UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

June 20, 2026

The Honorable Ho K. Nieh
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE CONSTRUCTION PERMIT APPLICATION FOR A BWRX-300 REACTOR AT THE CLINCH RIVER SITE

Dear Chairman Nieh:

During the 736th meeting of the Advisory Committee on Reactor Safeguards (ACRS), held from June 3 through 4, 2026, we completed our review of the safety aspects of the construction permit application (CPA) for a single unit, BWRX-300 plant, designated as Clinch River Nuclear Unit 1 (CRN-1), and the NRC staff's associated Advanced Safety Evaluation Report (SER). Our BWRX-300 Design Center Subcommittee reviewed this matter during subcommittee meetings on August 20, 2025, and April 7, 2026. During this review, we had the benefit of prior review of topical reports, discussions with U.S. Nuclear Regulatory Commission (NRC) staff and representatives from the applicant, Tennessee Valley Authority (TVA) and their chosen vendor General Electric Vernova Hitachi (GVH). We also benefitted from reviewing the reference documents.

This report fulfills the requirements of Section 182b of the Atomic Energy Act, as amended.[1]

CONCLUSIONS AND RECOMMENDATIONS

1. The construction permit application describes an innovative, natural-circulation boiling water reactor (BWR) design with a simplified plant architecture, passive decay heat removal, and coolant retention based on reactor isolation. The design and the preliminary safety analysis report (PSAR) were sufficiently complete for the construction permit (CP) stage, leading to an efficient review.

2. The GVH Safety Strategy's layered Defense Line methodology is a beneficial framework for ensuring the fundamental safety functions are met and the integrity of radionuclide release barriers is maintained given the explicit and traceable linkage of safety functions, classification of systems, structures and components (SSC), deterministic analyses,

[1] Section 182b of the Atomic Energy Act (AEA) states, in part, "The Advisory Committee on Reactor Safeguards shall review each application under section 103 or section 104 b. for a construction permit...and shall submit a report thereon which shall be made part of the record of the application and available to the public except to the extent that security classification prevents disclosure." The CRN-1 CPA was submitted under section 103 of the AEA.

probabilistic risk assessment (PRA) insights, risk-informed special treatments, and defense in depth.

3. Timely reconciliation of the GVH Safety Strategy with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 requirements is important to support the stated safety case. We encourage NRC staff to capture lessons learned from this review for efficiently reconciling such Defense-Line type approaches with prescriptive light water reactor regulatory requirements for future applications.

4. Important concerns to be resolved at the operating license (OL) stage and during initial startup include: operational testing to confirm natural circulation stability especially at startup and low power, confirmation of isolation condenser system (ICS) performance and associated control and interlock functions, and design extension conditions and practical elimination claims (beyond design basis events).

5. Our review supports issuance of the construction permit (CP) for CRN-1.

BACKGROUND

The BWRX-300 Clinch River design is a nominal 300 MWe (870 MWth) BWR with core heat removal by natural circulation. The BWRX-300 is designed to enhance safety through simplification and reduced dependence on human intervention by reliance on natural circulation and use of passive safety systems. The application is being made under 10 CFR Part 50 for a site that was previously granted an Early Site Permit.

The plant design as reflected in the PSAR was developed under the GVH Safety Strategy licensing framework, documented in licensing topical report (LTR) NEDC-33934 (hereinafter Safety Strategy LTR). This strategy is described in the LTR as "an innovative approach to develop and ensure that the design meets high levels of safety through the establishment of design rules based on defense in depth concepts consistent with International Atomic Energy Agency (IAEA) Specific Safety Requirements SSR-2/1, "Safety of Nuclear Power Plants: Design," Revision 1." The GVH Safety Strategy is generally a traditional deterministic safety analysis framework. It is informed early and iteratively by PRA insights, and it implements Defense Lines, coordinated SSC classification, and traceable safety requirements. The clear intent is development of a robust safety case with results that may be appropriately credited within the context of a licensing application to demonstrate compliance with the specific requirements of a range of regulatory jurisdictions.

DISCUSSION

Although BWR technology is well established within the 10 CFR Part 50 licensing framework and the legacy fleet, there are design, construction, and operational elements of the BWRX-300 plant that are unique, novel, or noteworthy (UNN) and, thus, were particular focus areas within our review.

<u>Approach to Review</u>:

Our review of the safety aspects of the CPA focused on the three fundamental safety functions, the treatment of defense in depth measures, and the GVH Safety Strategy framework as implemented within the design, which collectively define the safety case for the plant. Our UNN focus considered the following aspects of these fundamental safety functions:

- Control of Reactivity: How does the design address the role of natural circulation flow on reactor stability and the reliability and redundancy aspects of the reactor control and scram systems?

- Control of Heat Removal: How are reactor isolation, the various passive means of decay heat removal, and their assumed reliability validated?

- Retention of Radionuclides: Does the novel approach to reactor overpressure protection and use of isolation valves adequately provide both the emergency core cooling and radionuclide retention functions?

The GVH Safety Strategy framework is an alternative approach to past practice under NRC regulations for design, SSC classification, and defense in depth, and is integral to adequacy of the overall safety case. Therefore, our review considered whether the Safety Strategy LTR methodology, together with the design-specific analyses and commitments in the CPA, provides sufficient justification to issue a CP.

Control of Reactivity:

The GVH design reflects a bias toward use of known, proven technologies and physical components where practical. Control blades and elements of the scram and control rod drive (CRD) systems for reactivity control are based on designs currently employed in the Advanced Boiling Water Reactor (ABWR). Although the planned combination of hydraulic scram with a "run-in" function by the fine motion control rod drives does involve new elements, those new elements represent limited-scope evolutionary departures from the ABWR design. The design also includes a boron injection system to provide a defense in depth measure to maintain subcritical conditions under cold conditions, even if all control rods are fully withdrawn. These two independent shutdown systems satisfy the requirements of principal design criterion (PDC) 26.

Application of the Defense Lines approach to system design results in significant defense in depth for the reactor protection system function, with three independent, diverse instrumentation and control systems, each capable of initiating rod insertion when required to maintain plant conditions within safety limits.

Operationally, the reactor core power-to-flow relationship for BWRs is well understood, with extensive data for the forced flow operating fleet. Significantly less operational data are available from the small number of historical natural circulation BWRs. Likewise, reactor stability for natural circulation reactors in startup, shutdown, normal, and off-normal operation has been well-studied through reasonable analyses, but the limited relevant operational data represents a validation data gap. Therefore, startup, shutdown, normal, and off-normal stability analyses will require confirmation through appropriate lead-plant testing.

Nevertheless, we find the control of reactivity basis sufficient for the CP stage because the design combines proven BWR reactivity-control hardware, diverse protection-system actuation, and a natural-circulation stability basis that can be confirmed during startup and early power operations.

Control of Heat Removal:

The safety-related, emergency core cooling system (ECCS) function (performed by the reactor isolation valves (RIVs) together with the ICS) in the GVH design, including the ultimate heat sink, is located entirely within the reactor building. In event scenarios when no containment vessel isolation is initiated, heat rejection up to 25% full power can be accommodated by steaming to the condenser through the turbine bypass system. Loss-of-coolant accident events are managed through an inventory retention strategy whereby the reactor is isolated via single bodied redundant RIVs closely coupled to the reactor nozzles, actuated with diverse trip signal paths, and (with one exception discussed below under “Retention of Radionuclides”) failing safe (closed) on loss of power. However, breaks with an equivalent diameter smaller than 0.75” are not automatically isolated. If such a break were to occur, loss of reactor coolant system (RCS) inventory would be managed via cooldown and depressurization of the RCS by ICS leading to pressure equalization of the RCS with containment. Additionally, RCS makeup can be provided by the nonsafety-related CRD hydraulic subsystem, if available.

The three-train ICS provides decay heat removal and overpressure protection of the isolated reactor vessel via natural circulation heat rejection to dedicated pools within the reactor building. Always on standby, this system is actuated via dual parallel condensate return valves with diverse trip signal paths, and these valves fail safe (open) on loss of power. The ICS also incorporates a passive hydrogen recombiner subsystem for radiolytic gas management, a feature important to preserving natural-circulation condensation performance. Because this subsystem is somewhat complex, ongoing testing will be important to confirm the design basis for the OL application.

Application of the BWRX-300 Defense Lines safety strategy requires consideration that any one system, no matter how reliable, might fail due to an undefined cause. For example, the unintended loss of function of all three ICS trains due to a common cause failure is highly unlikely due to (a) the fail-as-is function defined for the ICS RIVs, (b) the physical interlock features to prevent isolation of more than one train, (c) trains split between two different Safety Class 1 control systems, and (d) the redundancy of the ICS.[2] However, as demonstrated during the Fukushima Daichi Unit 1 core damage progression, an unexamined complex interaction within system elements can lead to a common cause failure and unintended loss of all ICS function.[3] In our subcommittee meeting on April 7, 2026, TVA stated that the OL application will describe features that provide defense in depth to adequately control overpressure and remove decay heat on complete failure of the ICS function.

Heating within the containment vessel during any event is managed by the passive containment cooling system rejecting heat to separate pools within the reactor building. This system is entirely passive and is always in operation with no actuation required beyond the presence of a heat load sufficient to drive natural circulation flow.

We find the control of heat removal basis sufficient for the CP stage due to its reliance on passive functions and minimal active components.

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[2] The physical interlock feature is not described in the PSAR. The applicant discussed this feature in our April 2026 subcommittee meeting (ADAMS Accession No. ML26118B779) and stated that it will be described in the Final Safety Analysis Report.
[3] National Academy of Science, “Lessons Learned from the Fukushima Nuclear Accident for Improving Safety of U.S. Nuclear Plants (2014),” Section 4.3.1 and Sidebar 5.3 (https://www.nationalacademies.org/projects/DELS-NRSB-12-01/publication/18294)

Retention of Radionuclides:

The application credits the use of BWR fuel with a well-established operating history and known reliability to minimize fuel failures in-service.

The materials of construction throughout the reactor coolant pressure boundary (RCPB) have been proven through extensive operating history under relevant operating conditions. Likewise, appropriate materials management processes for the RCPB, including in-service inspection, are well-established from the legacy fleet operating history. Additionally, the applicant has committed to demonstrate that a break between the reactor pressure vessel (RPV) and the RIVs is sufficiently improbable and that core cooling in the unlikely case of such an event is adequate to satisfy 10 CFR 50.46(b) criteria and regulatory public dose limits.

RCPB overpressure protection does not employ safety valves and is instead provided via the ICS system. The ICS system RIVs (on the steam and condensate return lines) are open during normal plant operation, when the ICS system is in standby mode, to preserve its ECCS and overpressure protection functions. They are the only valves serving a reactor or containment isolation function that do not automatically close on loss of power. However, as an active function to manage a breach in a single ICS train, they have an alternate closed “safe” position and are commanded to shut using train-specific leakage monitoring, with appropriate mechanical interlock features planned to prevent isolation of the remaining two ICS trains. Therefore, the ICS RIVs “fail-as-is” to preserve their safe state. Leakage monitoring and interlock features will be further described in the OL application.

The plant employs a dry, nitrogen-inerted containment vessel fully within a safety-related, seismic category 1 reactor building. The reactor building is a deeply embedded structure, and both the containment vessel and reactor building are designed and constructed using steel plate concrete composite elements. A novel element of the design is the first-time use of diaphragm plate steel-plate composite (DP-SC) construction. This aspect of design warrants additional review when sufficient information is available at the OL stage.

The containment vessel design includes an overpressure vent flow path consisting of a rupture disc, a remote-actuated bypass valve, isolation valve, check valve, sparger, and associated piping. This subsystem is used in severe accident cases when containment failure by overpressure is threatened. Upon activation either by the rupture disc or bypass valve, the effluent is scrubbed via the sparger located in the Filter/Storage pool within the reactor building and this enclosed pool is then vented to atmosphere.

Reliable fuel performance, robust RCPB integrity, rapid inventory retention by isolation, passive containment heat removal, and filtered/scrubbed pressure relief for severe-accident conditions, when taken together, provide a layered radionuclide-retention strategy, sufficient for the CP stage.

Adequacy of the overall safety case:

Application of GVH Safety Strategy

Implementation of the Safety Strategy approach is an important element of the safety case. This approach establishes design rules based on defense in depth concepts consistent with IAEA guidance. The stated objective of the BWRX-300 Safety Strategy is:

". . . to organize both the safety design bases and the various safety analyses (deterministic and probabilistic) within a common framework based on [defense in depth] principles to enable a systematic demonstration of a comprehensive safety case. This is accomplished, in part, by applying a design approach based on [defense in depth] concepts and consisting of five levels of defense called Defense Lines (DLs)."

The Staff review of the Safety Strategy LTR is ongoing. Staff determined the completion of the review was not essential for the CPA safety evaluation. Our review of the BWRX-300 design for the Clinch River CPA has been conducted in full consideration of its development within the GVH Safety Strategy framework.

One notable issue in reconciling the Safety Strategy with existing approaches to meet 10 CFR Part 50 regulatory requirements is discussed extensively in Chapter 15 of the CPA safety evaluation. The Staff observes that the Safety Strategy LTR takes a different approach for mitigation of the more frequent upset conditions that are categorized as anticipated operational occurrences (AOOs) than previously applied to meet the corresponding AOO regulatory requirements in 10 CFR Part 50. This issue does not impact the CP application and will be resolved as part of the OL application.

This issue also highlights a potentially broader concern where a comprehensive alternative methodology (such as the GVH Safety Strategy) for rigorously developing and defending the plant safety case conflicts with prescriptive requirements in the applicable regulations or their past interpretations. In this regard, we offer the following observations:

- We consider implementation of the Safety Strategy LTR methodology at this stage of licensing to be acceptable and consistent with the Commission's risk-informed, performance-based policy for advanced reactors.

- We continue to regard the application of a layered Defense Line approach in an iterative manner as the plant evolves through the design and licensing life cycle to be beneficial in obtaining a high degree of consistency, engineering rigor, and transparency in demonstrating that the fundamental safety functions are met and the integrity of radionuclide release barriers is maintained.

- A Defense Lines approach, when demonstrated to be appropriately comprehensive and conservative, can be applied to ensure appropriate independence and diversity in system design, achieve higher reliability and safety, and demonstrate adequacy of defense in depth.

- We encourage adoption of reasonable regulatory flexibility and practical guidance to facilitate generic acceptance of such alternate approaches that exhibit a unified structure.

Safety Analysis, Uncertainties and Margins

The Committee considered whether the CPA safety analyses provide sufficient treatment of uncertainties and margin for this stage of licensing. The GVH Safety Strategy treats safety analysis as an iterative process in which deterministic and probabilistic analyses inform design decisions, establish design-to-analysis requirements, and confirm that the design meets defined acceptance criteria with appropriate margin. Analyses using nominal parameters may warrant additional review if margin is affected. The safety analysis framework also includes consideration of design extension conditions and practical elimination approaches to address beyond design basis events and residual risk in a methodical systematic manner. We find the applicant has provided a sufficiently complete Chapter 15, “Safety Analyses” for the CPA, while recognizing that uncertainty treatment and demonstration of adequate margin across the spectrum of design analyses will necessarily evolve in the development of the final safety analysis report.

SUMMARY

Early pre-application engagement with topical reports was noted to be beneficial for both the CP and OL application reviews in our March 19, 2026, lessons learned letter report. For the Clinch River CPA, the applicant included detailed technical content in the application and submitted topical reports relevant to the safety case for early review, both of which supported an efficient review. This application reflects safety enhancements associated with the reactor design that are significant when compared to prior BWR designs and support simplifications in the overall plant design and analysis. We look forward to reviewing the final safety case when it is submitted with an application for an OL.

Our review supports issuance of the construction permit for CRN-1.

We are not requesting a formal response from the staff to this letter.

Sincerely,

Signed by Halnon, Gregory on 06/20/26

Gregory H. Halnon
Chairman ACRS

Enclosure
List of Acronyms

REFERENCES

1. Advisory Committee on Reactor Safeguards (ACRS), “Early Site Permit – Clinch River Nuclear Site,” January 9, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19009A286)

2. U.S. Nuclear Regulatory Commission, “Final Safety Evaluation Report for the Early Site Permit Application for the Clinch River Nuclear Site,” June 14, 2019 (ADAMS Accession No. ML19162A157)

3. U.S. Nuclear Regulatory Commission, “Tennessee Valley Authority Clinch River Construction Permit Application Advanced Safety Evaluation Report,” December 29, 2025 (ADAMS Package No. ML25363A124, Non-Public)

4. U.S. Nuclear Regulatory Commission, “Safety Evaluation for Licensing Topical Report, NEDC-34270P/NEDO-34270, ‘BWRX-300 Stability Analysis,’ Revision 1,” May 15, 2026 (ADAMS Accession No. ML26078A299)

5. GE Hitachi Nuclear Energy, Licensing Topical Report, NEDO-34270, Revision 0, “BWRX-300 Stability Analysis,” March 31, 2025 (ADAMS Accession No. ML25090A107)

6. GE Hitachi Nuclear Energy, “Licensing Topical Report, BWRX-300 Safety Strategy,” NEDO-33934, Revision 2, March 2026 (ADAMS Accession No. ML26077A383)

7. Tennessee Valley Authority, “Submittal of the Preliminary Safety Analysis Report and Exemptions and Variances in Support of the Clinch River Nuclear Site Construction Permit Application,” May 20, 2025 (ADAMS Package No. ML25140A062)

8. GE Hitachi Nuclear Energy, “Licensing Topical Report, NEDO–33910-A, Revision 2, BWRX–300, Reactor Pressure Vessel Isolation and Overpressure Protection,” June 2021 (ADAMS Accession No. ML23167A086)

9. ACRS, “Safety Evaluation for Topical Report NEDC-33910P, ‘BWRX-300 Reactor Pressure Vessel RPV Isolation and Overpressure Protection’,” October 5, 2020 (ADAMS Accession No. ML20268B242

10. ACRS, “Lessons Learned from ACRS Reviews of New Reactor Applications,” March 19, 2026 (ADAMS Accession No. ML26069A573)

11. GE Hitachi Nuclear Energy, “Licensing Topical Report, NEDO-33911-A, Revision 3, BWRX–300, Containment Performance,” January 2022 (ADAMS Accession No. ML22007A024)

12. ACRS, “Safety Evaluation for Topical Report NEDC-33911P, ‘BWRX-300 Containment Performance’,” March 1, 2021 (ADAMS Accession No. ML21049A340)

13. GE Hitachi Nuclear Energy, “Licensing Topical Report, NEDO-33912-A Revision 2, BWRX–300, Reactivity Control,” February 2023 (ADAMS Accession No. ML23048A018)

14. ACRS, “Safety Evaluation for Topical Report NEDC-33912P, ‘BWRX-300 Reactivity Control’,” December 18, 2020 (ADAMS Accession No. ML20351A431)

15. GE Hitachi Nuclear Energy, “Licensing Topical Report, NEDC–33922-A, Revision 3, BWRX–300, Containment Evaluation Method (GOTHIC Application to BWRX–300)," June 2022 (ADAMS Accession No. ML22168A014)

16. ACRS, “Safety Evaluation for Topical Report NEDC-33922P, ‘BWRX-300 Containment Evaluation Method,’ Revision 2,” April 21, 2022 (ADAMS Accession No. ML22101A298)

17. GE Hitachi Nuclear Energy, “Licensing Topical Report, NEDO-33914-A, Revision 2, BWRX-300, Advanced Civil Construction and Design Approach Licensing Topical Report,” June 2022 (ADAMS Accession No. ML22168A010)

18. ACRS, "Safety Evaluation for Topical Report NEDO-33914, Revision 1, 'BWRX-300 Advanced Civil Construction and Design Approach'," April 21, 2022 (ADAMS Accession No. ML22105A106)

19. GE Hitachi Nuclear Energy, "Licensing Topical Report, BWRX-300 White Paper NEDO-33988, Revision 0, BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design," October 14, 2022 (ADAMS Accession No. ML22287A177)

20. GE Hitachi Nuclear Energy, "Licensing Topical Report, NEDO-33926, Revision 4, BWRX-300, Steel-Plate Composite Containment Vessel (SCCV) and Reactor Building (RB) Structural Design," December 2025 (ADAMS Accession No. ML25351A088)

List of Acronyms

10 CFR	Title 10 of the *Code of Federal Regulations*
ACRS	Advisory Committee on Reactor Safeguards
AEA	Atomic Energy Act
AOO	Anticipated Operational Occurrence
ABWR	Advanced Boiling Water Reactor
ADAMS	Agencywide Documents Access and Management System
BWR	Boiling Water Reactor
CPA	Construction Permit Application
CP	Construction Permit
CRD	Control Rod Drive
CRN-1	Clinch River Nuclear Unit 1
DLs	Defense Lines
DP-SC	Diaphragm Plate Steel-Plate Composite
ECCS	Emergency Core Cooling System
GVH	General Electric Vernova Hitachi
IAEA	International Atomic Energy Agency
ICS	Isolation Condenser System
LTR	Licensing Topical Report
MWe	Megawatts Electric
MWth	Megawatts Thermal
NRC	U.S. Nuclear Regulatory Commission
OL	Operating License
PDC	Principal Design Criterion
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
SER	Safety Evaluation Report
SSC	Systems, Structures and Components
TVA	Tennessee Valley Authority
UNN	Unique, Novel, or Noteworthy