ENCLOSURE 1

M210088

Response to Request for Additional Information eRAI 9849

Licensing Topical Report
NEDO-33914, Revision 0,
BWRX-300 Advanced Civil Construction and Design Approach

Non-Proprietary Information

SRP-Review Section: 01.05 - Other Regulatory Considerations
LTR Application Section: TR NEDO-33914 Sections 1.3 and 6.1.2

01.05-01 (eRAI 9849)
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Requirement

General Design Criterion (GDC) 2 requires that structures, systems, and components (SSCs) important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety function. GDC 2 also specifies that the design bases for these SSCs shall reflect the importance of the safety functions to be performed.

Regulatory Guide (RG) 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed In Light-Water-Cooled Nuclear Power Plants," provides the guidance to licensees and applicants in the design, construction, installation, and testing the SSCs of radioactive waste management facilities in light-water-reactor nuclear power plants.

Issue

Sections 1.3 and 6.1 of GE-Hitachi Nuclear Energy Americas, LLC (GEH) Pre-Application Submittal of NEDO-33914, BWRX-300 Advanced Civil Construction and Design Approach Licensing Topical Report (LTR), submitted to the NRC on January 20, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21020A136) state that the portions of the turbine building structure and foundation that support and enclose the main steam piping are designed as RG 1.143, class RW-IIa. They also state that RG 1.143 is used because the building contain SSCs used for management and containment of highly radioactive gas, liquid, and solid materials whose failure, considering the maximum inventory, would result in a potential unmitigated radiological release levels that may be higher than those specified in RG 1.143, Section 5.1.

RG 1.143 provides guidance for the classification and design of radwaste management systems and steam generator blowdown systems. RG 1.143 does not provide guidance for the classification or design of the main steam piping or surrounding structures. While the offgas system is used for management of radioactive gas, other SSCs in the turbine building, like the main steam piping and the main condenser are credited for main steam line fission product holdup and retention in the analysis of design-basis accident radiological consequences for boiling water reactor plants with no main steam isolation valve leakage control system. In this way, the main steam piping and condenser are used to mitigate the consequences of an accident. Appendix A to 10 CFR Part 100 requires that SSCs necessary to ensure the capability to mitigate the consequences of accidents remain functional during and after a safe-shutdown earthquake (SSE).

RG 1.143 seismic classification of RW-IIa, specifies ½ (SSE) as the earthquake design criteria for radwaste management SSCs. If the ½ SSE design requirement is applied to the condenser and portions of the main steam piping in the turbine building, the capability for those systems to mitigate the consequences of accidents and remain functional during and after an SSE would not be ensured.

Request

The staff requests GEH to clearly identify the applicability of RG 1.143 to the turbine building design. The response should also address any limitations on the applicability of RG 1.143 and clarify how the design will ensure SSCs in the turbine building that are used to mitigate the consequences of an accident be designed to remain functional during and after an SSE.

GEH Response to NRC Question 01.05-01

In the initial submittal of NEDO-33914, the decision to designate this portion of the turbine building (TB) as RW-IIa was associated with the offgas system (OGS) charcoal adsorbers that were located in the same area as the main steam piping. The OGS charcoal adsorbers are used for management of radiological gases are designated as RW-IIa, as required by the guidance in RG 1.143, C. Regulatory Position, 5. Classification of Radwaste Systems for Design Purposes, and 6. Natural Phenomena and Man-Induced Hazards Design for Radwaste Management Systems and Structures.

GEH has elected to move the OGS charcoal adsorbers to the radioactive waste building (RwB), thereby eliminating the need for the associated portion of the TB to comply with RG 1.143 "Design Guidance For Radioactive Waste Management Systems, Structures, And Components Installed in Light-Water-Cooled Nuclear Power Plants," and to be qualified as RW-IIa structure one-half safe shutdown earthquake (SSE). As a result, Sections 1.3, 2.4, 6.1 and 6.4 of licensing topical report (LTR) NEDO-33914 "BWRX-300 Advanced Civil Construction and Design Approach" are being revised to reflect the relocation of the OGS charcoal adsorbers to the RwB. The remaining equipment in the TB associated with the OGS such as the offgas cooler and refrigerant dryer are processing equipment that do not hold up or accumulate gaseous effluents, and are not required to meet the classification of RG 1.143 RW-IIa or RW-IIb classification. Therefore, the entire TB is now classified as non-seismic as described in LTR NEDO-33914, Sections 6.1 and 6.4. The II/I seismic interaction evaluation of the TB is described in LTR NEDO-33914, Section 6.2.

Regarding the second response request to clarify how the SSCs in the turbine building that are used to mitigate the consequences of an accident be designed to remain functional during and after an SSE, GEH offers the following response:

Like the Advanced Boiling Water Reactor (ABWR) and Economic Simplified Boiling Water Reactor (ESBWR), the BWRX-300 design safety analyses confirm that there are no design basis accidents (DBAs) that result in fuel damage. As a result, the main steam line and condenser are not credited for fission product holdup or retention of any radiological consequences.

The BWRX-300 has further refined the barrier design compared to the ABWR and the ESBWR by incorporating the use of reactor pressure vessel isolation valves attached directly to the reactor pressure vessel (RPV). These RPV isolation valves also function as inside containment isolation valves (CIVs), along with one outboard main steam isolation valve (MSIV) that are used to preserve reactor coolant system inventory for large and medium pipe break loss-of-coolant-accidents (LOCAs) and to prevent releases from containment. This configuration was approved for use in accordance with 10 CFR 50, Appendix A, General Design Considerations (GDC), 55 Reactor coolant pressure boundary penetrating containment, in the final safety evaluation for GEH LTR NEDC-33911P, Revision 0, Supplement 1, "BWRX-300 Containment Performance" dated March 12, 2021, Section 5.1.22, "10 CFR Part 50, Appendix A, GDC 55". NEDC-33911P-A, Revision 1, Section 2.2.7.1, states that the automatic CIVs outside containment are not required to be fast closing because there is no credible scenario in which fission products can be released to the containment within a few hours of a DBA. As a result, there are no credible DBAs where postulated fission product releases greater than those contained in normal reactor coolant could occur for small-, medium-, or large-break LOCAs. CIV closure times will be established based upon source term evaluations for postulated beyond design basis accident fission product releases, and closure times will be ensured by compliance to the limiting conditions for operation in the BWRX-300 technical specifications. In the final safety evaluation for NEDC-33911P-A, Section 5.3.12, the NRC staff agreed with GEH's statement in Section 2.2.7.1 by stating: "the NRC staff finds the above clarification on evaluating outboard CIV closure time acceptable, because valve closure time is to be established based on fission product release and source term evaluation, consistent with the guidance in SRP Section 6.2.4, Paragraph I, Item 1.E regarding the basis for selection of closure times of isolation valves."

NEDC-33911P-A, Revision 1, Section 2.2.7, states that leak-tightness of the CIVs is verified by 10 CFR Part 50, Appendix J "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors", Type C tests, and the leak-tightness of containment is verified by Type A testing. In NEDC-33911P-A, Revision 1, Section 5.2.7, GEH states that the design will include a containment leak test program that addresses integrated containment leakage rate (Type A tests), containment penetration leakage test (Type B tests) and CIV leakage rates (Type C tests) that complies with 10 CFR Part 50, Appendix J. Type A, B, and C tests are performed before operations and periodically thereafter to assure that the leakage rates through containment and through systems or components that penetrate containment do not exceed their maximum allowable rates. Maintenance of containment, including repairs on systems and components penetrating containment is performed as necessary to maintain leakage rates at or below acceptable values.

In the final safety evaluation for NEDC-33911P-A, Section 5.1.26, the NRC staff concluded: "The NRC staff finds the BWRX-300 design, as described in NEDC-33911P, would accommodate periodic integrated leakage rate testing and local leak rate test for CIVs, and containment penetrations; thus, the NRC staff finds that the design is consistent with 10 CFR Part 50, Appendix J and, therefore, is acceptable."

In the final safety evaluation for GEH LTR NEDC-33910P, Revision 0, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection" dated November 18, 2020, Section 2.2, the

NRC staff agreed with the robustness of the RPV isolation valve concept: “The NRC staff finds that, based on the description in NEDC-33910, together with its reference to NEDC-33911, the RPV isolation valve concept is consistent with 10 CFR Part 50, Appendix A, GDC 30, “Quality of reactor coolant pressure boundary,” and GDC 35 and is therefore acceptable.” Further in the final safety evaluation for GEH LTR NEDC-33910P, Revision 0, Section 2.4.1, “Connection of the Reactor Pressure Vessel Isolation Valves to Reactor Vessel,” the NRC staff further agreed with the robustness of the RPV isolations where they stated: “details of the bolted connections including threaded fastener design, leakage detection systems design, and ISI requirements, will demonstrate that the probability of gross rupture is extremely low. Section 2.4.1 specifies that these design and ISI requirements will provide assurance that potential failure mechanisms are detected before the onset of a catastrophic failure involving the fasteners of the bolted flange connections for the RPV isolation valves. Therefore, Section 2.4.1 asserts that a break at these locations need not be postulated.”

Section 2.7 of NEDC-33910P-A, Revision 2, states that two categories of break sizes are specified in the topical report that differentiates between large pipes with isolation valves and smaller pipes without isolation valves (e.g., the differential pressure instrument lines). The key concept related to meeting the design requirements is that larger isolation lines are rapidly isolated to minimize inventory loss in conjunction with cooling provided by the isolation condenser system to maintain reactor water level at or above the top of active fuel (TAF) or fuel cladding temperature within normal operating temperature range for at least 72 hours. For smaller lines that can remain unisolated, the differential pressure between the RPV and the containment is minimized such that inventory loss is reduced in the absence of injection or makeup flow and core cooling is provided by the isolation condenser system (ICS) such that the reactor water level is maintained at or above the TAF or the fuel cladding temperature is maintained within normal operating range for at least 72 hours. In the final safety evaluation for GEH LTR NEDC-33910P, Revision 0, Section 2.7 Categories of Pipe Breaks, the NRC staff agreed with GEH’s conclusion regarding pipe breaks: “The NRC staff finds that the categories of pipe breaks and the associated design requirements for the BWRX-300, as described in NEDC-33910, are consistent with 10 CFR 50.46(b).”

GEH has not determined that credit for holdup or retention of radionuclides in the main steam lines or the condenser for mitigating the consequences of beyond design basis accidents (BDBAs)/severe accidents (SAs) is necessary; however, based upon the robustness of the BWRX-300 RPV isolation valves and the outboard MSIV configuration that rely upon Class 1E battery-backed direct current power for automatic actuation and fail-safe design, and the stringent Appendix J leakage testing program, GEH’s current analysis has not found the necessity of this credit. Therefore, the TB does not need to be designed to meet more stringent seismic requirements as the main steam lines and the condenser have not been credited for radiological consequences mitigation.

Additionally, GEH notes that the NRC staff has concluded that the design of the non-seismic BWR turbine buildings are robust as discussed in DRA-ISG-2021-XX, “Supplemental Guidance for Radiological Consequences Analyses Using Alternative Source Terms Draft Interim Staff Guidance,” May 2021. In DRA-ISG-2021-XX, the NRC staff discusses the approved alternate pathway of the main steam piping downstream of the outboard MSIV and the main condenser that was justified by the design information and data collected in GE topical report NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," September 1993, ADAMS Accession No. ML010640286, and approved by U.S. Nuclear Regulatory Commission, "Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, 'BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems,' September 1993," March 3, 1999, ADAMS Accession No. ML010640286. The NRC staff notes that GE has demonstrated that the BWR turbine building is "seismically robust" at the plant's SSE and the alternate pathway for crediting holdup and retention in the main steam line from the outboard MSIV is acceptable. The NRC staff notes: "This ISG provides supplemental guidance to items III.6.c and IV.5 in SRP Section 15.0.1. The basis for the supplemental guidance is a technical assessment that uses knowledge and operating experience related to the power conversion system (PCS), including information on the seismic capacity and risk at nuclear power plants. The staff, through this ISG, should acknowledge the presence of the PCS and its ability to provide a large holdup and retention volume for MSIV leakage when staff determines that the requirements of the regulations are satisfied and the method of analysis conforms with accepted practices, but uncertainties remain in input parameters used in the deterministic dose calculations. In doing so, the staff should recognize that there is a high probability that doses will be lower than those estimated using deterministic methods that include accepted assumptions but do not credit holdup and retention of the MSIV leakage within the PCS." This assessment uses engineering information, such as operations and design knowledge, and probabilistic and risk information on the seismic capacity (i.e., the ability of SSCs) to withstand acceleration induced by a seismic event) of the SSCs in the realistic transport pathway to determine the risk of unavailability of the SSCs in the PCS pathway for fission product holdup and retention. The safety evaluation on NEDC-31858P, Revision 2, gives precedent for not relying on only safety-related or seismic Category I SSCs for mitigating the radiological consequences of a postulated release. That safety evaluation states that requiring the non-seismically analyzed portions of the main steam system piping and components to meet seismic Category I requirements is impractical because the modifications required to upgrade the system to seismic Category I requirements would be very costly. This interim guidance concludes that BWR turbine buildings are "seismically robust" to SSEs and further substantiates that the BWRX-300 TB does not require more stringent SSE compliance.

Proposed Changes to NEDO-33914 Revision 0

NEDO-33914, Sections 1.3, 2.4, 6.1, 6.1.2 and 6.4 will be revised to reflect the relocation of the OGS charcoal adsorbers to the RwB. The RwB structure and foundation basis remains the same.

1.3 Description of the BWRX-300

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TB encloses the turbine generator, main condenser, condensate, and feedwater systems, condensate purification system, off-gas system (OGS) <u>cooler and refrigerant dryer</u>, turbine-generator support systems and bridge crane.

The RwB, which houses the systems for management of radioactive gas, liquid, and solid radiological waste is categorized as RW-IIa in accordance with Regulatory Guide (RG) 1.143 "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Revision 2. ~~The portions of the TB structure and foundation that support and enclose the main steam piping and the OGS for management of radiological gases are also designed as Rw-IIa following the provisions of RG 1.143.~~

2.4 II/I Interaction Regulations

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- ~~The TB structure does not collapse to result in impairment of safety functions of the main steam piping or the OGS.~~

6.1 Control Building, Turbine Building and Radwaste Building Design Bases

CB, TB and RwB structures and foundations are designed in accordance with their seismic classification:

- Non Seismic Category for the CB and TB structures ~~excluding the portion of TB enclosing the main steam piping and offgas system (OGS)~~; and
- RW-IIa Category for the RwB structure ~~and the portion of TB structure enclosing the main steam piping and OGS~~.

6.1.2 Radwaste Category IIa Building Structure and Foundations Design Bases

RwB ~~and the portion of the TB enclosing the main steam piping and OGS~~ is classified as a RW-IIa structure~~s~~ because it contains SSCs used for managing and containment of highly radioactive gas, liquid, and solid materials whose failure, considering the maximum inventory, would result in a potential unmitigated radiological release levels that may be higher than those specified in RG 1.143, Section 5.1.

In accordance with RG 1.143, Table 1 guidance, the design of the BWRX-300 RwB steel structures follows the provisions of AISC N690 (Reference 8.25). The design of the RwB concrete structures and basemat ~~and the portion of the TB structure enclosing the main steam piping and OGS~~ is in accordance with ACI 349-13 (Reference 8.24). Based on RG 1.143, Table 2, the loads for the design of the RW-IIa RwB structure ~~and TB~~ includes:

6.4 Summary of Design Approach for II/I Interaction

The following aspects of the BWRX 300 graded design approach for II/I interaction of non-SC-I CB, TB and RwB with adjacent SC-I RB presented in this section of the report may be referenced during future licensing activities that are beyond the current guidelines in RG 1.29, “Seismic Design Classification,” Revision 5:

(1) General criteria for design of Non-Seismic CB and TB structures provided in Section 6.1.1, including the requirements for determining seismic and wind design loads.

(2) General criteria for design of RW-IIa, RwB ~~and TB~~ structures provided in Section 6.1.2, including the requirements for determining seismic, wind, tornado wind and missile design loads.