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General Comment

Please see attached document for comments.

Attachments

Comments on the NRC Proposed Rule_NRC-2019-0062

Comments on the NRC Proposed Rule: Preliminary Proposed Rule Language: Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (Docket ID: NRC-2019-0062)

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Introduction

We are a group of researchers from the Socio-Technical Risk Analysis (SoTeRiA) Research Laboratory in the Department of Nuclear, Plasma and Radiological Engineering at the University of Illinois at Urbana-Champaign. SoTeRiA is a multidisciplinary research laboratory that has a proven track record of developing scientific and innovative approaches for risk assessment, risk management, and risk-informed decision-making and regulation. SoTeRiA's research focuses on maintaining or improving a high level of safety in technological systems such as commercial nuclear power plants, while reducing the cost of operations. This scientific work helps create a pathway that would enhance the economic viability of the nuclear industry at a time when carbon-free energy resources play a critical role in mitigating climate change.

SoTeRiA advances the science of Probabilistic Risk Assessment (PRA) and its applications for complex technological systems. With the desire to create an international think-tank for safety, SoTeRiA has conducted research collaborations, for example, with the International Atomic Energy Agency (IAEA) to develop risk methodologies for advanced reactors and with the Japan Atomic Energy Agency (JAEA) to advance simulation models for risk-informed emergency preparedness and response. SoTeRiA's research studies have also received funding from several national sponsors, for example, from the Department of Energy (DOE) for advancing the PRA algorithm for the deployment of new technologies, DOE for Enterprise Risk Management (ERM), National Science Foundation (NSF) for big data analytics in PRA, and from the nuclear industry for the risk-informed resolution of Generic Safety Issue 191 (GSI-191) and for fire PRA. The results of these projects are reported in SoTeRiA publications (https://soteria.npre.illinois.edu/publications/).

This document provides 9 comments on the Preliminary Proposed Rule Language (Docket ID NRC-2019-0062)[1]. What follows is informed by our academic PRA expertise and by our industrial work experience in diverse nuclear power settings; US Navy nuclear operator (submarine service), fuel manufacture, national laboratory tests (LOFT program), and experience as Shift Technical Advisor, Unit Reactor Engineer, and Probabilistic Risk Analyst at a large commercial PWR power station.

[1] NRC contact information: Robert Beall, 301-415-3847, Robert.Beall@nrc.gov or William Reckley, 301-415-7490, William.Reckley@nrc.gov. Federal Rulemaking Website: https://www.regulations.gov Docket ID search term NRC-2019-0062

Comment 1: Safety functions (§53.210)

To be truly transformative and useful for implementation in a commercial setting, the Nuclear Energy Innovation and Modernization Act (NEIMA) regulation should focus on the following critical safety elements[2]:

1. Maximum power during production (licensed power level),
2. Prompt criticality,
3. Reactor shutdown, and
4. Fission product release

These four criteria can be accomplished by focusing resources on maintaining a high level of protection efficacy against fission product release, assuming the core inventory is present in containment at an elevated thermodynamic state commensurate with the energy contained by the process.

A summary statement for the design of safety function SSC requirements that applies to both "normal process" releases as well as "unexpected releases" should be included: "SSCs preventing radioactive release higher than what is allowed during normal operation must achieve a high level of protection efficacy."

The safety function design should be for protection against maximum radiation release. The proper stance would be to assume that this hazard is present and requires containment regardless of whether the process is intact or has failed. The reviewers identify bases for safety functions required for containment of maximum hazard potential as follows:

1. Prompt criticality has the potential to produce an extremely high energy release, an unbounded hazard; SSCs preventing prompt criticality should be considered as safety function SSCs.
2. The SSCs that keep the reactor at, or below, the licensed power level ensures the hazard potential is bounded by the process stored energy in addition to the energy released from radioactive decay; SSCs that prevent power from exceeding the licensed power level should be considered as safety function SSCs.
3. The SSCs that reach and maintain reactor shutdown ensure that the thermodynamic state can be known over time; SSCs that maintain reactor shutdown should be considered as safety function SSCs.[3]

Safety function should apply to SSCs that prevent radioactive leakage from exceeding the allowed limits. All other equipment should be treated as commercial grade. This concept in risk-informed and transformative regulation would assume that the hazard is present at all times; it cannot be avoided, but it can be managed.

The concern: Sharply focusing on the scope of regulatory prescription and oversight will contribute toward a continued reduction of nuclear energy production in the commercial marketplace. Recent experience with efforts to meet costs and schedules for new nuclear reactors has demonstrated the uncertainty felt by investors. From the scientific perspective, the risk-informed decision-making asked for in NEIMA would result in a greater production of energy by nuclear power, while a proper risk balance considering climate effects, pollution, and land use in energy production should be considered by regulators.

[2] https://www.congress.gov/bill/115th-congress/senate-bill/512. Accessed 7 December 2020.
[3] The amount of energy present would be the stored energy in the process plus the energy released from fission product decay less any energy removed over time.

Comment 2: "Reasonable assurance"

The proposed rule language in 10 CFR Part 53, "Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors", includes the wording *reasonable assurance of adequate protection:*

> "Each advanced nuclear plant must be designed, constructed, operated, and decommissioned such that there is reasonable assurance of adequate protection of the public health and safety and the common defense and security." (Preliminary Proposed Rule Language in §53.20, Subpart B - Technology-Inclusive Safety Requirements)[4]

This comment, in particular the use of the qualifier "reasonable assurance", was summarized during the public comment period in the NRC Public Meeting on November 18.[5] In response, the NRC staff stated that the term has been used in other regulation. In the reviewers' opinion, use of the descriptor is not an issue when used in connection with an engineering expectation, such as performance of a diesel generator or to an engineered limit; examples could be "The diesel generators must start and load with reasonable assurance" or "Worker dose limit shall not exceed 5 REM in a year with reasonable assurance." In the reviewers' opinion, such a statement does not refer to whether or not the required adequate protection standard has been met; it simply sets the expectation for the level of reliability provided by an SSC or a system of SSCs. The regulation should state what is required of the licensee, that is, those requirements which the Commission deems necessary for adequate protection. Therefore, it seems the regulation does not need to include "adequate protection" either, since the licensee would meet that standard by compliance with the regulation as determined by the Commission rather than by the licensee.

Addition of the qualifier "reasonable assurance" to "adequate protection" in the Preliminary Proposed Rule Language in Subpart B, Technology-Inclusive Safety Requirements, appears to imply that adequate protection may not necessarily need to be met in a plant's licensed condition; this change may lead to loss of public trust in the Commission's ability to ensure adequate protection.

"Reasonable assurance" does appear in the Atomic Energy Act of 1954 (AEA) with "adequate protection." For example, in Sections 189 "Hearings and Judicial Reviews," "... it shall allow operation during an interim period under the combined license." and Section 192 "Temporary Operating License," "...there is reasonable assurance that operation of the facility during the period of the temporary operating license in accordance with its terms and conditions will provide adequate protection to the public health and safety and the environment ..." These uses of the term "reasonable assurance" by Congress appear to be intended for interim or temporary relief from full assurance of adequate protection as required in Section 182, "License Applications" where no qualification is added. These uses seem consistent with the understanding stated above that the addition of the term infers uncertainty as to whether or not adequate protection is fully met. It might be useful to think in terms of the language of the AEA. If the AEA asked for technical specifications that give reasonable assurance of safety, then "reasonable assurance" would have been the standard; and, of course, that is not the case as the AEA asks for "adequate protection". Adding reasonable assurance to adequate protection, in the reviewers' opinion, creates a new and confusing standard.

The concern: This reviewer's concern is that adding the qualifier "reasonable assurance" to adequate protection leaves open the possibility that adequate protection may not necessarily be met in a plant's licensed condition.

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[4] https://www.regulations.gov/comment?D=NRC-2019-0062-0012, also https://www.regulations.gov/docket?D=NRC-2019-0062
[5] https://www.nrc.gov/pmns/mtg?do=details&Code=20201300. Accessed 7 December 2020.

Comment 3: Epistemic and aleatoric uncertainty

We recommend the language to be changed as indicated below.

> §53.25 Defense in Depth.
>
> "Measures must be taken for each advanced nuclear plant to ensure appropriate defense in depth is provided to compensate for ~~epistemic and aleatory~~ uncertainties such that there is high confidence that the safety criteria in this subpart B are met over the life of the plant. The ~~epistemic and aleatory~~ uncertainties to be considered include those related to the ability of barriers to limit the release of radioactive materials from the facility during routine operation and for licensing basis events and those related to the reliability and performance of plant SSCs and personnel, and programmatic controls. Measures to compensate for ~~these~~ uncertainties can include increased safety margins in the design of SSCs and providing alternate means to accomplish safety functions. No single design or operational feature, no matter how robust, should be exclusively relied upon to meet the safety criteria of 10 CFR part 53."

The concern: We do understand the scientific value of a comprehensive analysis of aleatory and epistemic uncertainties in risk estimations. We also value separating these two types of uncertainties in regulatory guidance (e.g., RG 1.174); however, inclusion of words such as "epistemic", "epistemology", "aleatory", or "aleatoric" in regulation may generate confusion for technicians and practicing engineers.

Comment 4: Analysis Requirements (§53.450)

In sub-point 53.450(b) (5), it is stated that PRA shall "Consider events that challenge plant control and safety systems whose failure could lead to the uncontrolled release of radioactive material to the environment. These include **internal events**, such as human error and equipment failure, and **external events**, such as earthquakes, identified in accordance with Subpart D of this part."

The concern: The reviewers believe that security analysis requires a PRA-based foundation in order to comprehensively consider internal and external events since security issues can also induce internal or external initiating events. Developing a PRA-based foundation for security analysis, leading to a unified platform for security and safety risk analysis, will add realism to the risk quantification of nuclear power plants.

Comment 5: Analysis Requirements (§53.450)

In sub-point 53.450(b) (7), it states that PRA shall "be maintained and upgraded to cover initiating events and modes of operation contained in generally accepted methods, standards, and practices in effect one year prior to each required PRA upgrade. The PRA must be upgraded **every two years** until the permanent cessation of operations under Subpart G of this part."

The concern: The reviewer suggests that the upgrade frequency of the PRA be changed from "every two years" to "every two years or whenever new information emerges that contradicts previously used information (e.g., Updated failure rates of components or updated failure modes) or if there is a design change."

Comment 6: Application of Analytical Safety Margins to Operational Flexibilities (§53.470)

In sub-section 53.470, it states that "Where an applicant or licensee so chooses, design criteria more restrictive than those defined in § 53.230(b) may be adopted to support operational flexibilities (e.g., emergency planning requirements under Subpart F of this part)."

The concern: This may go against the defense in depth principle. For example, some designs may focus on increasing the margin for a single barrier and reducing the margin for another barrier. The reviewers believe that the defense in depth principle assumes that any single barrier shall be designed to limit the release of radioactive material even if the other barriers fail. Increasing the margin of one barrier should not make any difference when dealing with another barrier.

Comment 7: Facility Safety Program Performance Criteria (§53.810)

In sub-point 53.810(a), it states that "Performance objectives for design features and programmatic controls must be established such that the risks to public health and safety from an advanced nuclear plant due to normal operation or licensing basis events must not be **a significant addition** to other societal risks."

The concern: In the reviewers' point of view, the wording "**a significant addition**" is ambiguous and should be expressed in a quantitative way based on multiple dimensions (e.g., economical loss, dose, health impacts). Also, types of "other societal risks" need to be specified.

Comment 8: Facility safety program plan (§53.820)

In sub-point 53.820(b)(2), there is a typo. It states that "(2) The methods used to analyze the technologies identified under paragraph **(f)(1)(i)** of this section against the criteria provided in § 53.810."

The concern: There is no sub part (i) under subpart (f)(1). The reviewers believe this might be a typo.

Comment 9: Facility Safety Program

In Subpart B and C, § 53.80 should read § 53.800, where it refers to the Facility Safety Program (FSP) stated in Subpart F, Requirements for Operation.

In Subpart B, § 53.24 and § 53.25 should read § 53.240, § 53.250 respectively.