WHITE PAPER

NEI White Paper: Proposed Control Room Dose Acceptance Criteria Supporting RG 1.183 R2

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Goal, Objectives, and Scope

The goal of this paper is to propose a framework that identifies a range of acceptable values for risk-informed control room design criteria based on sound regulatory precedents and leveraging scientific recommendations. The purpose is to enable deterministic evaluations, using traditional radiological consequence analyses, within the risk-informed boundaries provided by the proposed new control room design criteria.

The following objectives were established to ensure that the proposed criteria meet regulatory requirements, are backed by a sound technical basis, and support efforts to consistently license fuels enriched to greater than 5.0 and less than 20.0 weight percent uranium-235 (U-235) outside the exemption process. The objectives for the proposed criteria are to:

- Continue to provide an acceptable level of control room habitability design for design-basis events necessary to provide reasonable assurance that the control room would continue to be effectively staffed and operated to mitigate the effects of postulated accidents and protect public health and safety.

- Continue to maintain multiple lines of defense ensuring the effectiveness of physical barriers between radiation sources from workers, the public, and the environment both during normal operations and accident conditions.

- Align with various U.S. and international organizations recommendations for emergency dose limitations.

While the current LWR fleet is expected to use fuel enriched to less than 10.0 weight percent U-235, the scope of this paper is intended to support industry interest in the use of fuels enriched to greater than 5.0 weight percent and less than 20.0 weight percent U-235.

Key Insights and Conclusions

The key insights and conclusions from this white paper are:

- Industry supports a risk-based control room dose acceptance criteria of 10 rem to 25 rem total effective dose equivalent (TEDE).

- The intent of maintaining a habitable control room is to ensure that operators can respond in the unlikely event of an accident, to protect the public. Considering the importance of control room habitability, the proposed value of 25 rem TEDE as an upper bound for the control room design criteria is aligned with the various U.S. and international organizations recommendations for emergency dose limitations.

- The proposed value of 10 rem TEDE as a lower bound for the control room design criteria is based on substantial scientific data that shows the observed radiation effects in individuals are not statistically different from zero, as noted in Health Physics Society Position Statement PS010-4, “Radiation Risk in Perspective.”

- Releases postulated from a bounding licensing basis LOCA based on a 10 CFR 50.46 compliant thermal-hydraulic analysis are many orders of magnitude less than what is postulated in the deterministic Maximum Hypothetical Accident (MHA) LOCA.

- The difference in the proposed MHA LOCA design criterion and the non-MHA criterion reflects the difference in probability of general transients leading to core damage and the probability of an MHA LOCA resulting in a substantial meltdown of the core and release of appreciable quantities of fission products into an intact containment.

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1 INTRODUCTION

RG 1.183 provides an acceptable method to evaluate the radiological consequences of applicable design basis accidents (DBAs) at light-water reactors, including an accident whose source term meets the description provided in footnote 1 of 10 CFR 50.67(b).[1] This is referred to in RG 1.183 Rev. 1 as the maximum hypothetical accident (MHA) loss-of-coolant accident (LOCA). The MHA LOCA is a conservative surrogate accident that is intended to challenge aspects of the facility design and results in substantial meltdown of the reactor core with subsequent release of appreciable quantities of fission products into an intact containment. When discussing radiological consequences, an MHA LOCA is typically assumed as the design basis case for evaluating the performance of release mitigation systems and the containment, and for evaluating the proposed siting of a facility. The analysis should calculate the limiting dose consequences to the public and control room doses, assuming a deterministic substantial core damage source term released into an intact containment. Both General Design Criteria (GDC) 19 and 10 CFR 50.67(b)(2)(iii) provide a specific radiological dose-based criterion of 5 rem total effective dose equivalent (TEDE) for demonstrating the acceptability of the control room design.

The intent of maintaining a habitable control room is to ensure that operators can respond in the unlikely event of an accident, to protect the public. Considering the importance of control room habitability, the selection of updated values for control room design criterion should be more aligned with the various U.S. and international organizations recommendations for emergency dose limitations up to 25 rem TEDE. As described in the increased enrichment (IE) regulatory basis document, ML23027A059, the control room design criteria are not intended to be operational limits and should not imply what is an acceptable exposure during emergency conditions. Design criteria are used to demonstrate that a plant design is adequate to maintain a habitable control room during highly unlikely, hypothetical events up to and including the MHA LOCA.

Compliance with the control room design criterion is shown through deterministic radiological consequence analyses with a set of inputs and assumptions that bound what might be seen in an actual event (occupancy factors, breathing rates, filter efficiencies, flow rates, atmospheric dispersion, no KI or respiratory protection, etc.). Considering the exceedingly low probability of occurrence of an accident and the assumptions made during these analyses, a graded, risk-informed method for compliance with control room design criteria up to 25 rem TEDE is appropriate and would provide the necessary flexibility for current and future technologies.

2 KEY TERMS USED IN THIS PAPER

Maximum Hypothetical Accident (MHA) Loss-of-Coolant Accident (LOCA): The MHA LOCA is a conservative surrogate accident that is intended to challenge aspects of the facility design and results in substantial meltdown of the reactor core with subsequent release of appreciable quantities of fission products into an intact containment.

[1] 10 CFR 50.67 (b) Requirements. (1) A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents[2] previously analyzed in the safety analysis report.

[2] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

Design Basis Accident (DBA): These are postulated accidents that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety.

3 APPLICABLE REGULATIONS, POLICY STATEMENTS, AND RESEARCH

The significant regulations relevant to the control room design criteria are General Design Criterion (GDC) 19, "Control room," in Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," and 10 CFR 50.67(b)(2)(iii). The general design criteria in 10 CFR Part 50, Appendix A, for the control room (GDC 19) were developed and issued to establish minimum necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) that provide reasonable assurance that the facility can be operated without undue risk to public health and safety.

Both GDC 19 and 10 CFR 50.67(b)(2)(iii) provide a specific radiological dose-based criterion of 5 rem (0.05 Sv) total effective dose equivalent (TEDE) for demonstrating the acceptability of the control room design. They represent a distinct layer of defense in depth that assumes a major accident resulting in substantial meltdown of the reactor core with subsequent release of appreciable quantities of fission products. In application, they are "performance based," which require that a licensee or applicant provide a control room habitability design using traditional deterministic radiological consequence analyses methods to judge the acceptability of the design.

> "[10 CFR 50.67] … The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from consideration of possible accidental events, that would result in potential hazard not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."
>
> and,
>
> "GDC 19: Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident [5 rem TEDE for holders of operating licenses using an AST under 50.67]. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures."

EPA-400/R-17/001, “PAG Manual,” January 2017, was also reviewed and the guidelines for emergency workers align with a frequency/probability-based approach. Specifically, Table 3-1 in the PAG Manual provides increasing radiological dose-based guidelines (5 rem – 10 rem – 25 rem) going from typical occupational exposures to protecting critical infrastructure to activities that are lifesaving or to protect large populations.

Policy Statements:

In developing this whitepaper, industry considered Commission expectations related to the initiation of the Increased Enrichment Rulemaking as provided in Staff Requirements Memorandum (SRM) SRM-SECY-21-0109, “Staff Requirements—SECY-21-0109—Rulemaking Plan on Use of Increased Enrichment of Conventional and Accident Tolerant Fuels Designs for Light-Water Reactors,” dated March 16, 2022, (ML22075A103) with a focus on applying a risk-informed approach when developing this rule and the associated regulatory basis and guidance.

Additionally, the probabilistic risk assessment (PRA) Policy Statement (60 FR 42622, August 16, 1995) was also reviewed as this formalized the Commission's commitment to risk-informed regulation. Specifically, the PRA Policy Statement states,

> “The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data, and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.”

Research:

Also considered were Staff perspectives provided in, “Control Room Design Criteria and Radiological Health Effects,” (ML23027A059) by the Office of Nuclear Regulatory Research (RES) in response to the informal assistance request (IAR) NRR-2022-019, “Assessment of Radiation Protection Recommendations for Emergency Workers,” (August 26, 2022) from the Office of Nuclear Reactor Regulation. In this IAR, RES staff determined that there is, “…. ample operating and licensing experience, scientific data, and technical information; numerous recommendations from national and international organizations responsible for radiation protection standards; probabilistic risk assessment technology; and regulatory precedence that support a reevaluation of the control room design criteria of GDC 19 and 10 CFR 50.67(b)(2)(iii).”

4 DEFENSE-IN-DEPTH AND MITIGATIVE STRATEGIES

The licensing process is based on the concept of defense in depth, in which plant design, operation, siting, and emergency planning comprise independent layers of nuclear safety. This approach uses design-basis accidents (DBAs) with highly stylized source terms to compute radiological consequences when assessing the effectiveness of each line of defense. As such, the DBAs establish and confirm the design basis of the nuclear facility, including its safety related structures, systems, and components (SSCs) and items important to safety, ensuring that the plant design meets the safety and radiological criteria in the regulations and associated guidance. This approach also ensures there are multiple lines of defense to maintain the effectiveness of physical barriers between radiation sources from workers, the public, and the environment both during normal operations and accident conditions.

GDC 19, the control room design criterion, is a figure of merit used in the evaluation of plant design with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of exposure to radiation. The deterministic design analyses demonstrating how the design criterion is met use inputs and assumptions that are generally the most restrictive values of plant parameters selected from the range of design values possible during the specific event so that the postulated consequences of the event are maximized. The inputs and assumptions for source term development do not include measures that consider operator actions to mitigate an event, minimize the releases, or limit exposures. Due to the conservative, deterministic nature of the DBA analyses, there is a margin of safety such that control room design may be adequate for many events beyond the design basis. As a result, increasing the control room radiological dose-based design criterion does not mean control room operators will receive more dose should an event occur. Occupational exposure limits delineated in 10 CFR 20, regulate the amount of dose operators receive and would remain unchanged. Furthermore, 10 CFR Part 20 applies to all exposure situations—normal and abnormal. However, for cases in which compliance would limit actions that may be necessary to protect health and safety, 10 CFR 20.1001(b) states that, "nothing in this part shall be construed as limiting actions that may be necessary to protect health and safety."

Requirements of 10 CFR 50.47 ensure nuclear power plants have emergency plans for responding to a radiological event. These plans ensure there are adequate resources on site to respond to an event, and ensure agreements are in place with local and state emergency facilities for aid in responding to an event. The facility's emergency plan also directs the use of personal protective equipment, as necessary and requires the performance of real time dose assessment based on actual plant conditions during the event to aid the determination of protective action recommendations (PARs).

Nuclear power plants comply with the requirements of 10 CFR 50.46, which precludes significant core melt and release into containment. Nuclear power plants have emergency operating procedures (EOPs) that direct operator actions in responding to events consistent with the NRC requirements following Three Mile Island. If a core melt begins, Plant Operators are directed to follow the Severe Accident Management Guidelines (SAMGs) per 10 CFR 50.155, which contain provisions for Severe Accident Water Addition (SAWA) and Severe Accident Water Management (SAWM). The SAMG based mitigative strategies have been demonstrated quantitatively in NUREG-1935 to arrest melt and release progression to levels orders of magnitude lower than that communicated in the deterministic source term.

After the Fukushima-Daiichi event, new layers of mitigative strategies were implemented that provide plants with additional layers (in addition to those credited in NUREG-1935) with which to minimize the consequences of an event. These advancements allow for staging of portable equipment (onsite and at regional locations) along with standardized connections to allow for emergency service organizations to provide emergency power and water addition. Finally, nuclear power plant personnel always practice ALARA (as low as reasonably achievable) principles with respect to radiological conditions.

Additionally, the existing regulatory framework provides another layer of defense in depth to ensure adequate protection of public health and safety. SSCs important to safety and the mitigation of radiological consequences are generally controlled in licensees' Technical Specifications and subject to NRC review and approval prior to modifications. Additional changes relative to the control room design criterion can be made by a licensee through the 10 CFR 50.59 process. Implementation of this process has licensees conduct screenings and evaluations to identify if the proposed change(s) could affect the basis for licensing the facility so that any changes that might pose a safety concern are submitted for NRC review and approval to confirm their safety before implementation. There are eight criteria that are

evaluated to determine if a proposed change, test, or experiment requires prior NRC approval through a license amendment. The threshold for requiring NRC approval provided by the criteria of 10 CFR 50.59 ensures adequate protection of public health and safety. The criterion pertinent to this discussion, 10 CFR 50.59(c)(2)(iii), states that a license amendment is needed if the change would "result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated)." NRC review and approval of a change prior to implementation is required if any one of the eight criteria of 10 CFR 50.59 are exceeded.

In summary, defense in depth ensures there are multiple lines of defense to maintain the effectiveness of physical barriers between radiation sources from workers, the public, and the environment both during normal operations and accident conditions. Therefore, a change to the control room radiological dose-based design criterion does not mean control room operators will receive more dose should an event occur as the occupational dose limits governed by 10 CFR 20 have not changed, nor is there a reduction in defense in depth or strategies available for licensees to mitigate an event to be less consequential than that of the deterministic source term resulting from licensees compliance with 10 CFR 50.46, 50.47, and 50.155.

5 APPROACH TO REVISED CONTROL ROOM DESIGN CRITERIA

This white paper proposes a range of acceptable control room design criteria values based on sound regulatory precedents and leveraging scientific recommendations that support the dose calculation methodology provided in Section 4.0 of RG 1.183. Applying a risk-informed approach provides the necessary flexibility for the implementation of current and future fuel technologies.

Industry's perspective is that a radiological dose-based design criteria range of 10 to 25 rem TEDE provides an acceptable level of control room habitability necessary to provide reasonable assurance that the control room would continue to be effectively staffed and operated to mitigate the effects of the accident and protect public health and safety. The value of 10 rem TEDE would represent a lower bound and is based on substantial scientific data that shows the observed radiation effects in individuals are not statistically different from zero, as noted in Health Physics Society Position Statement PS010-4, "Radiation Risk in Perspective." In addition, 10 CFR Part 20, allows workers to receive up to 10 rem (100 mSv) TEDE occupationally, under a special circumstance, within a calendar year. The value of 25 rem TEDE would represent the upper bound and is based on various U.S. and international organizations recommendations (e.g., EPA PAG Manual) for emergency dose limitations up to 25 rem TEDE and the recognition that GDC 19 and the radiological dose-based criteria specified in 10 CFR 50.67(b)(2) are design criteria, not operational limits.

Under this proposal, the upper value of 25 rem TEDE would be contained in GDC 19 in Appendix A to 10 CFR Part 50 and 10 CFR 50.67 and provide a criterion that is also consistent with the current offsite criteria of 25 rem TEDE in 10 CFR 50.67. The application of a 25 rem TEDE design criterion reflects the extremely low probability of the MHA LOCA surrogate event and the understanding that the design criterion does not represent actual or expected occupational exposure. In fact, releases postulated from a LOCA based on a 10 CFR 50.46 compliant thermal-hydraulic analysis are many orders of magnitude less than what is postulated in the deterministic MHA LOCA.

The 25 rem TEDE for the control room design criterion would apply only to accidents whose frequency would classify them as severe accidents. These events would include the MHA LOCA, which involves multiple failures of redundant trains of safety systems and results in potential hazards not exceeded by

those from any accident considered credible with substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

For events with higher sequence frequencies (e.g., PWR locked rotor accident), the lower value of 10 rem TEDE for the control room design criterion would apply to ensure the risk to the operators is not increased based on the need to retain and maintain SSCs critical to safety and mitigating radiological consequences. The difference in the proposed MHA LOCA design criterion and the non-MHA criterion reflects the difference in probability of general transients leading to any core damage (based on NUREG/CR-6928 and INL/EXT-21-63577) and the probability of an MHA LOCA resulting in a substantial meltdown of the core and release of appreciable quantities of fission products into an intact containment. These event-specific criteria would be controlled in RG 1.183 Rev. 2, as they currently are in Table 7 of RG 1.183 Rev. 1 for offsite criteria.

6 SUMMARY AND CONCLUSIONS

In summary, industry supports a risk-informed control room design criteria with boundaries of 10 to 25 rem TEDE. This approach enables deterministic evaluations using traditional radiological consequence analyses, meets control room habitability requirements, maintains defense in depth, and aligns with US and international emergency dose recommendations. Aligning the control room radiological dose-based design criterion of 10 CFR 50.67(b)(2)(iii) and GDC 19 with emergency dose limitations of 25 rem TEDE versus the occupational limit of 5 rem TEDE is reasonable while maintaining a habitable control room from which actions can be taken to operate the nuclear power unit safely under normal conditions and in a safe condition under accident conditions. The radiological source term required by 10 CFR 50.67 reflects substantial core damage with failure of the reactor coolant system to facilitate a release into an intact containment. Such a source term is precluded by compliance with other regulations (e.g., 10 CFR 50.46). Therefore, the likelihood of such a source term existing is exceedingly low as it would require the failure of several layers of defense in depth measures required by the regulations as discussed in Section 4. The recommendation of an upper bound radiological dose-based control room design criterion of 25 rem TEDE for inclusion into 10 CFR 50.67(b)(2)(iii) and GDC 19 also aligns with various U.S. and international organizations (e.g., EPA PAG Manual) recommendations for emergency dose limitations. A lower bound of 10 rem TEDE is based on substantial scientific data that shows the observed radiation effects in individuals are not statistically different from zero and ensures the risk to the operators is not increased based on the need to retain and maintain SSCs critical to safety.

7 REFERENCES

[1] Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”, Revision 1, October 2023.

[2] NUREG-1935, State-of-the-Art Reactor Consequence Analyses (SOARCA) Report, November 2012.

[3] NUREG/CR-6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants”, February 2007.

[4] INL/EXT-21-63577, “Initiating Event Rates at U.S. Nuclear Power Plants, 2020 Update”, Idaho National Laboratory, November 2021.