



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555-0001

July 31, 2025

Michelle Catts, Senior Vice President  
Nuclear Programs  
GE-Hitachi Nuclear Energy  
3901 Castle Hayne Road  
Wilmington, NC 28402

SUBJECT: U.S. NUCLEAR REGULATORY COMMISSION STAFF RESPONSE TO  
INQUIRY TO SUPPORT BWRX-300 SAFETY STRATEGY LICENSING  
TOPICAL REPORT REVIEW

Dear Ms. Catts:

In your letter dated September 10, 2024, GE-Hitachi (GEH) requested clarification on whether ensuring [[

.]]<sup>1,2</sup> GEH also requested the U.S. Nuclear Regulatory Commission (NRC) [[

.]]” The NRC staff’s response is provided below.

10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” establishes, among other things, “the minimum requirements for principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission.”<sup>3</sup> Under the provisions of 10 CFR 50.34, an applicant for a construction permit must include the principal design criteria for the facility in their application. General Design Criterion (GDC) 10, “Reactor design,” states that appropriate margin exists to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of anticipated operational

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<sup>1</sup> Letter from S. Karkour to USNRC, “Request for Legal Interpretation to Support BWRX-300 Safety Strategy LTR Review,” dated September 10, 2024, Agencywide Document Access and Management System (ADAMS) Package Accession No. ML24254A447.

<sup>2</sup> 10 CFR 50.2, “Definitions:” Safety-related structures, systems and components means those structures, systems and components that are relied upon to remain functional during and following design basis events to assure: (1) The integrity of the reactor coolant pressure boundary; (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

<sup>3</sup> 10 CFR Part 50, Appendix A, Introduction.

occurrences (AOOs). SAFDLs are design-specific limits proposed by applicants and reviewed by the NRC<sup>4</sup>.

The NRC staff considers any SSC relied on to satisfy the GDCs, [[  
 ]], to be an SSC that is “important to safety.” As discussed in Generic Letter (GL) 84-01<sup>5</sup>, the terms “important to safety” and “safety-related” are not synonymous. The term “important to safety” refers to the broad scope of equipment defined in the GDCs while “safety-related” refers to a narrower set of equipment meeting the § 50.2 definition. “Important to safety” SSCs that also perform functions that meet the “safety-related” definition in § 50.2 are classified as “safety-related.” Depending on the safety classification of an SSC, different quality assurance requirements may apply. For “important to safety” SSCs, GDC Criterion 1, “Quality standards and records,” specifies that a quality assurance program be established and implemented to provide adequate assurance that these SSCs will satisfactorily perform their safety functions specified in the GDCs. “Safety-related” SSCs are subject to the special treatment requirements for “safety-related” SSCs contained in 10 CFR Part 50 (e.g., 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants”).

In determining whether or not an “important to safety” SSC is also subject to the additional “safety-related” special treatment requirements, the SSC’s functions are evaluated against the “safety-related” definition in § 50.2. Specifically, “safety-related” SSCs are relied upon to remain functional during and following design basis events to assure the following functional capabilities are met: (1) integrity of the reactor coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) the capability to prevent or mitigate the consequence of certain accidents. The staff considers AOOs to be design basis events.<sup>6</sup> If an SSC is relied upon to meet both the GDCs and the “safety-related” functional capabilities specified in § 50.2, it must meet both the 10 CFR Part 50 “safety-related” special treatment requirements and the programmatic quality standards of GDC Criterion 1.

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 ]].  
 The term “safe shutdown” is used in several different regulatory contexts including the definition of safety-related SSCs, fire protection, station blackout, and combustible gas control. These different regulatory contexts result in the term “safe shutdown” being

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<sup>4</sup> The staff uses the guidance contained in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 4.2, “Fuel System Design,” when evaluating applicant proposed SAFDLs.

<sup>5</sup> GL 84-01, “NRC Use of the Terms, ‘Important to Safety’ and ‘Safety Related’,” January 5, 1984, and enclosures, ADAMS Accession No. ML031150515.

<sup>6</sup> For example, in the context of environmental qualification, 10 CFR 50.49 defines design basis events as “conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed” to ensure safety-related functions are met.

applied differently depending on event type. When addressing this issue in the context of passive heat removal systems, the NRC defined a “safe shutdown” condition as the reactor is subcritical, reactor heat is being removed, and radioactive materials containment are properly maintained for the long term.<sup>7</sup>

The SAFDLs are not referenced in the § 50.2 definition of “safety-related” SSCs and the term is only used in the GDCs. However, SAFDLs have been used by applicants and licensees to support a demonstration of the capability of “safety-related” SSCs to maintain a safe shutdown condition during and following AOOs as described by § 50.2. [[

]]. GEH will also need to demonstrate that all the “safety-related” criteria of §50.2 can be fully met using only safety-related SSCs during and following design basis events, including AOOs.

[[

]]. However, the NRC staff did not rely on these precedents in formulating this response. Therefore, our response should not be viewed as an endorsement or agreement with the relevance of the precedents you provided to this issue.

[[

]]” the NRC staff applies the regulatory requirements in 10 CFR Part 50 when reviewing usage of these terms. Additional information on this topic can be found in GL 84-01.

This letter provides the NRC staff’s views and does not constitute a safety finding for the BWRX-300 design. The NRC staff will evaluate the safety of the design of the BWRX-300 during future licensing activities in accordance with 10 CFR Part 50 or 10 CFR Part 52.

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<sup>7</sup> SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” March 28, 1994, ADAMS Accession No. ML003708068, and associated Staff Requirements Memorandum, ADAMS Accession No. ML003708098.

M. Catts

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If you have any questions, please contact Stacy Joseph at 301-415-3256 or via email at [Stacy.Joseph@nrc.gov](mailto:Stacy.Joseph@nrc.gov).

Sincerely,

**/RA/**

Michele Sampson, Director  
Division of New and Renewed License  
Office of Nuclear Reactor Regulation

Docket No. 99900003

cc: Suzanne Karkour, GEH  
Kelli Roberts-Banks, GEH

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