Alternative Physical Security Requirements for Advanced Reactors Proposed Rule: DIFFERING VIEW

An Office of Nuclear Security and Incident Response (NSIR) staff member provided a differing view with four problem statements on the proposed rule. The staff member's differing view arises from the proposed provisions within Title 10 of the *Code of Federal Regulations* (10 CFR) 73.55(s)(1)(ii), "Eligibility," where "[t]he applicant or licensee must demonstrate that the consequences of a postulated radiological release that results from a postulated security-initiated event do not exceed the offsite dose reference values defined in §§ 50.34 and 52.79 of this chapter," and 10 CFR 73.55(s)(1)(iv), "Analysis," where "[t]he applicant or licensee electing to meet one or more of the alternative security requirements in paragraph (s)(2) of this section must perform a technical analysis demonstrating how it meets the criteria in paragraph (s)(1)(ii) of this section." The differing view is characterized in the following problem statements:

- The proposed rule imposes unnecessary regulatory burden, which would be an avoidable impediment to a licensee or applicant that wants to apply alternative physical security requirement(s) in the design of a physical protection program to meet the requirements of 10 CFR 73.55.

- The proposed rule and its implementation set forth a radiation dose of 25 rem total effective dose equivalent (25 rem TEDE) (in any 2-hour period following the onset of the postulated fission product release) as an acceptable dose limit for members of the public and a consequence-based approach that uses this 25 rem TEDE as the acceptable criterion for determining offsite release that would not endanger public health and safety.

- The proposed rule and implementation of 10 CFR 73.55(s)(1)(ii) and (s)(1)(iv) allows for relying on human actions in lieu of plant design features, structures, systems, and components (SSCs) and barriers that would not meet the Commission's expectations in 2008 Policy Statement on Regulation of Advanced Reactors to reduce reliance on human actions.

- The proposed rule, a more specific requirement in 10 CFR Part 73.55, provides a regulatory pathway for circumventing regulatory requirements established in the current framework for safety and security.

The details of the bases for the differing view indicated above, along with the potential impact on mission and alternatives for resolutions are provided in this enclosure as Problem Statements No.1, No.2, No.3.a, and No.3.b.

<u>PROBLEM STATEMENT NO.1</u>

The U.S. Nuclear Regulatory Commission's (NRC's) issuance of the proposed rule could impose unnecessary regulatory burden, which would be an avoidable impediment to a licensee or applicant that wants to apply alternative physical security requirements in the design of a physical protection system to meet the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage."

Regulatory and Technical Basis

- ***The current regulatory framework in 10 CFR 73.55 does not require a licensee or applicant to perform a consequence analysis (e.g., to analyze the consequences of security-initiated events (those initiated by the design-basis threat (DBT))).***

In the proposed rule, 10 CFR 73.55(s)(1) states the following:

> (1) General requirements.
> (i) *Applicability*. The requirements of this section apply to an applicant for or holder of a license for a small modular reactor, as defined in § 171.5 of this chapter, or a non-light-water reactor under part 50 of this chapter or part 52 of this chapter.
>
> (ii) *Eligibility*. The applicant or licensee must demonstrate that the consequences of a postulated radiological release that results from a postulated security-initiated event do not exceed the offsite dose reference values defined in §§ 50.34 and 52.79 of this chapter.
>
> (iii) *Identification and documentation*. The applicant or licensee must identify the specific alternative physical security requirement(s) it intends to implement as part of its physical protection program and demonstrate how the requirements set forth in this section are met when the selected alternative(s) is used.
>
> (iv) *Analysis*. The applicant or licensee electing to meet one or more of the alternative security requirements in paragraph (s)(2) of this section must perform a technical analysis demonstrating how it meets the criteria in paragraph (s)(1)(ii) of this section. The licensee must maintain the analysis until the certifications required by § 50.82(a)(1) of this chapter or § 52.110(a) of this chapter have been docketed by the NRC.

The proposed provisions in paragraphs (ii) and (iv) above establish that a licensee (or applicant for an operating or combined license) must perform a site-specific analysis to evaluate potential offsite radiological consequences of security-initiated events. The proposed rule further states, as illustrated by the table below, that before implementing any of the alternatives in 10 CFR 73.55(s)(2), a licensee or applicant must satisfy the requirements of 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv). This includes providing an analysis of potential offsite radiological consequences from postulated security-initiated (DBT-initiated) events, to show that such an event would result in an offsite release below the dose value of 25 rem TEDE (i.e., the radiation dose in any 2-hour period following the onset of the postulated fission product release would not exceed 25 rem TEDE).

Comparison of Proposed Rule and Existing Rule	
Proposed 10 CFR 73.55(s)	**Existing 10 CFR 73.55(r), Alternative measures**
(1) General requirements. (i) *Applicability*. The requirements of this section apply to an applicant for or holder of a license for a small modular reactor, as defined in § 171.5 of this chapter, or a non-light-water reactor under part 50 of this chapter or part 52 of this chapter.	(1) The Commission may authorize an applicant or licensee to provide a measure for protection against radiological sabotage other than one required by this section if the applicant or licensee demonstrates that:
(ii) *Eligibility*. The applicant or licensee must demonstrate that the consequences of a postulated radiological release that results from a postulated security-initiated event do not exceed the offsite dose reference values defined in §§ 50.34 and 52.79 of this chapter.	Not required.
(iii) *Identification and documentation*. The applicant or licensee must identify the specific alternative physical security requirement(s) it intends to implement as part of its physical protection program and demonstrate how the requirements set forth in this section are met when the selected alternative(s) is used.	(1)(i) The measure meets the same performance objectives and requirements specified in paragraph (b) of this section…. (2) The licensee shall submit proposed alternative measure(s) to the Commission for review and approval in accordance with §§ 50.4 and 50.90 of this chapter before implementation. (3) In addition to fully describing the desired changes, the licensee shall submit a technical basis for each proposed alternative measure. The basis must include an analysis or assessment that demonstrates how the proposed alternative measure provides a level of protection that is at least equal to that which would otherwise be provided by the specific requirement of this section.
(iv) *Analysis*. The applicant or licensee electing to meet one or more of the alternative security requirements in paragraph (s)(2) of this section must perform a technical analysis demonstrating how it meets the criteria in	Not required.

paragraph (s)(1)(ii) of this section. The licensee must maintain the analysis until the certifications required by § 50.82(a)(1) of this chapter or § 52.110(a) of this chapter have been docketed by the NRC.	

Contrary to the proposed rule, the Commission's current requirement in 10 CFR 73.55(r), which includes the necessary exemptions to prescribed requirements for implementing alternatives under 10 CFR 73.5, "Specific exemptions," does not require a licensee or applicant to perform an analysis to demonstrate that the consequences of a postulated radiological release resulting from a postulated security-initiated event do not exceed the offsite dose reference values defined in 10 CFR 50.34, "Contents of applications; technical information," and 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report." Such an analysis to justify a request to implement an alternative is not required either under 10 CFR 73.55(r) or for the necessary exemption from prescribed security requirements in 10 CFR 73.55. Under the regulatory framework in 10 CFR 73.55, the technical basis must demonstrate that the alternative measure provides a level of protection that is equal to that of the corresponding specific requirement in 10 CFR 73.55 (i.e., the alternative meets the performance objective and requirements in 10 CFR 73.55(b)). This justifies the implementation of the proposed alternative. The same technical basis justifies specific exemptions from prescribed security requirements that may be necessary to implement the alternative. This technical basis is the same as that required in the proposed rule, as the acceptability of the alternative is based on how the applicant or licensee would design and implement the alternative physical security requirements to meet the requirements of 10 CFR 73.55. This is evident from the proposed rule in 10 CFR 73.55(s)(1)(iii), which requires the licensee or applicant to demonstrate how it will meet the requirements in 10 CFR 73.55(b)(3) when using the selected alternatives.

Despite the similarities between the current and proposed rule, the proposed rule imposes the unnecessary burden that a licensee or applicant must perform consequence analyses. A consequence analysis for security-initiated events, based on the DBT of radiological sabotage defined in 10 CFR 73.1 (i.e., intentional acts that target SSCs and barriers), is given as an acceptable way for an applicant or licensee to meet the condition for eligibility to implement alternative physical security requirements in the design of its physical protection program. This is an unnecessary impediment for advanced reactor licensees or applicants because they could request implementation of the same alternative physical security requirements through 10 CFR 73.55(r) without performing any consequence analyses. Therefore, the differing view problem statement is that the proposed rule imposes an unnecessary burden on advanced reactor designers, licensees, and applicants that are considering and applying alternatives in their physical protection program designs. By issuing the proposed rule, the NRC could impede the efficient industrywide adoption of alternative means and methods, including innovative approaches, in the designs of physical protection programs for advanced reactors.

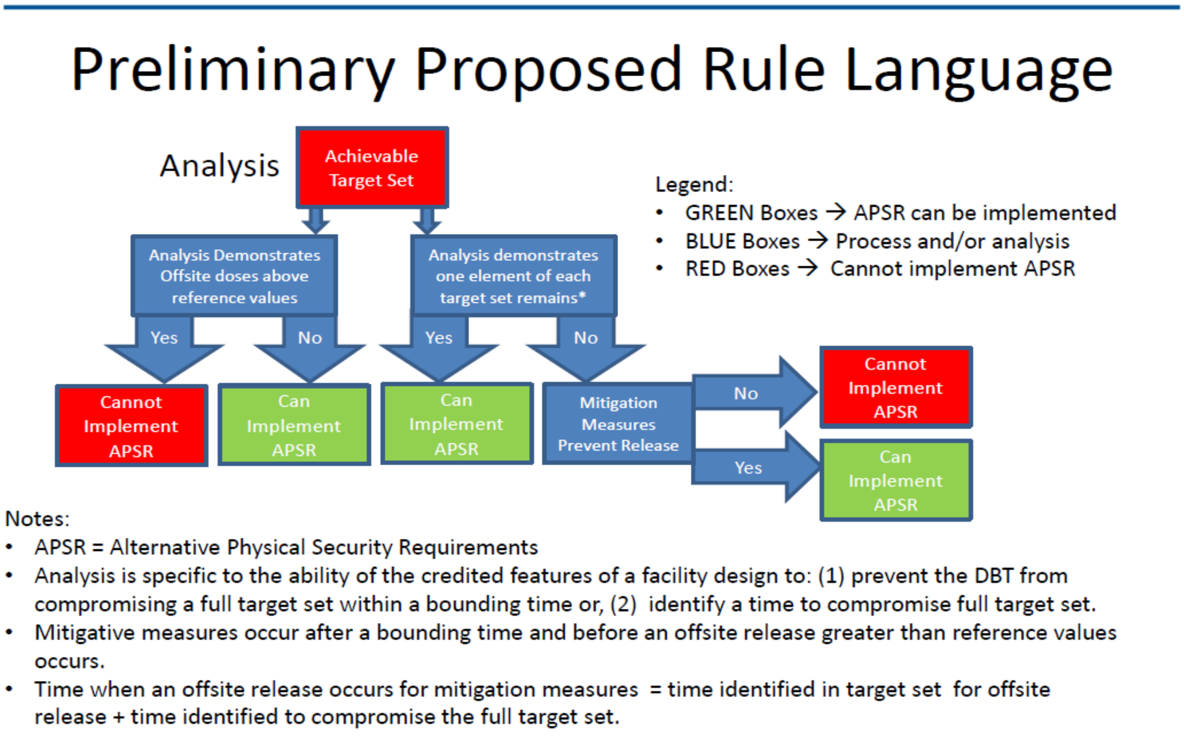
- ***The safety of reactors (including operating light-water reactors, light-water and nonlight-water small modular reactors, and advanced reactors) is ensured by the comprehensive safety requirements and safety-related SSCs that are documented in the final safety analysis report.***

The current safety regulatory framework, in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR Part 52, "Licenses, Certifications, and Approvals for

Nuclear Power Plants," establishes necessary and sufficient safety requirements through safety and hazards analyses and assessments of the site and the facility, which identify design features (e.g., SSCs and barriers) to be incorporated to protect a reactor, ensuring extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. Specifically, the provisions in 10 CFR 50.34(a)(1)(ii) and 10 CFR 50.34(a)(1)(ii)(D) for a 10 CFR Part 50 operating license, and similarly the provisions in 10 CFR 52.79(a)(1)(vi) and 10 CFR 52.79(a)(2)(iv) for a 10 CFR Part 52 combined license, along with hazards analyses and analyses of design-basis accidents, ensure a comprehensive identification of safety-related SSCs and barriers, as well as risk-significant nonsafety-related SSCs, that must function as designed to ensure safe reactor operation. This identification establishes the basis for licensing.

A licensee's final safety analysis report documents the safety basis established by meeting the regulatory requirements in the current safety regulatory framework. The analyses, assessments, and evaluations performed under the current safety regulatory framework do not include consequence analysis for intentional acts, either internal or external, based on the characteristics, attributes, and capabilities of the DBT of radiological sabotage specified in 10 CFR 73.1, "Purpose and scope." The currently required design-basis accidents analyses and safety assessments, including aircraft impact assessments, do not consider failure of the design features, SSCs, and barriers due to security threats up to and including the DBT of radiological sabotage. The current regulations do not require a licensee or applicant to perform additional beyond-design-basis analyses, assessments, and evaluations of DBT-initiated accident scenarios; to determine progressions of accidents not previously analyzed; or to assess the potential offsite radiological consequences. (Such analysis would include, for example, identifying DBT-initiated events; assessing DBT-caused fuel, systems, and facility damage ratios; evaluating DBT-caused release fractions; and analyzing the potential offsite consequences of DBT-caused accident sequences and DBT-caused dispersion of radiological source term.) Instead, the safety basis for licensing, as analyzed and documented for a safety envelope of operations that the Commission finds acceptable, relies on the licensee's meeting the security requirements in 10 CFR 73.55. When adequately designed and implemented, a physical protection program that satisfies the requirements in 10 CFR 73.55 is deemed to provide adequate protection against the DBT of radiological sabotage. This protection forms the technical and regulatory bases for the Commission's finding of assurance that the licensed activities do not constitute an unreasonable risk to the public health and safety.

Unlike the current safety and security regulatory framework, the implementation of the proposed rule requires that an analysis be performed to evaluate potential offsite consequences based on a consequence threshold of 25 rem TEDE, including additional analysis of DBT-initiated scenarios. This requirement, illustrated below, was presented in public meetings on October 19, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21291A238), and January 20, 2022 (ADAMS Accession No. ML22019A075).



Under the current regulations and regulatory framework, the licensee or applicant uses 25 rem TEDE as the reference value in its analyses, assessments, and evaluations to identify the necessary design features, SSCs, and barriers. For example, compliance with 10 CFR 50.34(a)(1)(ii) and 10 CFR 50.34(a)(1)(ii)(D) or with 10 CFR 52.79(a)(1)(vi) and 10 CFR 52.79(a)(2) means that, crediting the safety functions of plant features, SSCs, and barriers identified, the postulated fission product release (using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate accident consequences) would not lead to a radiation dose above 25 rem TEDE. The current requirements of 10 CFR 73.55, when met and acceptably implemented, provide adequate protection to ensure maintenance of the safety basis as analyzed and documented in the final safety analysis report. The current regulations and regulatory framework for the safety/security interface make it unnecessary to require an analysis within the framework of 10 CFR 73.55; doing so would impose an arbitrary applicability requirement (or eligibility condition) on licensees wishing to apply the alternatives in 10 CFR 73.55(s)(2) in their designs. The differing view problem statement asks why the consequence analysis requirement in the proposed rule is necessary.

Under the current regulations and regulatory framework, an additional analysis of potential offsite consequences, implemented as illustrated above, to include analysis of DBT-initiated events, would intentionally not meet the requirement of 10 CFR 73.55 (i.e., would result in inadequate protection) and would be otherwise unnecessary, without any regulatory or technical merit.

The following explains this in the context of licensing, to show the unnecessary burden resulting in part from the proposed requirement, which goes beyond the current security regulation and regulatory framework. The logic is that, if the two points described below are true, then the proposed rule would be creating an unnecessary requirement. In other words, if a licensee's or applicant's design has already met the requirements, through the analyses, assessments, and evaluations completed for either an NRC-certified design or an operating license or combined license, then the design already identifies plant design features, SSCs, and barriers based on the 25 rem TEDE reference. If the licensee or applicant has also demonstrated that it has a physical protection program that satisfies the performance objective and requirements in 10 CFR 73.55 using alternative physical security requirements, then, by the finding of the Commission, it has provided adequate security to maintain the safety design features.

The question is why the consequence analysis requirement in the proposed rule is necessary if the following are true:

- On the safety side, the NRC uses 25 rem TEDE as a reference value during the design certification process to ensure that a reactor design has the necessary design features, SSCs, and barriers to adequately protect against release of fission product that would endanger the public, design-basis internal random events, and external events. That is, the design features, SSCs, and barriers will be sufficiently available and reliable, through redundancy, diversity, and independence, to perform their intended safety functions. These design features, SSCs, and barriers are the reason for the low likelihood that postulated accidents as analyzed will cause unacceptable offsite consequences.

- On the security side, meeting the requirements of 10 CFR 73.55 provides reasonable assurance that a licensee can adequately defend against the DBT adversary (i.e., intentional, nonrandom internal and external hazards). The physical protection program minimizes the likelihood that intentional acts (i.e., DBT-initiated events) will be able to compromise the design features, SSCs, and barriers and cause unacceptable offsite consequences. The physical protection program also protects against accidents and consequences beyond those analyzed on the safety side (i.e., it eliminates the need to analyze consequences of intentional acts based on the DBT characteristics, attributes, and capabilities described in 10 CFR 73.1).

Therefore, the differing view problem statement is that the proposed rule would impose an unnecessary regulatory requirement for applying alternative measures and an increased burden in demonstrating compliance with such a requirement in the security basis for licensing. (At a minimum, it would increase the licensee's or applicant's burden in preparing and submitting analyses, as well as the NRC staff's burden in reviewing how the analyses evaluate the potential offsite radiological consequences, in accordance with 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv), within the security plans.) By issuing the rule as proposed, the NRC will impede the use of alternatives in the physical protection designs of advanced reactor designers and applicants.

- ***The security regulations are structured to ensure adequate protection for the minimum sets of safety-related SSCs, so that those SSCs will be available to perform the safety functions designed to protect public health and safety by preventing radiological sabotage by the DBT adversary.***

The security regulatory framework of 10 CFR Part 73, "Physical Protection of Plants and Materials," establishes graded standards of physical protection commensurate with the risks of activities involving special nuclear material (i.e., in terms of material attractiveness and radiological consequence). For power reactors, regardless of the reactor design, 10 CFR 73.55(b)(1) states the following:

> The licensee shall establish and maintain a physical protection program, to include a security organization, which will have as its objective to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

When the performance and prescriptive requirements in 10 CFR 73.55 are met and implemented, the Commission has found, and will find, that a licensee has provided, or an applicant for an operating license has demonstrated, adequate protection against the DBT for radiological sabotage. That is, the licensee or applicant has protected against the potential for the DBT to cause the failure of safety-related design features, SSCs, or barriers, and has thus maintained the safety and licensing basis as analyzed, providing reasonable assurance that activities licensed do not constitute unreasonable risks to public health and safety or to the protection of the environment.

Contrary to the Commission's current regulatory framework and regulations, the proposed rule would require a licensee or applicant to perform a site-specific analysis to evaluate potential offsite radiological consequences, despite a finding of adequate protection. The proposed rule would require unnecessary analysis and would establish a new regulatory position that is contrary to the longstanding regulatory and technical basis for the Commission finding of adequate protection.

Furthermore, the analysis required by 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv), as shown in the figure above, will allow a licensee or applicant to undermine the current safety and security regulatory framework. Specifically, a licensee or applicant could evaluate the potential offsite consequences of a loss of identified design features, SSCs, or barriers (e.g., those identified through safety analysis, assessments, and evaluations using reference values, the DBA, etc.) due to a DBT-initiated event, based on availability of mitigation equipment and ability to move freely to perform mitigation to prevent release up to an offsite dose of 25 rem TEDE. By establishing a provision in 10 CFR 73.55 that allows for reliance on mitigation measures (human actions) to prevent release, the NRC would enable licensees and applicants to intentionally erode current safety and security standards. Problem Statements No. 2 and No. 3 of this enclosure address this point in more detail.

The proposed guidance for implementation specifies that an acceptable implementation of the required consequence analysis is one based on DBT-initiated events with an acceptable offsite release of up to 25 rem TEDE to members of the public. The proposed rule, as implemented, will impose an unnecessary burden and create a regulatory impediment for licensees and applicants considering or applying alternative measures in their physical protection program designs.

- ***Applicants or licensees must perform a site-specific analysis to evaluate potential offsite radiological consequences.***

The first part of the proposed requirement, 10 CFR 73.55(s)(1)(ii), states, “The applicant or licensee must demonstrate that the consequences of a postulated radiological release that results from a postulated security-initiated event do not exceed the offsite dose reference values defined in §§ 50.34 and 52.79 of this chapter.” The second part, 10 CFR 73.55(s)(1)(iv), states, “The applicant or licensee electing to meet one or more of the alternative security requirements in paragraph (s)(2) of this section must provide a technical analysis demonstrating how it meets the criteria in paragraph (s)(1)(ii) of this section.” The applicant or licensee must also show that its physical protection program design, with the alternative(s), meets the design requirement of preventing a significant radiological release.

For an acceptable implementation of 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv), a licensee or applicant wishing to demonstrate eligibility to use some or all of the alternative security measures in 10 CFR 73.55(s)(2) should develop scenarios testing its ability to uphold the site’s physical security plan (e.g., to protect target set equipment or prevent an offsite release from exceeding reference doses) while employing the alternative measures. Possible scenarios to evaluate include, but are not limited to, the following:

(1) A DBT-initiated event that compromises some or all target sets and does not involve human actions to mitigate a potential radiological release. Such an event should not result in offsite doses above the reference values in 10 CFR 50.34(a)(1)(ii)(D)(*1*)–(*2*) and 10 CFR 52.79(a)(1)(vi)(A)–(B).

(2) A DBT-initiated event that compromises some or all target sets and results in core damage or causes a release of radionuclides from any source before offsite doses exceed the reference values in 10 CFR 50.34(a)(1)(ii)(D)(*1*)–(*2*) and 10 CFR 52.79(a)(1)(vi)(A)–(B). The response to such an event may involve both onsite and offsite resources to interdict the adversary force and mitigate the release.

The consequence analysis required by the proposed rule, as described for implementation above, uses a threshold of 25 rem TEDE in a 2-hour period as an acceptable dose limit for members of the public.

Unlike the proposed rule and implementation, the current regulatory framework requires that a licensee or applicant identify all safety-related SSCs, including barriers for safety of reactor operations protecting against risk of core damage and risk of release of radiological nuclides (i.e., 10 CFR 50.34 or 10 CFR 52.79 analysis, assessment, and evaluation). The proposed rule modifies the design performance objective from “prevent significant core damage” to “prevent significant release,” to ensure the protection of those SSCs and barriers whose failure would lead to offsite release endangering public health and safety.

According to the proposed implementation guidance, a licensee or applicant would consider intentional acts of radiological sabotage based on the characteristics, attributes, and capabilities of the DBT adversary. Whether the radiological consequences of DBT-initiated scenarios would be considered a danger to public health and safety would depend on whether the resulting radiation exposure was above 25 rem TEDE, the threshold defined for significant release. A dose of up to 25 rem TEDE would not be considered a significant release, but a dose greater than 25 rem TEDE would be considered a significant release and therefore a danger to public health and safety.

The differing view is that the technical analysis required by the proposed rule in 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv) is an unnecessary burden. This is because the current safety and security regulatory framework, by requiring safety and hazards analyses and assessments such as those of 10 CFR 50.34 and 10 CFR 52.79, establishes what safety-related SSCs and barriers must remain reliable and available to perform their intended safety functions (i.e., to prevent core damage or prevent release of radiation hazards to the environment). In the discussion below, this is referred to as Step A.

The design of the physical protection program in accordance with 10 CFR 73.55, referred to as Step B, enables plant features, SSCs, and barriers to perform their required safety functions by protecting them from threats up to and including the DBT of radiological sabotage. A licensee or applicant wishing to use alternative physical security requirements must demonstrate that the resulting physical protection program will meet all the performance and prescriptive requirements of 10 CFR 73.55. This ensures that the identified SSCs and barriers will perform the required safety functions and are adequately protected against intentional acts based on the DBT of radiological sabotage. Under the current regulatory framework for security, this notion of adequate protection constitutes a necessary and sufficient standard, and a necessary and sufficient regulatory footprint, for the Commission to make its finding.

In the current safety and security regulatory framework, the Commission does not require a licensee or applicant to evaluate potential offsite radiological consequences when either considering or applying an alternative measure. Nor does it expand its regulatory footprint to impose additional analysis of potential offsite radiological consequences (referred to as Step C), after the licensee has satisfied the requirements that the Commission has deemed necessary and sufficient for adequate protection.

To reiterate, the key technical and regulatory concern is that the proposed rule and implementation would require licensees and applicants to perform Step C despite having completed Steps A and B. Under the current regulatory framework, Step C is not required; instead, the SSCs and barriers determined to be safety-related are considered adequately protected if the requirements of 10 CFR 73.55 are met (e.g., the protective strategy will interdict and neutralize the DBT of radiological sabotage; the design of the physical protection program prevents significant releases that would endanger public health and safety or the environment). (For licensees and applicants applying 10 CFR 50.69, “Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors,” the SSCs and barriers in question may include RISC-2 SSCs (which are nonsafety-related but perform safety-significant functions), along with, from a security perspective, any equipment or systems whose failure would lead to common-cause failure of RISC-1 and RISC-2 SSCs.)

The requirement of Step C in the proposed rule and implementation is the additional burden that is otherwise not required based on adequate protection. Step C is the analysis of offsite consequences of intentional acts based on the DBT; it requires licensees and applicants to identify and evaluate accident scenarios not previously considered, based on the intentional failure of plant features, SSCs, and barriers that safety analysis has already shown to be reliable and available. This consequence analysis is not well-defined, and the NRC staff has proposed only high-level guidance that does not sufficiently explain how to perform the analysis. This means that the analysis will be complex and costly to complete, and costly for the staff to review, without adding any information needed for the Commission to make its findings of adequate safety and security.

The following example illustrates the burden imposed by Step C in the context of 10 CFR 50.150, “Aircraft impact assessment.” (Another example would be the design for mitigation of loss of offsite power for all reactor plants to protect against the risk of station blackout.)

In 10 CFR 50.150, the Commission has established the following regulatory basis for adequate protection from the potential impact of a large commercial aircraft. The requirements relevant to this discussion are the following:

- The regulation at 10 CFR 50.150(a) states that each applicant must perform a design-specific assessment of the effects on the facility of the impact of a large commercial aircraft. Using realistic analyses, the applicant must identify and incorporate design features and functional capabilities to show that, with reduced use of operator actions, (i) the reactor core remains cooled, or the containment remains intact, and (ii) spent fuel cooling or spent fuel pool integrity is maintained.

- The regulation at 10 CFR 50.150(a)(2) states that the assessment must be based on the beyond-design-basis impact of a large commercial aircraft used for long-distance flights in the United States, with the aviation fuel loading typically used in such flights, and an impact speed and angle of impact considering the ability of both experienced and inexperienced pilots to control large commercial aircraft at the low altitude representative of a nuclear power plant’s low profile.

The first provision above requires the licensee or applicant to perform an assessment and identify and incorporate design features to protect the reactor core, containment, and spent fuel from the potential impact of a commercial aircraft. The licensee or applicant must show that these features protect the SSCs required to maintain core cooling or containment integrity, and to maintain spent fuel cooling or spent fuel pool integrity. The assessment required by this provision is Step A; this corresponds to the safety and hazards analyses and assessments of 10 CFR 50.34 and 10 CFR 52.79, which identify plant features that must remain reliable and available to perform their intended safety functions of maintaining core cooling or containment integrity and spent fuel cooling or spent fuel pool integrity.

In Step B, the licensee or applicant designs its protection strategy (e.g., relying on interposing structures, the design of building outer structures, the reinforcement of inner building walls for structural integrity, the use of fire-separating barriers, the fireproofing of structures, reconfiguration of the automatic fire suppression system, etc.) so that the plant can withstand beyond-design-basis impacts of commercial aircraft with the characteristics of 10 CFR 50.150(a)(2). The protection strategy, if adequately designed and incorporated, provides assurance that the required SSCs or barriers can perform the safety functions described in 10 CFR 50.150(a) with reduced use of operator actions. When the requirements for aircraft impact assessment above have been met through Steps A and B, the Commission will find that the licensee or applicant has shown reasonable assurance of adequate protection from the risk of the beyond-design-basis impact of a large commercial aircraft.

For this example, Step C would be a site-specific analysis to evaluate the potential offsite radiological consequences of a failure of the design features intended to protect against the impact of a large commercial aircraft. Such an analysis is unnecessary in this case, as it is also for the requirement of 10 CFR 73.55, because the licensee or applicant has already met the

Commission standard for adequate protection of the design features required to maintain core and spent fuel cooling or containment and spent fuel pool integrity against the beyond-design-basis impact of a large commercial aircraft.

As illustrated in the example, under the current security regulatory framework, the Commission makes a finding of adequate protection when a licensee or applicant has met the requirements of 10 CFR 73.55 to protect against threats up to and including the DBT of radiological sabotage. The proposed requirement of an additional consequence analysis has no regulatory or technical justification and is an unnecessary burden to licensees and applicants wishing to apply alternative physical security requirements in their physical protection program designs.

Potential Impact on Mission

The requirements of 10 CFR 73.55(s)(1), as proposed and implemented, will result in an unnecessarily large regulatory footprint and regulatory overreach and create an impediment to advanced reactor designers and applicants wishing to apply alternative methods or approaches to meet the requirements of 10 CFR 73.55.

This rule, if made final as proposed, will adversely affect the NRC's plan for efficiency, clarity, and reliability in accomplishing its mission, which is to license and regulate the Nation's civilian use of radioactive materials so as to provide reasonable assurance of adequate protection of public health and safety, to promote the common defense and security, and to protect the environment. Specifically, the adoption of the proposed rule and its implementation will result in the following:

- Inefficiency: The proposed regulations are not consistent with the degree of risk reduction they would achieve, as the requirements are unnecessary and would not minimize the use of resources or lead to regulatory decisions made without undue delay.

- Lack of clarity: The proposed regulations are not coherent, logical, and practical. There is no clear nexus between the proposed regulations and agency goals and objectives, whether explicitly or implicitly stated. The agency's longstanding principle of adequate protection would no longer be readily understood and easily applied.

- Absence of reliability: The proposed regulations would undermine the currently established regulations, which have been deemed reliable for maintaining acceptably low levels of risk based on the best available knowledge from research and operational experience, and considering safety and security interactions, technological uncertainties, and the diversity of licensee and regulatory activities. The proposed regulations would not be consistent with current regulations and would not contribute to regulatory stability for advanced reactors.

Proposed Alternative

The following changes to the proposed rule in 10 CFR 73.55(s)(1) would eliminate the unnecessary burden and remove regulatory impediments for an applicant or a licensee wishing to implement alternative measures:

(1) General requirements.

(i) *Applicability*. ~~The requirements of this section apply to~~ An applicant for or holder of a license for a small modular reactor, as defined in § 171.5 of this chapter, or a non-light-water reactor under part 50 of this chapter or part 52 of this chapter~~.~~ may elect to meet one or more of the alterative security requirements in § 73.55(s)(2).

~~(ii) *Eligibility*. The applicant or licensee must demonstrate that the consequences of a postulated radiological release that results from a postulated security-initiated event do not exceed the offsite dose reference values defined in §§ 50.34 and 52.79 of this chapter~~.

(ii~~i~~) *Identification and documentation*. The applicant or licensee must identify the specific alternative physical security requirement(s) it intends to implement as part of its physical protection program and demonstrate how the requirements set forth in this section are met when the selected alternative(s) is used.

~~(iv) *Analysis*. An applicant or licensee electing to meet one or more of the alternative security requirements in in paragraph (s)(2) of this section must perform a technical analysis demonstrating how it meets the criteria in paragraph (s)(1)(ii) of this section. The licensee must maintain the analysis until the certifications required by § 50.82(a)(1) of this chapter or § 52.110(a) of this chapter have been docketed by the NRC~~.

There are concerns about the use of “preventing significant core damage” as a performance objective for advanced reactor physical protection programs, since this objective would not encompass advanced reactors in which radiation hazards may reside outside of the reactor core in a reactor vessel. To address these concerns, the NRC should consider the following modification of 10 CFR 73.55(b)(3):

(b)(3) For a licensee holding an operating license under the provisions of part 50 of this chapter or a combined license under the provisions of part 52 of this chapter for a non-light-water reactor, other than a small modular reactor, as defined in § 171.5 of this chapter, the physical protection program must be designed to prevent significant core damage and spent fuel sabotage. For a small modular reactor licensee or a non-light-water reactor licensee licensed under part 50 of this chapter or part 52 of this chapter, the physical protection program must be designed to protect against the loss of structures, systems, components, and barriers that prevent a significant release of radionuclides from any source.

PROBLEM STATEMENT NO. 2

The proposed rule and its implementation set forth a radiation dose of 25 rem total effective dose equivalent (25 rem TEDE) (in any 2-hour period following the onset of the postulated fission product release) as an acceptable dose limit for members of the public and a consequence-based approach that uses this 25 rem TEDE as the acceptable criterion for determining offsite release that would not endanger public health and safety.

Regulatory and Technical Basis

- ***The proposed rule redefines a dose of up to 25 rem TEDE as acceptable level of exposure for members of the public. It applies this new standard as the threshold for the staff to determine whether a given release of radiation hazards is acceptable and will not endanger the public health and safety.***

The implementation of the proposed rule requiring a site-specific analysis to evaluate offsite consequences uses the value of 25 rem TEDE over a 2-hour duration as a consequence threshold, with doses up to 25 rem TEDE to members of the public being acceptable, and doses over 25 rem TEDE not being acceptable. For an acceptable implementation of 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv), a licensee or applicant wishing to demonstrate eligibility to use some or all of the alternative security measures in 10 CFR 73.55(s)(2) should develop scenarios testing its ability to uphold the site's physical security plan (e.g., to protect target set equipment or prevent an offsite release from exceeding reference doses) while employing the alternative measures. Possible scenarios to evaluate include, but are not limited to, the following:

(1) A DBT-initiated event that compromises some or all target sets and does not involve human actions to mitigate a potential radiological release. Such an event should not result in offsite doses above the reference values in 10 CFR 50.34(a)(1)(ii)(D)(*1*)–(*2*) and 10 CFR 52.79(a)(1)(vi)(A)–(B).

(2) A DBT-initiated event that compromises some or all target sets and results in core damage or causes a release of radionuclides from any source before offsite doses exceed the reference values in 10 CFR 50.34(a)(1)(ii)(D)(*1*)–(*2*) and 10 CFR 52.79(a)(1)(vi)(A)–(B). The response to such an event may involve both onsite and offsite resources to interdict the adversary force and mitigate the release.

The consequence analysis required by the proposed rule, as described for implementation above, uses a threshold of 25 rem TEDE in a 2-hour period as an acceptable dose limit for members of the public. For comparison, 25 rem TEDE is the dose limit for workers performing emergency services to save lives or protect large populations (without informed consent).

Contrary to the proposed rule, the differing view problem statement is that the 25 rem TEDE consequence threshold used in the proposed rule far exceeds the Commission's established dose limits in 10 CFR 20.1301, "Dose limits for individual members of the public," which are 2 mrem per hour and 100 mrem per year for individual members of the public, excluding dose from background radiation and medical exposure.

- ***Establishing the dose to emergency workers as acceptable for members of the public conflicts with Commission regulations.***

Currently, the dose limits for workers performing emergency services to save lives or protect large populations are (1) greater than 25 rem TEDE only on a voluntary basis, for persons informed of the risk and selected healthy individuals, preferably over the age of 45, and (2) up to 25 rem TEDE (without informed consent) when a lower dose limit is not practicable. It should be emphasized that these limits apply to emergency conditions. In addition, the dose limit associated with the protection of valuable property is up to 10 rem when a lower dose is not practicable, or by planned special exposure if time permits. For a worker recovering deceased victims, the dose limit is no more than 5 rem or by planned special exposure.

The proposed rule and implementation set forth a dose limit for members of the public that equals the current limit for workers performing emergency services. Unlike radiation workers, members of the public are not informed individuals who give consent and willingly, knowingly, and voluntarily accept the risks of radiation exposure. The implementation of the proposed rule would suggest, through guidance, that a drastically higher dose limit (12,500 mrem per hour as opposed to 2 mrem per hour) is acceptable for members of the public. This limit exceeds the 10-rem limit established for emergency workers protecting valuable property. It also exceeds other public limits. For example, in 40 CFR Part 190, “Environmental Radiation Protection Standards for Nuclear Power Operations,” the Environmental Protection Agency establishes a dose limit of 25 mrem per year as acceptable for any member of the public. This annual dose rate is thousands of times lower than the 25,000 mrem in 2 hours that the proposed guidance considers acceptable.

The dose limit of 25 rem TEDE for members of the public is not supported by current NRC regulations or by regulations outside of the agency. The differing view problem statement is that, through guidance on implementing the proposed requirements of 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.5(s)(1)(iv), the staff has set forth a new Commission standard on acceptable dose limit for members of the public that equals the limit for radiation workers, namely 25 rem TEDE.

- ***The use of 25 rem TEDE as a consequence-based criterion is outside of the current regulatory framework for safety analyses, assessments, or evaluations.***

The regulations at 10 CFR 50.34(a)(1)(ii) and 10 CFR 50.34(a)(1)(ii)(D) establish the regulatory basis for using 25 rem TEDE as a reference value in the evaluation of plant design features with respect to postulated reactor accidents, in order to ensure extremely low risk of reactor accidents and low risk of public exposure to radiation. Specifically, 10 CFR 50.34(a)(1)(ii) states that the preliminary safety analysis report must include the following:

> A description and safety assessment of the site and a safety assessment of the facility. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products.

The areas to be covered by the safety assessments include those described in 10 CFR 50.34(a)(1)(ii)(D):

> The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release[6] from the core into the containment assuming that the facility is operated at the ultimate power contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that:
>
> (1) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem[7] total effective dose equivalent (TEDE).
>
> (2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE);

Footnote 6 to these regulations clarifies the following:

> The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

Footnote 7 states the following:

> A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, in order to assure that such designs provide assurance of low risk of public exposure to radiation, in the event of such accidents.

The implementation guidance for the analysis required by paragraphs (ii) and (iv) of the proposed rule, 10 CFR 73.55(s)(1), confirms as acceptable a consequence-based approach, rather than a risk-based approach (i.e., one based on the product of consequence and likelihood). Current NRC regulations promote a risk-based approach, in line with the NRC's goal of being a risk-informed and performance-based regulator. In the risk-based approach, a physical protection program design could achieve high assurance of protection by ensuring that the DBT of radiological sabotage would have very low likelihood. Contrary to the established regulations and regulatory framework, the consequence-based approach using 25 rem TEDE as an acceptable consequence for public health and safety conflicts with the Commission's risk-based approach. This consequence-based approach has not been reviewed or approved by the Commission, especially not in the context of the limited-scope security rulemaking for advanced reactors.

Furthermore, under the consequence-based approach using the 25 rem TEDE threshold, since offsite release of radiation hazards up to 25 rem TEDE would constitute an acceptable dose to the public, the plant features and barriers preventing such a release would no longer be considered safety-related, and licensees would not be required to protect them accordingly. Specifically, SSCs that would be treated as safety-related under the current regulations (for licensees and applicants applying 10 CFR 50.69, these include RISC-2 SSCs, which are nonsafety-related but perform safety-significant functions, and any equipment or systems whose failure would lead to common-cause failure of RISC-1 or RISC-2 SSCs) could be reclassified as RISC-3 or RISC-4, corresponding respectively to safety-related or nonsafety-related SSCs that perform functions of low safety significance. Under the consequence-based approach of the proposed rule with a reference value of 25 rem TEDE, licensees would no longer have to protect SSCs and barriers as target set elements if their failure would result in offsite release of no more than 25 rem TEDE. The approach would no longer be risk-based or risk-informed, and would allow licensees to reduce or eliminate SSCs and barriers that would otherwise be categorized and treated as being required for assurance of low risk of public exposure to radiation. Licensees would no longer be required to protect these SSCs and barriers from the DBT of radiological sabotage.

Based on the regulatory and technical discussions above, the differing view problem statement is that the proposed rule and its implementation guidance should not assert that 25 rem TEDE is an acceptable dose limit for members of the public. This limit was established for emergency workers, and the NRC's current regulations and regulatory framework do not support its use for members of the public, nor has the Commission considered or approved it.

In addition, the use of 25 rem TEDE as an acceptance criterion in a consequence-based approach is contrary to the current regulations and the risk-based (i.e., risk-informed and performance-based) regulatory framework. It falls outside of the Commission's directions and the scope of the limited-scope rule on security for advanced reactors using alternatives in designs of physical protection programs.

Potential Impact on Mission

The adoption of this proposed rule and its implementation, which set forth a consequence-based approach with 25 rem TEDE as the acceptance criterion, will adversely affect the NRC's plan for efficiency, clarity, and reliability in accomplishing its mission, which is to license and regulate the Nation's civilian use of radioactive materials so as to provide reasonable assurance of adequate protection of public health and safety, and to promote the common defense and security, and to protect the environment. Specifically, the adoption of the proposed rule and its implementation will result in the following:

- Inefficiency: The proposed regulations are not consistent with the degree of risk reduction they would achieve, as the consequence-based approach using 25 rem TEDE as an acceptable dose limit for members of the public is contrary to the current risk-based (risk-informed and performance-based) regulatory framework. The risk-based approach is a longstanding standard practice for reasonable assurance of protection of public health and safety. The implementation of a consequence-based approach that conflicts with or undermines the current risk-based safety and security framework would cause undue delay in regulatory decisions and could compromise the safety bases for findings of reasonable assurance of adequate protection.

- Lack of clarity: The proposed regulations are not coherent, logical, and practical. There is no clear nexus between the proposed regulations and agency goals and objectives, whether explicitly or implicitly stated. The consequence-based approach in the proposed regulations would circumvent the agency's longstanding principle of adequate protection through a risk-informed and performance-based approach, and the regulations and regulatory framework would no longer be readily understood and easily applied.

- Absence of reliability: The proposed regulations would undermine the currently established regulations, which have been deemed reliable for maintaining acceptably low levels of risk based on the best available knowledge from research and operational experience, and considering safety and security interactions, technological uncertainties, and the diversity of licensee and regulatory activities. The consequence-based criteria (with the 25 rem TEDE threshold) in the proposed regulations would not result in prompt, fair, and decisive administration and would not contribute to regulatory stability for advanced reactors.

<u>Proposed Alternative</u>

The following changes to the proposed rule in 10 CFR 73.55(s)(1) would eliminate the use of the consequence-based approach and the 25 rem TEDE acceptance criterion and would thus resolve the differing view problem statement:

> (1) General requirements.
> (i) *Applicability*. An applicant or licensee of a small modular reactor, as defined in § 171.5 of this chapter, or non-light-water reactor that is licensed under part 50 of this chapter or part 52 of this chapter may elect to meet one or more of the alterative security requirements in § 73.55(s)(2).
>
> ~~(ii) *Eligibility*. The applicant or licensee must demonstrate that the consequences of a postulated radiological release that results from a postulated security-initiated event do not exceed the offsite dose reference values defined in §§ 50.34 and 52.79 of this chapter~~.
>
> (ii~~i~~) *Identification and documentation*. The applicant or licensee must identify the specific alternative physical security requirement(s) it intends to implement as

part of its physical protection program and demonstrate how the requirements set forth in this section are met when selected alternative(s) is used.

~~(iv) Analysis. An applicant or licensee electing to meet one or more of the alternative security requirements in in paragraph (s)(2) of this section must perform a technical analysis demonstrating how it meets the criteria in paragraph (s)(1)(ii) of this section. The licensee must maintain the analysis until the certifications required by § 50.82(a)(1) of this chapter or § 52.110(a) of this chapter have been docketed by the NRC~~.

There are concerns about the use of "preventing significant core damage" as a performance objective for advanced reactor physical protection programs, since this objective would not encompass advanced reactors in which radiation hazards may reside outside of the reactor core in a reactor vessel. To address these concerns, the NRC should consider the following modification of 10 CFR 73.55(b)(3):

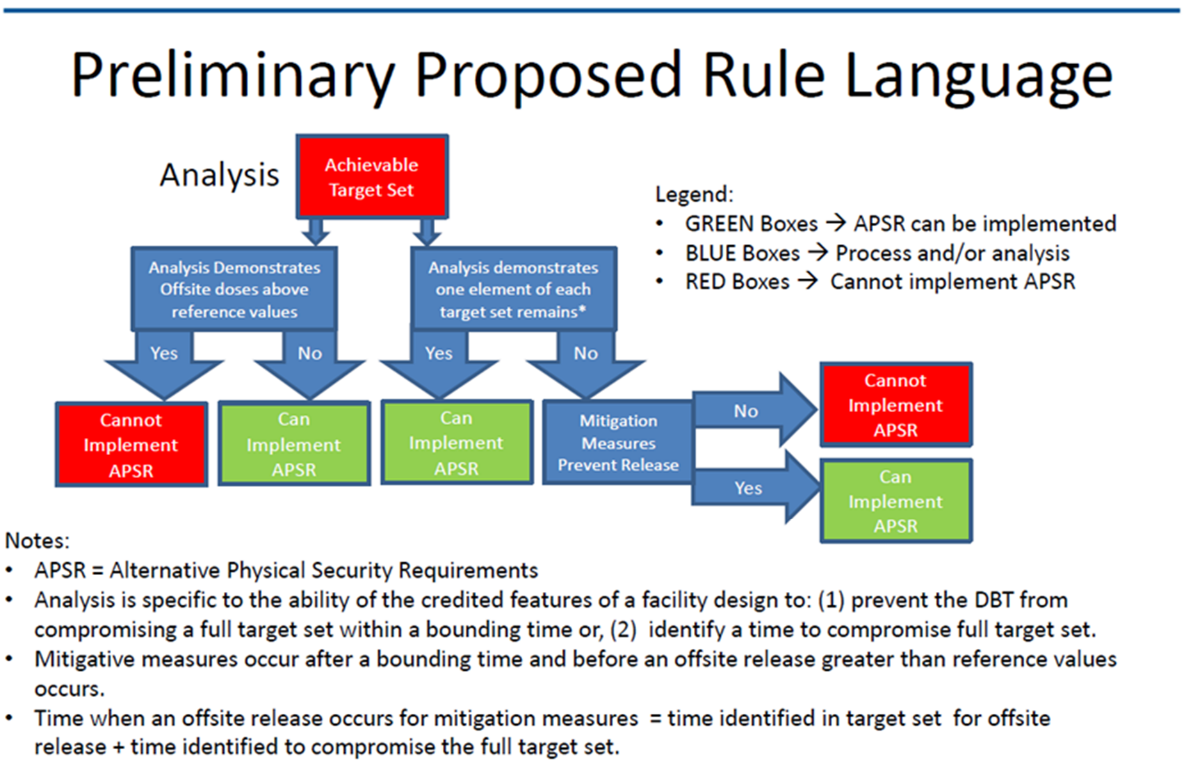
(b)(3) For a licensee holding an operating license under the provisions of part 50 of this chapter or a combined license under the provisions of part 52 of this chapter for a light light-water reactor, other than a small modular reactor, as defined in § 171.5 of this chapter, the physical protection program must be designed to prevent significant core damage and spent fuel sabotage. For a small modular reactor licensee or a non-light-water reactor licensee licensed under part 50 of this chapter or part 52 of this chapter, the physical protection program must be designed to protect against the loss of structures, systems, components, and barriers that prevent a significant release of radionuclides from any source.

PROBLEM STATEMENT NO. 3a

The implementation of 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv) is contrary to the Commission's expectation, established in the Policy Statement on the Regulation of Advanced Reactors (Volume 73 of the *Federal Register*, page 60612; October 14, 2008), that advanced reactor designs should include reduced reliance on human actions.

Technical Basis

The acceptable implementation of the proposed rule requires that an analysis be performed to evaluate potential offsite consequences based on a consequence threshold of 25 rem TEDE, including additional analyses of security-initiated (i.e., DBT-initiated) scenarios, as illustrated below.



As stated earlier, this illustration was presented during public meetings on October 19, 2021, and January 20, 2022. It shows how the implementation of the proposed rule would allow for a licensee or applicant, through analysis performed under the proposed rule, to substitute mitigation measures relying on human actions for plant features, SSCs, and barriers identified through safety analyses, assessments, and evaluations in the safety and design bases. For example, under the proposed rule, the SSCs and barriers maintaining core cooling or containment integrity and spent fuel cooling or spent fuel pool integrity may be eliminated if analysis shows that mitigation measures can prevent any release resulting in a dose above 25 rem TEDE.

In addition, the illustration shows how the proposed rule undermines the current safety and security regulatory framework by allowing a licensee or applicant to downgrade the categorization, and the associated treatment, of SSCs and barriers based on 10 CFR 50.69. That is, under the proposed rule, a licensee or applicant may show that with mitigation measures, the failure of certain SSCs and barriers that are currently classified as RISC-1 or RISC-2 (or whose failure would lead to common-cause failure of RISC-1 or RISC-2 SSCs) would result in offsite release of no more than 25 rem TEDE. The licensee could then reclassify these SSCs and barriers as RISC-3 or RISC-4 or eliminate them altogether, considering them unnecessary for ensuring low risk of offsite release, although they are considered necessary under current design requirements for safety.

In relation to the safety/security interface, the implementation of the proposed rule allows the licensee or applicant to use a consequence analysis, based on consequences mitigated by reliance on human actions, to justify reducing the plant security posture, eliminating the protection of SSCs and barriers that otherwise would have been identified as target set equipment and protected by the design of the physical protection program. It should be noted that mitigation measures would be applied within a defense-in-depth strategy, to provide sufficient margin in safety and security designs to account for the uncertainties in the risk of public exposure to radiation resulting from design-basis accidents and beyond-design-basis events (e.g., aircraft impacts, Fukushima Dai-ichi event, and the DBT of radiological sabotage).

The following discussion uses the previous example of aircraft impact assessment to illustrate how the proposed rule may be implemented. Mitigation measures for potential consequences of an aircraft impact are required by 10 CFR 50.54(hh)(1), which states, in part, the following:

> Each licensee shall develop, implement and maintain procedures that describe how the licensee will address the following areas if the licensee is notified of a potential aircraft threat:
>
> (iii) Contacting all onsite personnel and applicable offsite response organizations;
>
> (iv) Onsite actions necessary to enhance the capability of the facility to mitigate the consequences of an aircraft impact;
>
> (vi) Dispersal of equipment and personnel, as well as rapid entry into site protected areas for essential onsite personnel and offsite responders who are necessary to mitigate the event; and
>
> (vii) Recall of site personnel.

If the licensee or applicant has procedures to address the onsite actions necessary to enhance the capability of the facility to mitigate the consequences, dispersal of equipment and personnel and the recall of site personnel, as specified in 10 CFR 50.54(hh)(1), then under the proposed rule, human actions may be substituted for protection against potential offsite consequences, based on consequence analysis using the 25-rem TEDE criterion. With respect to security, the design of the physical protection program may be based on mitigation measures established to satisfy 10 CFR 50.54(hh)(1), without the design features, SSCs, or barriers identified as necessary through safety analysis, assessments, and evaluations. The level of safety for advanced reactors licensed under this framework would not equal that of currently licensed reactors, which rely on design features, engineered SSCs, and barriers, together with planning and contingencies for mitigation measures, to reduce risk and establish defense in depth.

The Commission's 2008 Policy Statement on the Regulation of Advanced Reactors states the following:

> Regarding advanced reactors, the Commission expects, as a minimum, at least the same degree of protection of the environment and public health and safety and the common defense and security that is required for current generation light-water reactors (LWRs). Furthermore, the Commission expects that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions.

Among the attributes that the Commission recommends for consideration in advanced reactor designs are the following:

- Designs that minimize the potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity, and independence in safety systems, with an emphasis on minimizing the potential for accidents over minimizing the consequences of such accidents….

- Designs that incorporate the defense-in-depth philosophy by maintaining multiple barriers against radiation release, and by reducing the potential for, and consequences of, severe accidents….

- Designs that include considerations for safety and security requirements together in the design process such that security issues (e.g., newly identified threats of terrorist attacks) can be effectively resolved through facility design and engineered security features, and formulation of mitigation measures, with reduced reliance on human actions.

The Commission also expects that "the safety features of these advanced reactor designs will be complemented by the operational program for Emergency Planning."

It is recognized that neither the Commission's policy statements nor staff-developed regulatory guides constitute regulatory requirements. Licensees and applicants are not obligated to adhere, in full or in part, to Commission policy statements or NRC-issued regulatory guides. They may choose to apply the staff guidance from a regulatory guide in full or in part, as they see fit, or to modify it, or to use other methods than described.

In particular, licensees and applicants are not legally obligated to comply with the guidance in DG-5071, "Target Set Identification and Development for Nuclear Power Reactors," or in DG-5072, "Guidance for Alternative Physical Security Requirements for Non-Light-Water Reactors and Small Modular Reactors." Neither the language nor the regulatory history of 10 CFR 73.55(f), on target sets, compels the interpretation that a licensee or applicant must conform to the guidance in DG-5071 and DG-5072, or to apply the guidance in one before or in consideration of applying the other, to perform the analysis required by the proposed rule in 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv).

Contrary to the expectations in the Commission's policy statement, the staff position on an acceptable method for the analysis required by the proposed rule provides for reliance on human actions to perform mitigation measures that would allow licensees and applicants to eliminate design features, SSCs, and barriers identified as necessary in the safety analysis, or to downgrade their risk categorization and treatment. As discussed above, this reduces both safety and security and compromises defense in depth. It also contravenes the Commission's expectation that safety features of advanced reactor designs will be complemented by emergency preparedness planning and response; it allows for safety features and SSCs instead to be replaced by mitigation measures.

In summary, the proposed rule allows licensees and applicants to rely on human actions in place of designed and engineered safety features. This is contrary to the Commission's expectation that advanced reactor designs should emphasize safety and security through design and engineering features, complemented by mitigation measures, with reduced reliance on human actions. The proposed rule allows licensees and applicants to eliminate the physical protection of design features, SSCs, and barriers that would otherwise be protected from the DBT of radiological sabotage. The differing view problem statement is that the proposed rule, through the analysis described in its implementation, should not allow licensees and applicants to rely on mitigation measures (human actions) for safety, because this would conflict with the Commission's expectations in the Policy Statement on the Regulation of Advanced Reactors.

Potential Impact on Mission

The proposed rule and its implementation, by allowing for reliance on mitigation measures (human actions), will affect the effectiveness of the NRC's licensing and regulations in providing reasonable assurance of adequate protection of public health and safety. Specifically, the adoption of the proposed rule and its implementation will result in the following:

- Inefficiency: The proposed regulations are not consistent with the degree of risk reduction they would achieve, as their implementation would allow for mitigation measures (human actions) to replace adequate physical protection of safety-related design features, SSCs, and barriers for preventing an offsite radiological release. This is contrary to safety requirements and to the risk-based approach of the current regulatory framework. It would reduce the security measures protecting against the DBT of radiological sabotage for advanced reactors, which would cause undue delay in regulatory decisions and potentially undo the current safety licensing basis for findings of reasonable assurance of protection.

- Lack of clarity: The proposed regulations are not coherent, logical, and practical. There is no clear nexus between the proposed regulations and agency goals and objectives, whether explicitly or implicitly stated. The consequence-based approach and reliance on mitigation measures in the proposed regulations undermine the agency's longstanding position of using a risk-based approach to apply established requirements for adequate protection. The regulations and regulatory framework would no longer be readily understood and easily applied.

- Absence of reliability: The proposed regulations would undermine the currently established regulations, which have been deemed reliable for maintaining acceptably low levels of risk based on the best available knowledge from research and operational experience, and considering safety and security interactions, technological uncertainties, and the diversity of licensee and regulatory activities. The implementation of mitigation measures relying on human actions, rather than on design features, SSCs, and barriers, would not be consistent with current regulations and would not lead to prompt, fair, and decisive administration contributing to regulatory stability for advanced reactors.

Proposed Alternative

Paragraphs (ii) and (iv) of the proposed rule in 10 CFR 73.55(s)(1) should be removed to eliminate the requirement to perform an analysis to evaluate potential offsite consequences. In the current security framework of 10 CFR 73.55, the design of the physical protection program is aimed at protecting the design features, SSCs, and barriers that have been determined, through a safety analysis, to be necessary for assurance of adequate safety. The suggested change removes the proposed requirement that would allow licensees and applicants to circumvent the current safety and security requirements. It also removes the reliance on human actions implied by the implementation of the proposed rule, which is contrary to the Commission's Policy Statement on the Regulation of Advanced Reactors. The suggested change is to remove paragraphs (ii) and (iv) from the proposed rule text of 10 CFR 73.55(s)(1):

> ~~(ii) Eligibility. The applicant or licensee must demonstrate that the consequences of a postulated radiological release that results from a postulated security-initiated event do not exceed the offsite dose reference values defined in §§ 50.34 and 52.79 of this chapter.~~
>
> ~~(iv) Analysis. An applicant or licensee electing to meet one or more of the alternative security requirements in in paragraph (s)(2) of this section must perform a technical analysis demonstrating how it meets the criteria in paragraph (s)(1)(ii) of this section. The licensee must maintain the analysis until the certifications required by § 50.82(a)(1) of this chapter or § 52.110(a) of this chapter have been docketed by the NRC~~.

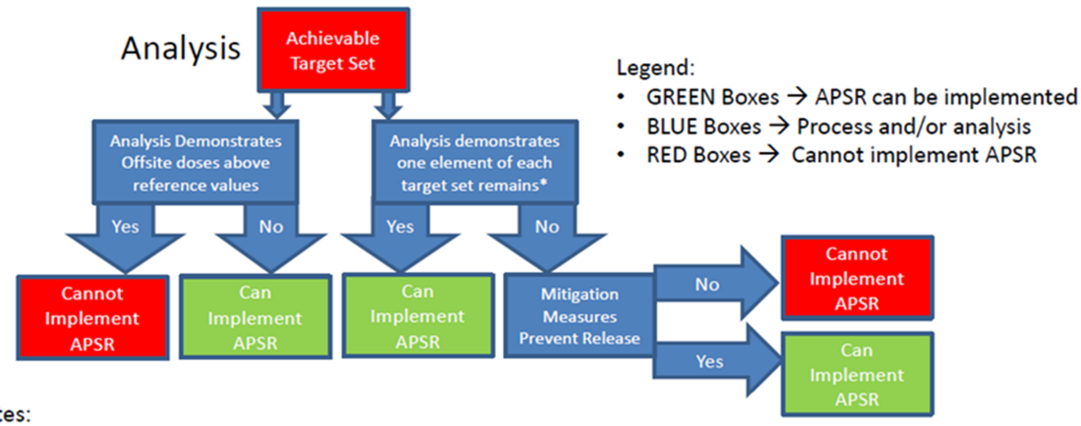
PROBLEM STATEMENT NO. 3b

The proposed rule, at 10 CFR 73.55(s)(1)(ii) and (iv), introduces a more specific requirement for consequence analysis in 10 CFR 73.55 and provides a regulatory pathway for circumventing requirements established in the current safety and security framework for power reactors.

Regulatory Basis

The acceptable implementation of the proposed rule requires that an analysis be performed to evaluate potential offsite consequences based on a consequence threshold of 25 rem TEDE, including additional analyses of security-initiated (i.e., DBT-initiated) scenarios, as illustrated below.

Preliminary Proposed Rule Language



Notes:

- APSR = Alternative Physical Security Requirements
- Analysis is specific to the ability of the credited features of a facility design to: (1) prevent the DBT from compromising a full target set within a bounding time or, (2) identify a time to compromise full target set.
- Mitigative measures occur after a bounding time and before an offsite release greater than reference values occurs.
- Time when an offsite release occurs for mitigation measures = time identified in target set for offsite release + time identified to compromise the full target set.

As previously stated, the illustration above, presented during public meetings on October 19, 2021, and January 20, 2022, captures a method that the staff finds acceptable for performing the analysis required under the proposed provisions of paragraphs (ii) and (iv) of 10 CFR 73.55(s)(1). The implementation guidance for the proposed rule appears in DG-5071 and DG-5072.

The proposed rule, in 10 CFR 73.55(s)(1)(ii) and (iv), gives more specific requirements for analysis of potential offsite consequences. It provides a regulatory pathway for circumventing the regulatory requirements established in the current framework for the safety of nuclear power reactors, such as the provisions of 10 CFR 50.34(a)(1)(ii)(D) for analysis, assessment, and evaluation of offsite consequences. The more specific provisions of 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv) would control over the more general provisions of 10 CFR 50.34(a)(1)(ii)(D) for analysis of potential offsite consequences. The provisions of 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv) on analysis of offsite consequences are narrower in scope than the provisions in 10 CFR Parts 50, 52, and 100 (e.g., in 10 CFR 50.34, 10 CFR 50.69, 10 CFR 52.79, and 10 CFR 100.11, “Determination of exclusion area, low population zone, and population center distance”).

The example below, on specific exemptions, illustrates how a more specific provision would control over a more general provision. The regulations in 10 CFR 50.12, “Specific exemptions,” state the following:

> (a) The Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part, which are—
>
> (1) Authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security.
>
> (2) The Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever—
>
> (i) Application of the regulation in the particular circumstances conflicts with other rules or requirements of the Commission; or
>
> (ii) Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule; or
>
> (iii) Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated; or
>
> (iv) The exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption; or
>
> (v) The exemption would provide only temporary relief from the applicable regulation and the licensee or applicant has made good faith efforts to comply with the regulation; or
>
> (vi) There is present any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption. If such condition is relied on exclusively for satisfying paragraph (a)(2) of this section, the exemption may not be granted until the Executive Director for Operations has consulted with the Commission.

The regulations in 10 CFR 52.7, "Specific exemptions," state the following:

> The Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part. The Commission's consideration will be governed by § 50.12 of this chapter, unless other criteria are provided for in this part, in which case the Commission's consideration will be governed by the criteria in this part. Only if those criteria are not met will the Commission's considerations be governed by § 50.12 of this chapter. The Commission's consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts.

The regulations in 10 CFR 73.5 state the following:

> The Commission may, upon application of any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in this part as it determines are authorized by law and will not endanger life or property or the common defense and security, and are otherwise in the public interest.

The regulations in 10 CFR 73.5 address specific exemptions to the requirements of 10 CFR Part 73. The regulations in 10 CFR 50.12 and 10 CFR 52.7 set forth the criteria by which the Commission may grant exemptions to the requirements of 10 CFR Part 50 and 10 CFR Part 52, respectively.

The more specific provision in 10 CFR 73.5 is controlling over the more general provisions in 10 CFR 50.12 and 10 CFR 52.7. The criteria for approval of exemptions under 10 CFR 73.5 are specific to security objectives and are narrower in scope than the general exemption criteria in 10 CFR 50.12 and 10 CFR 52.7. In the last quarter of 2020, the Commission granted the requests of currently operating power reactor licensees to follow the provisions of 10 CFR 73.5 over those of 10 CFR 50.12; this demonstrates that the more specific requirement, in this case that of 10 CFR 73.5, is controlling.

With respect to the proposed rule, this means that a licensee or applicant would have to perform the analyses of 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv) in lieu of the analyses, assessments, and evaluations of potential offsite consequences required under the general provisions. A licensee or applicant could no longer rely on the latter (e.g., on 10 CFR 50.34(a)(1)(ii)(D)) or on the results of analyses that did not cover security-initiated (i.e., DBT-initiated) events beyond those required in the current regulatory framework for safety. Licensees and applicants would no longer be obligated to meet both the general and the specific provisions for analysis of potential offsite consequences; through the specific provisions of 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv), they could consider and incorporate mitigation measures (human actions) to achieve protection of advanced reactors. The differing view problem statement is that the proposed rule would control over the more general provisions on analysis of potential offsite consequences (e.g., 10 CFR Parts 50, 52, and 100), thus permitting a licensee or applicant to circumvent the regulatory requirements in the current framework for safety and security for power reactors.

<u>Potential Impact on Mission</u>

This proposed rule, incorporating specific provisions in 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv) that would be controlling over more general provisions for analysis of potential offsite consequences, will affect the effectiveness of the NRC's licensing and regulations in providing reasonable assurance of adequate protection of public health and safety. Specifically, adoption of the proposed rule will result in the following:

- Inefficiency: The proposed regulations are inconsistent with the degree of risk reduction they achieve, as their implementation would allow for mitigation measures (human actions) to replace adequate physical protection of safety-related design features, SSCs and barriers for preventing an offsite radiological release. The method proposed as acceptable under 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv) would circumvent the safety requirements and the risk-based approach of the current regulatory framework. It would also reduce security, causing undue delay in regulatory decisions and potentially undoing the current safety licensing basis for findings of reasonable assurance of adequate protection.

- Lack of clarity: The proposed regulations are not coherent, logical, and practical. There is no clear nexus between the proposed regulations and agency goals and objectives, whether explicitly or implicitly stated. The proposed security requirements in 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv), which are based on a consequence analysis allowing reliance on human actions for mitigation, would be controlling over the more general requirements elsewhere in the NRC's regulations, and would therefore undermine the agency's longstanding principle of adequate protection through a risk-based approach. The regulations and regulatory framework would no longer be readily understood and easily applied.

- Absence of reliability: The proposed regulations would undermine the currently established regulations, which have been deemed reliable for maintaining acceptably low levels of risk based on the best available knowledge from research and operational experience, and considering safety and security interactions, technological uncertainties, and the diversity of licensee and regulatory activities. The implementation of the proposed requirements would be inconsistent with current regulations and would allow a licensee or applicant to circumvent current safety requirements. It would not lead to prompt, fair, and decisive administration or contribute to regulatory stability for advanced reactors.

Proposed Alternative

The NRC should remove 10 CFR 73.55(s)(1)(ii) and 10 CFR 73.55(s)(1)(iv) in the proposed rule to eliminate the requirement to perform an analysis to evaluate potential offsite consequences. This will ensure that there is no specific provision for such analysis that would be controlling over the general provisions elsewhere in the regulations for analysis, assessments, and evaluations of potential offsite consequences. In the current security framework of 10 CFR 73.55, the design of the physical protection program is aimed at protecting the design features, SSCs, and barriers that have been determined to be necessary for assurance of adequate safety. The suggested change removes the provisions that would allow licensees and applicants to circumvent the current requirements for analysis, assessments, and evaluations for safety and security. The suggested change is to remove the proposed requirements in 10 CFR 73.55(s)(1)(ii) and (iv):

> ~~(ii) Eligibility. The applicant or licensee must demonstrate that the consequences of a postulated radiological release that results from a postulated security-initiated event do not exceed the offsite dose reference values defined in §§ 50.34 and 52.79 of this chapter.~~
>
> ~~(iv) Analysis. An applicant or licensee electing to meet one or more of the alternative security requirements in in paragraph (s)(2) of this section must perform a technical analysis demonstrating how it meets the criteria in paragraph (s)(1)(ii) of this section. The licensee must maintain the analysis until the certifications required by § 50.82(a)(1) of this chapter or § 52.110(a) of this chapter have been docketed by the NRC~~.