Doc # 2023-19452

Submitter Information

Email: holdercm@westinghouse.com
Organization: PWR Owners Group

General Comment

See attached file(s).

Dear Ms. Carrie M. Safford,

This letter transmits the PWR Owners Group comments on the Regulatory Basis Supporting Increased Enrichment of Conventional and Accident Tolerance Fuel Designs for Light-Water Reactors. The PWROG appreciates the opportunity to comment on this regulatory basis.

If you have any questions, please do not hesitate to contact me at (815) 520-3023 or Mr. D. Olinski, Executive Director of the PWR Owners Group, Program Management Office at (412) 374 3025.

Sincerely yours,

James Lynde
Chairman and COO
PWR Owners Group

Attachments

OG-24-15

Program Management Office
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066

January 16, 2024

OG-24-15

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
ATTN: Rulemakings and Adjudications Staff

Subject: PWR Owners Group
PWROG Comments on the Regulatory Basis Supporting Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors, 88 FR 61986; Docket ID NRC-2020-0034

Dear Ms. Carrie M. Safford,

This letter transmits the PWR Owners Group comments on the Regulatory Basis Supporting Increased Enrichment of Conventional and Accident Tolerance Fuel Designs for Light-Water Reactors. The PWROG appreciates the opportunity to comment on this regulatory basis.

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Sincerely yours,

James Lynde
Chairman and COO
PWR Owners Group

DRPB

Attachment: PWROG Comments on the Regulatory Basis Supporting Increased Enrichment of Conventional and Accident Tolerance Fuel Designs for Light-Water Reactors (Non-Proprietary)

cc: PWROG PMO
PWROG Analysis Committee
PWROG Licensing Committee
PWROG Risk Management Committee
J. Lynde, PWROG/Constellation
B. Mount, PWROG/Dominion Energy
B. Dolan, PWROG/TVA
D. Rapp, PWROG/Energy Harbor
D. Richards, PWROG/STP
J. Vaughan, PWROG/Duke Energy
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D. Mirizio, PWROG
T. Laubham, PWROG
L. Fields, US NRC

PWROG 50.67 COMMENT 1

Statement in Guidance:
Section 5.7 Questions for Public Comment (page 5-3)

> Would the numerical selection of the control room *design* criteria be better aligned with regulations designed to limit occupational exposures during emergency conditions (e.g., 10 CFR 20.1206, "Planned special exposures," and 10 CFR 50.54(x)), or regulations designed to limit annual occupational radiation exposures during normal operations (e.g., 10 CFR 20.1201, "Occupational dose limits for adults," specifically the requirements in 10 CFR 20.1201 (a)(1)(i))? Please provide a basis for your response.

Comment:
The Control Room *design* criteria should be more consistent with the various U.S. and international organizations recommendations for emergency dose limitations up to 25 rem TEDE.

Proposal:
The numerical selection of the control room *design* criteria should be better aligned with regulations designed to limit occupational exposures during emergency conditions.

Basis for Proposal:
As described in the regulatory basis document, the control room design criteria are not intended to be operational limits and should not be used to imply what an acceptable exposure is during emergency conditions. However, the criteria are used to demonstrate that a plant design is adequate to maintain a habitable control room during hypothetical events up to and including the maximum hypothetical accident (MHA). Therefore, consistency with 10 CFR 20 and 10 CFR 50.54 for occupational exposures during normal operations and emergency conditions is not appropriate. The 10 CFR 50.67 and GDC 19 criteria should be viewed as complimentary to the occupational dose limits for normal and emergency conditions.

See Comment #2 for additional details.

PWROG 50.67 COMMENT 2

Statement in Guidance:
Section 5.7 Questions for Public Comment (page 5-3)

> Would a graded, risk-informed method, to demonstrate compliance with a range of acceptable control room *design* criterion values instead of a single selected value, such as the current 5 rem (50 mSv) TEDE, provide the necessary flexibilities for current and future nuclear technologies up to but less than 20.0 weight percent U-235 enrichment? Please provide a basis for your response.

Comment:
A single value (25 rem) is suggested in the rulemaking document for the control room design criterion in 10 CFR 50.67 and GDC 19. The PWROG agrees with a value of 25 rem for rulemaking updates to 10 CFR 50.67 and GDC 19. The PWROG also recommends that a graded, risk-informed method be considered for establishing the limits within the regulatory guidance. This approach is recommended to provide the necessary flexibilities for current and future nuclear technologies.

Proposal:
The PWROG agrees with single value of 25 rem in the affected regulations. This provides the NRC with the flexibility to control application of the limit within the regulatory guidance similar to how the off-site dose limits are controlled.

The PWROG recommends that the NRC consider a simple, graded approach in order to bin the radiological consequences scenarios in support of issuing RG 1.183, Rev. 2 in a timely manner to support the industry desire to deploy batch loads of fuel enriched above 5 weight percent U-235 by the mid-to-late 2020s. As an example of this approach for the regulatory guidance, two criteria can be established in order to bin the radiological consequences scenarios. The first criterion would encompass events which typically postulate fuel failure during a reactor transient (e.g., Maximum Hypothetical Accident (MHA), etc.) with the Control Room dose limited to 25 rem TEDE. This would be consistent with the current maximum value for offsite dose locations in 10 CFR 50.67. The second criterion would encompass more frequent design basis events which do not postulate fuel failure during a reactor transient (e.g., steam generator tube rupture, etc.) with the Control Room dose limited to 10 rem TEDE. This would be the lower range suggested by the NRC staff in the rulemaking document.

PWROG 50.67 Comment 5 offers initial thoughts on a potential graded, risk-informed approach that could be established within the regulatory guidance (e.g., RG 1.183, Rev. 3). The conversion to a risk-informed dose analysis design and licensing basis is recognized as being a significant change that will require additional time beyond that allowed by the current industry desire to deploy batch loads of fuel enriched above 5 weight percent U-235 by the mid-to-late 2020s.

Basis for Proposal:
A graded, risk-informed method to determine the Control Room acceptance criteria would be consistent with the regulatory limits provided in Regulatory Guides 1.183, Revisions 0 and 1 for the offsite dose criteria, which establish different limits based on the relative probabilities of the events.

PWROG 50.67 COMMENT 3

Statement in Guidance: Section 3.1.1: 10 CFR 50.67 Accident Source Term Requirements (pages 3-2 and 3-3)
Comment: The staff has identified three evaluated alternatives for 10 CFR 50.67 rulemaking. It was identified that the NRC recommended Alternative 2 (Rulemaking to revise the control room design criteria). The PWROG agrees with this recommendation; however, the PWROG also supports additional dose evaluation enhancements for Alternative 2, as well as, continued future development of Alternative 3 (update the Current Regulatory Guidance with Revised Assumptions and Models).
Proposal: The PWROG understands that the approach for Alternative 2 is to increase the dose limit without specific updates to the guidance expected in future revisions of Regulatory Guide 1.183 (i.e., Revision 2). The PWROG recommends that the staff consider the incorporation of some modeling improvements into the near-term future guidance updates. These modeling improvements would not be risk-informed dose analyses (scope of Alternative 3), but rather deterministic applications utilizing best-estimate methods which incorporate statistical selections of inputs and assumptions. Deterministic best estimate applications combined with accurate uncertainty quantification would result in more accurate margin determination (i.e., more accurate event mitigation and hence safer plant operation) along with a bounding licensing basis determination (i.e., conservative results). The PWROG also understands that Alternative 3 is a long-term initiative that would not satisfy the needs of some licensees to implement increased enrichment core designs in the timelines of interest. However, improved dose models, such as risk-informed dose analysis models, may be needed for future operation of the nuclear fleet. The PWROG proposes that the NRC develop (or continue to develop) Alternative 3 in order to minimize the delay between the deployment of Alternative 2 and a future deployment of Alternative 3.
Basis for Proposal: See the discussion above.

PWROG 50.67 COMMENT 4

Statement in Guidance:
Section 3.1.1.2: 10 CFR 50.67 Alternative 2 (pages 3-2 and 3-3)

Comment:
The proposed rulemaking focuses only on the Control Room dose acceptance criteria in 10 CFR 50.67 and GDC 19. The regulations also identify offsite dose criterion of 25 rem TEDE for the limiting 2-hour period at the Exclusion Area Boundary (EAB) and for the duration of the accident at the Low Population Zone (LPZ). While it was identified that these limits are not expected to be challenged for the increased enrichment changes, the applicable regulatory guidance, such as Regulatory Guide 1.183, limits the offsite acceptance criteria to either 10% (small fraction of) or 25% (well within) the 10 CFR 50.67 criteria.

Some licensees are limited at offsite dose locations for a variety of design basis radiological consequence events with the "small fraction of" and "well within" the regulatory guide acceptance criteria. If the need for relief is anticipated for the Control Room acceptance criteria, then relief should also be anticipated for the offsite dose locations.

Proposal:
It is proposed that the dose criteria in Table 7 of RG 1.183, Revision 1 be updated to relax the limits for the offsite locations that are a "small fraction of" and "well within" the full acceptance criterion of 25 rem TEDE (AST).

The PWROG supports reclassifying the events such that the following offsite dose limits apply:

- Full 10 CFR 50.67 limit (25 rem TEDE) for the MHA and other events which consider fuel failure during a reactor transient.
- An additional dose limit of 10 rem TEDE for events which do not consider fuel failure during a reactor transient.

Basis for Proposal:
A revision to the regulatory guidance acceptance criteria for the "small fraction of" and "well within" events would provide the relief at the offsite dose locations needed to adopt core designs with increased enrichment. This relief would be consistent with the proposed change in rulemaking to increase the Control Room acceptance criteria from 5 rem TEDE to a limit which is still within the full 25 rem TEDE limit. The proposed values are consistent with the example limits from PWROG 50.67 Comment 2 above.

PWROG 50.67 COMMENT 5

Statement in Guidance:
Section 5.7 Questions for Public Comment (page 5-3)

> Would a graded, risk-informed method, to demonstrate compliance with a range of acceptable control room design criterion values instead of a single selected value, such as the current 5 rem (50 mSv) TEDE, provide the necessary flexibilities for current and future nuclear technologies up to but less than 20.0 weight percent U-235 enrichment? Please provide a basis for your response.

Comment:
This comment is based on the premises that Alternative 2 and Alternative 3 for 10 CFR 50.67 are not necessarily mutually exclusive but are rather a potential natural evolution of a risk-informed and performance-based approach for radiological dose consequence analyses.

In the spirit of establishing a more risk-informed approach to control room dose limits, the NRC could structure the allowable control room dose along the lines of that proposed in NEI 18-04 using a frequency-consequence map (see Figure 1). A number of future nuclear technologies are planning to use the framework and construct of the Licensing Modernization Process (LMP) in NEI 18-04 which is endorsed by the NRC in RG 1.233. In the LMP context, the dose limits are risk informed based on the frequency and consequence of the likely challenges. A simplified application of that process to control room dose limits is presented in Figure 1, where a red line, representing a frequency-consequence (F-C) map for the control room (CR), is superimposed on the NEI 18-04 F-C map. In this illustration, events with higher frequencies and lower consequences, such as anticipated operational releases, [uncomplicated steam generator releases], could have allowable dose limits maintained at 10 rem (frequency < $\sim 10^{-3}$/year), typical design basis transients with significant clad rupture (e.g., LOCAs) have estimated occurrence frequencies on the order of 10^{-4}/year (as illustrated in the figure), could have allowable dose limits raised to 20 rem and the maximum hypothetical accident (MHA) involving a partial core melt would have a frequency less than $\sim 10^{-5}$/year the allowable dose could be raised to 25 rem. The intent of this approach is for the frequencies associated with individual scenarios (e.g., initiating event frequencies, consequential failure probabilities and component/recovery unavailability/failures) to be plant-specific and consistent, where applicable, to the accepted values used in the PRA. This graded approach provides flexibility in design and ensures a high level of protection commensurate with the nature of the challenge. However, tiered limits may require an additional step of binning challenges based on frequency and potentially may complicate the analysis.

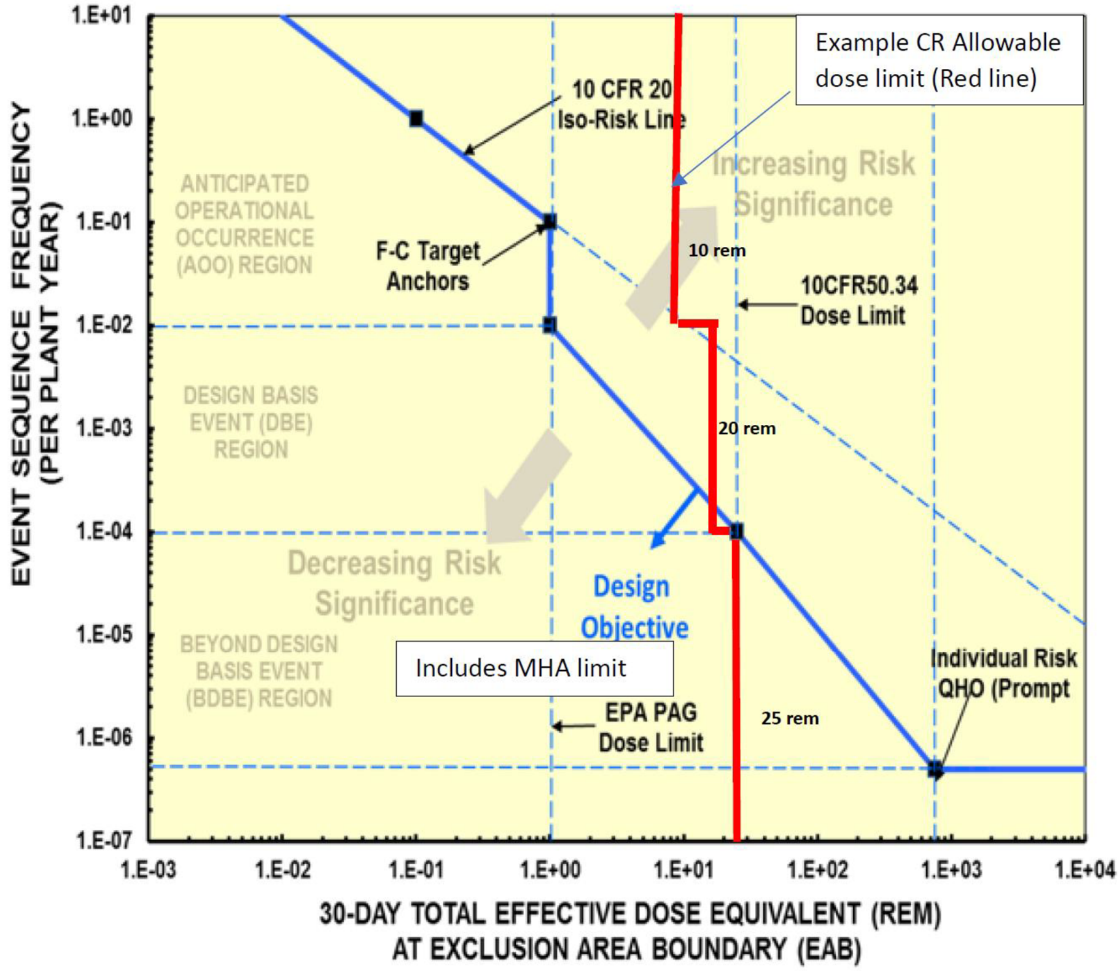

Figure 1: Potential Control Room Dose Limit (Red)

Proposal:
The control room design criteria should be graded, with a risk-informed method based on event frequency, to demonstrate compliance with a range of acceptable control room design criterion values, with a single selected value representing a maximum dose criterion for events of the lowest frequency. This provides the necessary flexibilities for current and future nuclear technologies.

The PWROG supports further development of Alternative 3 as a longer-term initiative to introduce the concept of a scaled consequence limit based on event frequency, that is consistent with NEI 18-04, so that a consistent framework to that which is already being applied in the LMP program may be applied to the current operating nuclear fleet.

It is noted that, since RG 1.174 specifically addresses a risk-informed framework in the context of CDF and LERF as figure of metrics, to fully implement a risk-informed approach for the assessment of control room dose, a revision to RG 1.174 may be considered to extend its applicability to scenarios that do not necessarily relate to Core Damage as defined in PRA.

Basis for Proposal:
A plant-specific example is shown here for illustrating the concept. Estimated Control Room dose values from a PWR **[current values not accounting for higher enrichment impacts]** are reported below:

Accident	TEDE (rem)
Loss of Coolant Accident (LOCA)	4.5
Steam Generator Tube Rupture (SGTR)	2.6
Loss of AC Power (LOAC)	<2.9

Plotting the estimated CR dose in the consequence vs. frequency space discussed above in Figure 1 (note that the EAB thresholds from the LMP are also provided for reference), the above results are essentially straight lines at a constant dose (see Figure 2).

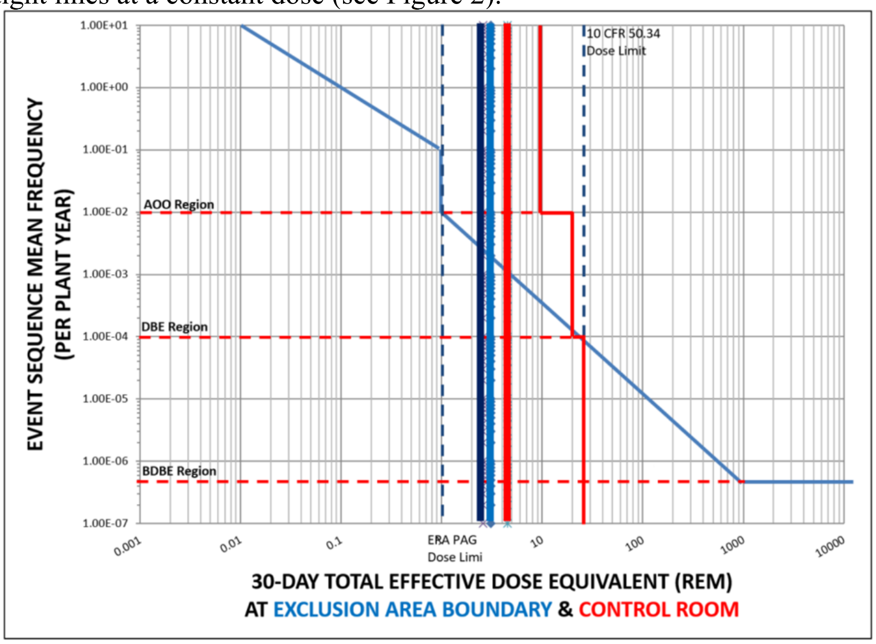

Figure 2: Control Room dose (no frequency) for **LOCA**, **SGTR** and **LOAC**

Figure 3 provides a plot of representative calculated CR doses for radiological release scenarios associated with various initiating events. A rough estimate of the frequency of the scenarios was performed by simply assigning the generic initiating event frequency (IEF) normally used in the PRA for Large LOCA, SGTR and LOOP (from NUREG-CR/6928) to the design basis control room dose associated with that initiating event. This is considered to be a conservative estimate of the frequency of the specific scenario evaluated in the dose estimate because additional failures (e.g., concurrent LOOP, single failure, maximum TS conditions, etc…) have not been assessed for a realistic frequency estimate that would further reduce the design basis scenario frequencies plotted in Figure 3. The intent of this figure is to show the progressive margin towards the proposed thresholds. That margin is essential to accommodate the anticipated impacts from increased enrichment fuel management strategies which are the impetus for the proposed rulemaking.

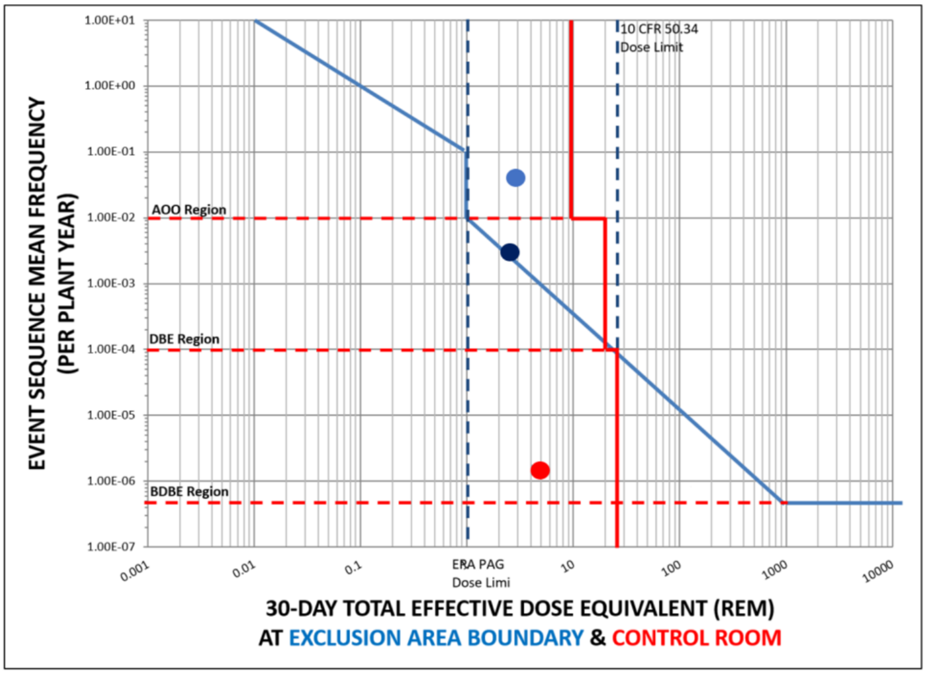

Figure 3: Control Room dose (with frequency estimate) for **LOCA**, **SGTR** and **LOAC**

PWROG 50.68(B) COMMENT 1

Statement in Guidance:
Table ES-1 Recommended Rulemaking Actions and Table 8-1 (pages vii and 8-1)

50.68	Criticality accident requirements	Rulemaking to eliminate the enrichment level in criticality accident requirements and rely on a minimum k_{eff} value (Alternative 3)

Comment:
The PWROG agrees that Alternative 3 is the preferred rulemaking option for 10 CFR 50.68. The PWROG feels that Alternatives 1 and 2 should not be pursued principally because the method for determining the new k_{eff} limit is unknown. Alternative 3, on the other hand, can more readily build upon the previous work documented in Regulatory Guide 1.240 with less new work needing to be performed.

Proposal:
None.

Basis for Proposal:
See the comment above.

PWROG 50.68(B) COMMENT 2

Statement in Guidance:
Table ES-1 Recommended Rulemaking Actions and Table 8-1 (pages vii and 8-1)

50.68	Criticality accident requirements	Rulemaking to eliminate the enrichment level in criticality accident requirements and rely on a minimum k_{eff} value (Alternative 3)

Appendix B, 3.3.1 Description of Alternative 3 (page B-5) – bold font and italics added for emphasis
Under this alternative, the NRC would pursue rulemaking to remove the current 5.0 weight percent U-235 limit in 10 CFR 50.68(b)(7) and instead allow for a plant-specific criticality safetylimit based on the limit specified in TS 4.3.

"The wording in 10 CFR 50.68(b)(7) would be changed to specify a process in which the maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to the value specified by the licensee or applicant in their TS 4.3. TS 4.3 is required by 10 CFR 50.36(c)(4). When licensees or applicants apply for a fuel transition LAR for enrichment increases, the licensee would have to demonstrate that the k_{eff} limits specified by 10 CFR 50.68(b)(2), (b)(3), and (b)(4) would be maintained at the increased enrichment levels at the same value, probability, and confidence levels."

Comment:
The summary description of recommended rulemaking action for 10 CFR 50.68 in Table ES-1 and in Table 8-1 do not appear to be consistent with the description in Appendix B. The description in Appendix B implies that this alternative would rely on the plant specific Technical Specifications to limit the allowed maximum nominal U-235 enrichment versus relying on the minimum k_{eff} as summarized in Table ES-1.

The NUREG-143X series of Standard Technical Specifications (STS) contains two relevant criticality requirements relevant to this discussion as shown below.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of [4.5] weight percent,

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in [Section 9.1 of the FSAR],

Proposal:
It is recommended that the NRC use the following update to the description provided in Table ES-1 and Table 8-1 for Alternative 3.

> Rulemaking to eliminate the enrichment level in criticality accident requirements and rely on plant-specific criticality requirements specified in their Technical Specifications (TS) (Alternative 3 of Appendix B).

Basis for Proposal:
The proposed summary description of Alternative 3 for the recommended rulemaking actions continues to acknowledge that the plant specific TS contain the criticality requirements in a more general fashion.

If the NRC does not agree with the proposed change, then the phrase “minimum k_{eff} value” should be corrected to “maximum k_{eff} value” to align with the intent of the k_{eff} safety limit.

PWROG 50.68(B) COMMENT 3

Statement in Guidance: Appendix B, Section 3.0 Discussion of Alternatives There are several references to TS 4.3 as containing criticality accident requirements.
Comment: This statement is not generically true for the current operating light water reactor fleet. Several plants have not adopted the format and content of the NUREG-143X series of STS and use the previous STS (e.g., NUREG-0452 for Westinghouse NSSS Plants).
Proposal: Do not include an explicit reference to a TS number in the changes to the rule.
Basis for Proposal: Inclusion of a reference to a specific TS number as containing a plant’s criticality requirements would result in the need for licenses to request an exemption from the rule or to revise their TSs to be consistent with the criticality requirements with the revised rule.

PWROG 50.68(B) COMMENT 4

Statement in Guidance: Appendix B, Section 2.0 Regulatory Issue (pages B-2 and B-3) In NUREG-1520, "Standard Review Plan for Fuel Cycle Facility Applications," Revision 2, issued June 2015 (ML15176A258), chapter 5 and appendix 5B outline the regulatory guidance for establishing nuclear criticality safety (NCS) analyses. Section 5.4.3.1.7.3 of NUREG-1520 specifies enrichment as a NCS control parameter. Changing that parameter requires reevaluation and reestablishment of a new calculated minimum margin of subcriticality (MMS) (MMS = 1 - keff) safety limit to ensure the same level of safety margin (conservatism) as in the current rule.
Comment: Why is NUREG-1520 cited in the regulatory basis assessment of 10 CFR 50.68? NUREG-1520 applies to fuel cycle facilities, not operating reactors.
Proposal: N/A
Basis for Proposal: N/A

PWROG 50.68(B) COMMENT 5

Statement in Guidance: Appendix B, Section 2.0 Regulatory Issue (page B-3) Increasing enrichment would require lowering the k_{eff} safety limit; however, the guidance in NUREG-1520 does not provide a simple calculational method to determine the safety equivalent keff for an enrichment of up to but less than 20.0 weight percent U-235. Section 3.2.2 Assessment of Alternative 2 (page B-5) and a similar statement in Section 3.1.2.2 (page 3-4) This would entail conducting research to determine the appropriate lower keff safety limit for enrichment up to but less than 20.0 weight percent U-235 and then apply that new safety limit to fuel enriched to greater than 5.0 and less than 20.0 weight percent U-235.
Comment: What is the basis for this statement and why is further research required? Is this more of a concern for going above a specific enrichment such as 10 weight percent U-235 as opposed to small increases above 5 weight percent U-235? Industry efforts to date are evaluating U-235 enrichments above 5 weight percent intended for fuel cycles associated with power uprates and longer cycles (i.e. 24-month cycles) have shown no significant trends in bias or bias uncertainties up to the enrichments (as high as 8 weight percent U-235) expected to be utilized in these initiatives. Additionally, assuming that the biases and uncertainties increase with enrichment, the probability and confidence requirements of 10 CFR 50.68(b) would force the best estimate k_{eff} to decrease to meet the 10 CFR 50.68(b) k_{eff} safety limits. So it is unclear why the k_{eff} safety limits would need a further reduction for increased enrichment.
Proposal: Provide further justification why reducing the k_{eff} limit is needed and why extra research is needed.
Basis for Proposal: N/A

PWROG 50.68(B) COMMENT 6

Statement in Guidance:
Table 2-1 (page 2-15)

Table 2-1 Guidance to Update

Regulation	Guidance to Update
10 CFR 50.67(b)(2)(iii)	Regulatory Guide 1.183***
10 CFR 50.68	None

Comment:
The PWROG feels that Regulatory Guide 1.240, Fresh and Spent Fuel Pool Criticality Analyses should be revised, to reflect the rulemaking efforts associated with 10 CFR 50.68.

Proposal:
Add Regulatory Guide 1.240 to Table 2-1 as guidance that needs to be revised in coordination with future rulemaking on 10 CFR 50.68.

Basis for Proposal:
Reg Guide 1.240 is an important connection to 10 CFR 50.68. A revision to Regulatory Guide 1.240 will presumably require some coordination with the industry through NEI and/or EPRI as the primary document referenced by Regulatory Guide 1.240, NEI 12-16, is revised in part to address future rulemaking.

PWROG 50.68(B) COMMENT 7

Statement in Guidance: Section 3.1.2.3 Alternative 3 (page 3-4) The NRC would perform research into the criticality safety response of increasing enrichments beyond current levels…
Comment: Similar to Comment 4 which discussed research. The document should describe why the research is needed, who would perform the research, and what the timeline is for performing and reporting on the research.
Proposal: Provide justification for why additional research is needed.
Basis for Proposal: N/A

PWROG 50.68(B) COMMENT 8

Statement in Guidance: Section 8 Conclusion (page 8-1) … (2) amend the criticality accident requirements in 10 CFR 50.68 to remove the current enrichment limit and instead allow for a plant specific criticality safety limit…
Comment: The phrase “plant specific criticality safety limit” is not clear.
Proposal: Suggest replacing “criticality safety limit” with “upper enrichment value supported by the licensee’s criticality safety analysis to meet the current k-eff limits specified in 10 CFR 50.68.”
Basis for Proposal: N/A

PWROG 50.68(B) COMMENT 9

Statement in Guidance: Section 10, References (pages 10-1 thru 10-9)
Comment: Regulatory Guide 1.240 should be included in the reference list.
Proposal: Add Regulatory Guide 1.240, "Fresh and Spent Fuel Pool Criticality Analyses," to the reference list.
Basis for Proposal: PWROG recommends adding Regulatory Guide 1.240 to Table 2-1. Therefore, it would also need to be added to the reference list.

PWROG 50.68(B) COMMENT 10

Statement in Guidance: Appendix B, Section 3.2.2 Assessment of Alternative 2 (page B-5) … because the larger enrichment upper bound entails a higher administrative statistical burden to maintain the probability and confidence levels required by 10 CFR 50.68(b)(2)-(4). … because the larger enrichment upper bound entails a higher administrative statistical burden to maintain the keff acceptance criteria at the required probability and confidence levels.
Comment: What does “higher administrative statistical burden” mean? What analysis shows the statement in question to be true?
Proposal: Please explain the concept of “higher administrative statistical burden” and why and how it applies to this situation.
Basis for Proposal: N/A

PWROG DISPERSAL COMMENT 1

Statement in Guidance:
Appendix F, Section 9.0 Staff Recommendation (pages F-26 and F-27)

(1) Are there any other alternatives not described in appendix F of the regulatory basis on FFRD that the NRC should consider? Are there elements of the alternatives presented or other alternatives that the NRC should consider? Please provide a basis for your response.

(6) What are the pros and cons of each alternative, including the degree to which each alternative is consistent with the principles of good regulation?

Comment:
The NRC should leverage an integrated decision-making process based on risk insights for the assessment of fuel dispersal during a Loss of Coolant Accident (LOCA) similar to the final GSI-191 resolution for PWR in-vessel debris effects (IVDEs). Some of the five options already contain minor elements of this approach by describing dispersal during a LBLOCA as beyond the design basis.

The primary consideration in evaluating the pros and cons of each of the options is a balance between the regulatory certainty provided by the option against the anticipated timeframe to complete development of the required guidance. Experience from the 10 CFR 50.46a rule development indicates it is unlikely for Large Break LOCA to be removed from the regulations and the timeline for new rulemaking will take several years to complete. Experience from the GSI-191 PWR IVDE testing supports the perception that testing to develop acceptable licensing models analyzing fuel dispersal or to quantify an appropriate source term will take many years to complete (see the Basis for Proposal for additional information). The many years expected for either rulemaking or additional testing to be completed (based on the before mentioned examples) would be challenging to support the industry needs associated with implementing ATF/LEU+/HBU to remain a competitive, low cost, source of carbon free energy in the mid-2020s as requested by NEI (Ref. ML22172A135). Conversely, determination of the safety significance, when considering the low initiating event frequencies, would likely demonstrate that fuel dispersal has a low safety significance. If a low safety significance is indeed determined, policy making commensurate with the risk can be pursued.

Cohesion around a single solution path commensurate with the risk significance would assist in aligning the staff and the industry around a single solution ensuring the effective use of resources and reducing the burden of pursuing different resolutions.

Proposal:
The NRC should consider the lessons learned from the final resolution for PWR in-vessel debris effects (IVDEs) related to GSI-191 when developing proposed rule changes to address fuel dispersal. An integrated decision-making process based on risk insights would assist in resolving fuel dispersal in a manner commensurate with its safety significance in an efficient and timely manner. Therefore, it is recommended that this process be considered for applicability to all proposed options to resolve fuel dispersal. Resolution of the IVDEs would have occurred sooner had the integrated decision-making process based on a risk assessment been pursued at the beginning of the resolution of GSI-191.

Basis for Proposal:
As directed by the Commission, risk should be considered in future NRC efforts associated with increased enrichment. Indeed, many of the alternatives summarized in Appendix F depend on risk insights. While risk insights are not excluded from the current regulation, their utilization is not explicitly defined. The recent PWROG program for GSI-191 is an example of this. The final resolution for in-vessel debris effects (IVDEs) related to GSI-191 successfully used risk insights within the current regulatory structure. However, it took considerable time and resources before risk insights were included

in the solution. A review of the history of addressing IVDEs and the final NRC conclusion may provide insights into efficiently addressing FFRD or other core cooling issues potentially arising in the future.

BRIEF BACKGROUND OF GSI-191

GSI-191 was identified as a core cooling issue for US PWRs in 2004. The initial emphasis was on clogging the sump strainers that could result in exceeding the net positive suction head (NPSH) of the emergency core cooling system (ECCS) pumps leading to their failure and ultimately compromising long-term core cooling. It was also recognized that downstream effects of debris could have the same effect even if the NPSH of the pumps was not exceeded.
The initial attempt to resolve IVDEs was documented in WCAP-16793-NP, Rev. 0. The solution attempted to provide "*reasonable assurance* that long-term cooling for PWRs will be established and maintained for post-LOCA considering the presence of debris in the RCS and core." The analyses to support this conclusion were generic in nature and bounded the US PWR fleet; however, no quantitative limit on the amount of acceptable debris in the core was provided. Subsequently, the PWROG undertook a test program to quantify a tolerable amount of debris (documented in WCAP-16793-NP-A, Rev. 2). This testing was designed to conservatively bound all types of debris that could be present at the core inlet at any time following a LOCA. Further, the testing did not credit beneficial design characteristics of the reactor vessel. After three years of testing, the result was the very conservative limit of 15 g/FA.
The PWROG recognized that this limit was too restrictive for the majority of the US PWR fleet and began a new program to credit insights gained from the previous testing. This program is documented in WCAP-17788-P and took five years to prepare and submit for review. The testing and analysis took credit for the timing of the chemical precipitate formation (a major challenge to core cooling) and design characteristics of the reactor vessel (i.e., alternate flow paths). In all, six volumes of material were submitted for NRC review and approval. Each volume considered conservative inputs to each portion of the solution because it was unclear what the NRC-standard should be for review and approval.

FINAL RESOLUTION OF IVDES FOR GSI-191

After many years of back and forth between the industry and the NRC staff, the final resolution of IVDEs relied on an integrated decision-making process based on a risk assessment. The details are described in the Technical Evaluation Report (TER) for IVDEs issued on June 13, 2019 (ML19073A044) and summarized here.

The NRC describes their integrated decision-making process in Section 5.5 of the TER:

> "Integrated decisionmaking considers information from a variety of sources including risk insights, traditional engineering evaluations and insights, operational experience and historical plant performance, engineering judgment, and current regulatory requirements. In addition, evaluation of DID and safety margins are an important part of the integrated decisionmaking process."

It is essentially divided into two parts:

1. Determination of safety significance
2. Identify safety margins (or defense in depth (DID) analyses)

A quantitative analysis of the risk associated with IVDEs following a hypothetical LOCA at each U.S. PWR was not required. Instead, the NRC staff combined risk insights with its engineering evaluation, including the low initiating frequency of LOCAs that can challenge Long Term Core Cooling (LTCC) via IVDEs to determine the safety significance of IVDEs. They used existing decision-making guidance included in LIC-504, NUREG/BR-0058, RG 1.174, and the Significance Determination Process. They concluded that IVDEs were of low safety significance. The resolution of the issue depended on a defense in depth analysis and safety margin assessment. The safety margins associated with the evaluation of IVDEs depend on plant-specific parameters used in the associated analyses discussed in the TER. These margins and DID analyses are summarized in Section 5.5.3 of the TER. To ensure the applicability of the staff assessment, plants must demonstrate that their plants meet certain inputs and boundary conditions that were defined in the NRC review guidance for IVDEs (ML19228A011). This review guidance provided the framework for licensees to complete their response to Generic Letter (GL) 2004-02. The NRC requested information from licensees via GL 2004-02 to assist in determining, on a plant-specific basis, the impact on sump screen performance and other related effects of extended post-accident operation with debris-laden fluids based on GSI-191.

CONCLUSION

Through integrated decision-making, the NRC determined “that for the operating PWR fleet, and within the range of assumed plant parameters, debris penetrating the sump strainer is very unlikely to prevent adequate LTCC following LOCAs via IVDEs.” A similar process can be applied to FFRD. For FFRD, fine fuel fragmentation and dispersal only occurs in high burnup fuel rods with cladding burst following a LOCA. Certain LOCA scenarios, dependent on the break size and break location, are needed to reach cladding burst temperatures. Further, core cooling would only be challenged by dispersal from certain burst locations at specific times following the LOCA. This information can be used to define fuel dispersal as a low probability event. To supplement the risk assessment, extensive testing and analysis of the FFRD have been done and are in process. These tests along with expert elicitation (i.e., Phenomena Identification and Ranking Table Panels) and analysis on the effects can be used to form the basis for defense-in-depth analysis of the consequences of fuel dispersal.

While this process was used successfully to address IVDEs for GSI-191, it took 15 plus years to arrive at a conclusion and provide the necessary guidance for licensees to address IVDEs. It has also taken 2 – 3 years for utilities to provide their responses and receive approval for their response to GL 2004-02. For FFRD, the opportunity exists to engage this process much earlier and provide a more expedient resolution to address FFRD following a LOCA such that utilities can begin to implement economical, high burnup core designs prior to 2030.