ENCLOSURE 2

M210014

NEDO-33912-A Revision 1

Non-Proprietary Information

IMPORTANT NOTICE

This is a non-proprietary version of NEDC-33912P-A Revision 1, from which the proprietary information has been removed. Portions of the enclosure that have been removed are indicated by an open and closed bracket as shown here. [[ ]].

Note the NRC's Final Safety Evaluation is enclosed in NEDO-33912-A Revision 1. Portions of the Safety Evaluation that have been removed are indicated with double square brackets as shown here. [[ ]].

GE Hitachi Nuclear Energy

NEDO-33912-A
Revision 1
February 2021

Non-Proprietary Information

Licensing Topical Report

BWRX-300 Reactivity Control

GE-Hitachi Nuclear Energy Americas, LLC

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

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

January 12, 2021

Ms. Michelle Catts
Senior Vice President, Nuclear Programs
GE-Hitachi Nuclear Energy Americas, LLC
P.O. Box 780, M/C A-18
Wilmington, NC 28402

SUBJECT: FINAL SAFETY EVALUATION FOR GE-HITACHI LICENSING TOPICAL REPORT NEDC-33912P, REVISION 0, "BWRX-300 REACTIVITY CONTROL"

Dear Ms. Catts:

By letter dated March 31, 2020, GE-Hitachi Nuclear Energy Americas, LLC (GEH), submitted Licensing Topical Report (LTR) NEDC-33912P, Revision 0, "BWRX-300 Reactivity Control" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20092A016), to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval in support of a future licensing application for the GEH small modular reactor (SMR) under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," or Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." By letter dated September 4, 2020 (ADAMS Accession No. ML20248H540), GEH submitted Revision 0, Supplement 1 of the LTR.

The NRC staff has found LTR NEDC-33912P, Revision 0, as updated by the September 4, 2020, supplement, to be acceptable for referencing in licensing applications for the GEH SMR design to the extent specified in the enclosed safety evaluation (SE). The SE defines the basis for acceptance of the LTR.

In accordance with the guidance provided on the NRC's LTR website (http://www.nrc.gov/about-nrc/regulatory/licensing/topical-reports.html), the NRC requests that GEH publish an accepted version of this LTR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed, so information is readily located. Also, it must contain in its appendices historical review information, such as requests for additional information and accepted responses, and original LTR pages that were replaced. The accepted version shall include an "-A" (designated accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusion in this letter (that the LTR is acceptable) is invalidated, GEH and/or an applicant referencing the LTR will be expected to revise and resubmit its respective documentation or submit justification for the continued applicability of the LTR without revision of the respective documentation.

If you have any questions or comments concerning this matter, I can be reached via e-mail at Rani.Franovich@nrc.gov.

Sincerely,

/RA/

Rani Franovich, Senior Project Manager
New Reactor Licensing Branch
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation

Docket No.: 99900003

Enclosure: As stated

cc w/o encl.: GEH BWRX-300 NEDC-33912P ListServ

SUBJECT: FINAL SAFETY EVALUATION FOR GE-HITACHI LICENSING TOPICAL REPORT NEDC-33912P, REVISION 0, “BWRX-300 REACTIVITY CONTROL” DATED: JANUARY 12, 2021

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SE: ML20275A373 (Public)
SE: ML20275A372 (Non-Public) ***via email** **NRR-106**

OFFICE	DNRL/NRLB:PM	DNRL/NRLB:LA	DNRL/NRLB: BC
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DATE	01/05/2021	01/07/2021	01/12/2021

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT NEDC-33912P, REVISION 0

BWRX-300 REACTIVITY CONTROL

GE-HITACHI NUCLEAR ENERGY AMERICAS, LLC

1.0 Introduction

The purpose of GE-Hitachi Nuclear Energy Americas, LLC (GEH), licensing topical report NEDC-33912, Revision 0, "BWRX-300 Reactivity Control," dated March 31, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20092A016), and supplemented September 4, 2020 (ADAMS Accession No. ML20248H540), is to provide the design requirements, acceptance criteria, and regulatory basis for the BWRX-300 reactivity control design functions. Specifically, the report specifies design requirements for the following systems or functions:

- reactor protection system (RPS)
- [[ ]]
- alternate rod insertion (ARI)
- [[ ]]
- rod control system

In this safety evaluation (SE), the U.S. Nuclear Regulatory Commission (NRC) staff describes its review of NEDC-33912 and the acceptability of licensing topical report provisions for reactivity control for the BWRX-300 small modular reactor (SMR). In response to an NRC staff request for additional information, GEH submitted a letter dated August 3, 2020 (ADAMS Accession No. ML20216A748). The NRC staff will evaluate the compliance of the final design of the reactivity control features for the BWRX-300 SMR during future licensing activities in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," or 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants," as applicable. In this SE, double brackets indicate proprietary information.

2.0 Technical Description of Reactivity Control

2.1 General Introduction

Section 2.1, "General Introduction," of NEDC-33912 provides high-level information about the BWRX-300 and reactivity control. The BWRX-300 is a water-cooled, natural circulation-driven SMR with a power output of about 300 megawatts electric and target applications to include baseload and load-following electrical generation. GEH described how the BWRX-300 built upon nine previous generations of the boiling-water reactor (BWR) and evolved from the NRC-licensed economic simplified boiling-water reactor (ESBWR). GEH stated that the

BWRX-300 incorporates design, analysis, and operating experience from the BWR operating fleet, advanced boiling-water reactor (ABWR), and ESBWR and adds design improvements and new defense-in-depth (DID) design features and functions.

2.2 Systems and Components for Control of Reactivity

Section 2.2, "Systems and Components for Control of Reactivity," of NEDC-33912 states that the BWRX-300 relies on control rods and burnable poisons for reactivity control. Control rods are the primary means of achieving shutdown in normal operations, anticipated operational occurrences (AOOs), postulated accidents, beyond-design-basis events, and severe accident scenarios. GEH provided the following design requirements:

- The core design and control rods together provide ample shutdown margin to ensure that the reactor can remain shut down in a cold, xenon-free condition at any time in cycle with the highest worth control rod pair associated with an individual hydraulic control unit (HCU) withdrawn.
- Control rods are positioned in fine increments for normal operation and may be inserted rapidly by multiple means to achieve shutdown.

Section 2.2.1, "Control Rods," and Section 2.2.2, "Control Rod Drives," of NEDC-33912 describe those components for the BWRX-300 and identify associated design requirements. The control rods are designed for significant power changes during reactor startup and shutdown, for normal power changes during operation, and to provide ample shutdown margin. The BWRX-300 fine motion control rod drives (FMCRDs) use two diverse motive forces for positioning the control rods. During normal operation, each control rod is positioned by a non-safety-related electric motor drive. For a rapid shutdown, control rods are inserted via a safety-related hydraulic scram that is initiated by opening the scram valves on each accumulator water discharge path.

The staff notes that the BWRX-300 design also relies on [[ ]] and feedwater level control system to control reactivity and meet pertinent regulatory requirements. Section 4.0 of this SE documents the staff's review of specific regulations associated with reactivity control.

2.3 BWRX-300 Associated Mitigating Systems

2.3.1 Isolation Condenser System

The BWRX-300 ICS is a safety-related, passive system designed to remove heat from, and provide overpressure protection for, the reactor when the normal heat removal system is unavailable due to sudden reactor isolation at power operating conditions, station blackout, anticipated transients without scram (ATWS), and loss-of-coolant accidents. NEDC-33910P, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection" (ADAMS Accession No. ML20174A574), and associated staff SE (ADAMS Accession No. ML20176A446) describe the ICS in detail. Section 2.3, "BWRX-300 Associated Mitigating Systems," of NEDC-33912 states that [[ ]]

resulting from events that require a rapid reactor shutdown if the reactor scram fails or is delayed. Section 2.3.1, “Isolation Condenser System,” of NEDC-33912 states that the [[ ]], which the NRC staff notes would also have the effect of [[ ]].

Sections 3.7.1 and 4.1.1 of this SE provide additional GEH design requirements for the ICS specific to ATWS and the staff’s evaluation regarding the requirements in 10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.”

3.0 Defense-in-Depth of Reactivity Control Functions

3.1 General Overview of Defense-in-Depth Concept

GEH used a plant-level DID concept based on International Atomic Energy Agency (IAEA) Specific Safety Requirements (SSR)-2/1, “Safety of Nuclear Power Plants: Design,” for the BWRX-300 design, including the arrangement of design features and functions into defense lines (DLs) analogous to the levels of defense defined in IAEA SSR-2/1. NEDC-33912 states that the IAEA DID concept defines the design and analysis rules governing that arrangement, such that DLs have good alignment with the safety assessments defined in a BWRX-300 safety assessment framework used to demonstrate plant safety. The DID concept is applied to the BWRX-300 systems and equipment responsible for performing functions assigned to one of five DLs. For each DL, NEDC-33912 provides a brief description and a list of design features or measures associated with reactivity control.

Although NEDC-33912 provides detailed information related to design philosophy and the DL approach adopted for the BWRX-300, the staff does not intend to review or make a determination on the acceptability of IAEA SSR-2/1 as part of this SE. Instead, the staff evaluated and based its findings on the design that resulted from the engineering design process applied to develop the BWRX-300.

Sections 3.2 through 3.6 of this SE summarize the five DLs applied to the BWRX-300 related to reactivity control, as described in NEDC-33912. These sections also identify the resultant BWRX-300 reactivity control design features associated with each defense line.

3.2 Defense Line 1

DL1 minimizes the potential for accidents to occur by applying high quality and conservatism in plant design, construction, operations, and maintenance. DL1 does not include performance of plant functions.

3.3 Defense Line 2

DL2 encompasses plant functions designed to control or respond to initiating events before any plant parameters reach a DL3 actuation setpoint. DL2 functions are not considered

safety-related; however, appropriate quality and reliability measures are applied to ensure functional performance as a DID measure.

DL2 features important for reactivity control include the following:

- [[ ]]
- [[ ]]
- [[ ]]
- normal control of control rod system
- control rod blocks to mitigate incorrect rod withdrawals

NEDC-33912 specifies that the DL2 functions must be performed independently from DL3 and DL4 functions, and any portion of DL2 functions subject to common-cause failure must be performed diversely from corresponding portions of DL3 or DL4 functions.

3.4 Defense Line 3

DL3 contains plant functions that mitigate an initiating event by preventing fuel damage when possible, protecting the integrity of fission product barriers, placing the plant in a safe state, and maintaining the plant in a safe condition following an event until normal operations are resumed. DL3 functions typically include reactor scram and actuation of engineered safety features. DL3 functions are needed when DL2 is not effective. DL3 functions are considered safety-related.

DL3 features important for reactivity control include the following:

- RPS hydraulic scram
- [[ ]]

3.5 Defense Line 4

DL4a functions can place and maintain the plant in a safe state following initiating events with failure of DL3 functions. The DL4a functions are intended to prevent the progression of accidents and radioactive release to the public.

DL4b functions prevent or mitigate a severe accident while maintaining radioactive releases at acceptable levels. DL4b also provides protection for events that exceed DL1 assumptions regarding initiating events as a result of extreme events, multiple events, or multiple failures.

DL4 features important for reactivity control include the following:

- ARI, which provides hydraulic scram in the event of an HCU failure
- electric motor run-in of the FMCRDs

3.6 Defense Line 5

DL5 addresses offsite emergency preparedness to protect the public from substantial radioactive releases. DL5 does not include the performance of plant functions.

3.7 Specific Reactivity Control Events Considered in Defense-in-Depth Concept

Section 3.7, "Specific Reactivity Control Events Considered in Defense-in-Depth Concept," of NEDC-33912 summarizes how the BWRX-300 reactivity control features affect select reactivity events. The descriptions are not intended to capture all BWRX-300 event sequences.

3.7.1 Anticipated Transient Without Scram

Section 3.7.1, "Anticipated Transient Without Scram (ATWS)," of NEDC-33912 describes diverse scram features of the BWRX-300 that prevent or mitigate ATWS events. In addition to the safety-related RPS, the BWRX-300 design includes [[ ]] to perform a reactor shutdown. As required by 10 CFR 50.62(c)(3), the BWRX-300 includes an ARI system as a backup means to depressurize the HCU air header in the event the HCUs receive a valid scram signal but fail to insert the control rods. If hydraulic insertion fails, [[ ]]. Because the BWRX-300 uses natural circulation, the typical means to adjust power and flow with recirculation pumps is not applicable. Instead, the BWRX-300 uses the following thermal-hydraulic means of suppressing or limiting power:

- [[ ]]
- [[ ]].

NEDC-33912, Section 3.7.1, also identifies acceptance criteria for evaluating the effectiveness of the diverse reactivity control system shutdown methods and design requirements associated with ATWS prevention and mitigation. The specific design requirements include the following:

- [[
-]]
- The RPS initiates a reactor scram based on signals and setpoints needed to support safety analysis credited trips.
- [[
-
-]]
- The ARI provides a diverse means to actuate the HCUs upon sensing a failure to scram.
- FMCRDs receive an electric motor run-in signal upon sensing a parameter requiring a scram.

- [[ ]]
- FMCRD insertion time limits are established based on meeting the acceptance criteria of the safety analyses.

The NRC staff considers the BWRX-300 to be an evolutionary design. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 15.8, "Anticipated Transients Without Scram," states, "[a]pplicants [for evolutionary designs] must demonstrate that the failure probability of failing the ATWS success criteria is sufficiently small because either: (1) the criteria are met, or (2) a diverse scram system is installed that reduces significantly the probability of a failure to scram."

The NRC staff finds the acceptance criteria in NEDC-33912, Section 3.7.1, for evaluating reactivity control system effectiveness for ATWS events consistent with SRP Section 15.8 item (1) above and, therefore, acceptable. Alternatively, an applicant for a BWRX-300 SMR could use a probabilistic approach for the reliability of the diverse scram system, as described in Section 4.1.1 of this SE, to satisfy SRP Section 15.8 item (2) above. In addition, the NRC staff finds the above design requirements associated with ATWS prevention and mitigation are consistent with the requirements of 10 CFR 50.62. When the NRC receives an application for a BWRX-300 SMR, the staff will perform a detailed evaluation of the analysis that demonstrates the effectiveness of the ATWS mitigation systems described above to confirm that 10 CFR 50.62 is met or that an appropriate justification for an exemption is included. Section 4.1.1 of this SE provides a detailed regulatory assessment of 10 CFR 50.62.

3.7.2 Control Rod Drop Accident

Section 3.7.2, "Control Rod Drop Accident," of NEDC-33912 describes the design features of the BWRX-300 that prevent and mitigate a control rod drop accident (CRDA). Consistent with the NRC-approved ESBWR design, the BWRX-300 uses FMCRDs that use a bayonet-style coupling, which is a different design than that used in the operating fleet. This coupling requires a 45-degree rotation to uncouple, which is physically prevented because the fuel assemblies in the core constrain the cruciform control blade on all sides. In addition, the BWRX-300 employs dual separation detection devices that implement a control rod withdrawal block if they detect separation of the control rod and the drive mechanism.

The NRC staff finds that GEH's design requirements, along with a design-basis safety analysis of the CRDA event for the BWRX-300, as described in Section 4.1.11 of this SE and in conformance with Limitation and Condition 5.2 of this SE, are consistent with the requirements of 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 28, "Reactivity limits," and are therefore acceptable. The NRC staff will perform a detailed evaluation to confirm that the final design features and associated analysis satisfy the regulatory requirements of GDC 28 when the agency receives an application for a BWRX-300 SMR. SE Section 4.1.11 provides a detailed regulatory assessment of GDC 28.

3.7.3 Rod Withdrawal Error

Section 3.7.3, “Rod Withdrawal Error,” of NEDC-33912 describes the design and mitigating features for ensuring that specified acceptable fuel design limits (SAFDLs) are not exceeded for rod withdrawal error events. The BWRX-300 rod control system employs redundancy to limit the effect of single failures. If a malfunction of the rod control system during operation results in a rod withdrawal error, nuclear instrumentation is used to generate a rod block or reactor scram.

The NRC staff finds that the design requirements of the (1) source range neutron monitors to provide a period-based rod block function during startup, (2) source range neutron monitors to provide a period-based reactor scram signal during startup, and (3) average power range monitors to provide a high-flux reactor scram signal during power operations, as described in NEDC-33912, Section 3.7.3, are consistent with the requirements of GDC 25, “Protection system requirements for reactivity control malfunctions,” and are therefore acceptable. The NRC staff will perform a detailed evaluation, relative to rod withdrawal errors, to confirm GDC 25 is met when the agency receives an application for a BWRX-300 SMR. SE Section 4.1.8 provides a detailed regulatory assessment of GDC 25.

4.0 Regulatory Evaluation

Section 4.0, “Regulatory Evaluation,” of NEDC-33912 provides statements of compliance for the regulations in 10 CFR Part 50 GEH determined to be related to the reactivity control design features of the BWRX-300 SMR and design-specific information associated with pertinent NRC guidance.

NEDC-33912 describes the intent to meet each of the relevant regulatory requirements for the BWRX-300 SMR. In some instances, NEDC-33912 indicates that specific design requirements for the BWRX-300 systems and components will be provided during future licensing activities.

The sections below provide the staff’s evaluation of the preliminary design information related to each regulation. The staff will conduct additional evaluations during future licensing activities.

4.1 10 CFR Part 50 Regulations

This section addresses only those regulations that GEH included in NEDC-33912. When the NRC receives an application for a BWRX-300 SMR, the staff will review the application against all applicable regulatory requirements.

4.1.1 10 CFR 50.62

The regulations in 10 CFR 50.62 address the reduction of risk from ATWS for commercial light-water reactors. In 10 CFR 50.62(c)(3) through (c)(5), the NRC provides requirements specific to BWRs that are discussed below.

10 CFR 50.62(c)(3)

The regulation in 10 CFR 50.62(c)(3) requires that each BWR must have an ARI system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device. Section 2.2.2.2 of NEDC-33912 describes the ARI system for the BWRX-300. The ARI system provides a diverse means of depressurizing the scram air header to ensure that the HCU stored energy is released to cause a reactor scram. NEDC-33912 specifies that the BWRX-300 design will meet the requirements of 10 CFR 50.62(c)(3). The staff finds the approach described in NEDC-33912 consistent with 10 CFR 50.62(c)(3) and, therefore, acceptable. The staff will conduct a detailed evaluation to confirm that 10 CFR 50.62(c)(3) is met when the NRC receives an application for a BWRX-300 SMR.

10 CFR 50.62(c)(4)

The regulation in 10 CFR 50.62(c)(4) requires that each BWR must have a standby liquid control system (SLCS), capable of injecting a highly borated water solution into the RPV. The SLCS initiation must be automatic and designed to perform its function in a reliable manner.

Section 4.1.1, “10 CFR 50.62,” of NEDC-33912 states that the BWRX-300 includes a [[ ]]. For example, the reactor can be shut down by using the FMCRD electric motor run-in function. [[ ]]. GEH concluded that [[ ]].

According to GEH, the NRC outlined the basis for ATWS rule requirements in SECY-83-293, “Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events,” dated July 19, 1983, which concluded that additional ATWS safety requirements were justified and included the stipulation to reduce the risk of core damage because of ATWS to be less than $1x10^{-5}$ per reactor year. NUREG-1780, “Regulatory Effectiveness of the Anticipated Transient Without Scram Rule,” issued September 2003, reiterates that during the ATWS rulemaking, the NRC staff set a goal that the P(ATWS) should be no more than $1x10^{-5}$ per reactor year. P(ATWS) was defined as the annual frequency of an ATWS leading to plant conditions that exceed certain design parameters that can result in core melt, containment failure, and the release of radioactivity and can be viewed as the expected core damage frequency of an unmitigated ATWS. NEDC-33912 states that [[ ]] GEH asserted that [[ ]] Based on the above, GEH concluded [[ ]]

In addition to the documents GEH referenced in NEDC-33912, the staff notes that Staff Requirements Memorandum (SRM)-SECY-90-016, “SECY-90-16—Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements,” dated June 26, 1990, provides the NRC position that if an applicant can demonstrate that the consequences of an ATWS are acceptable, the staff should accept the demonstration as an alternative to the prescriptive requirements of 10 CFR 50.62. While the

Commission direction in SRM-SECY-90-016 was specific to the diverse scram requirements of 10 CFR 50.62, the staff considers this direction as also being applicable to the prescriptive requirements for a SLCS, which historically serves as an additional means to shut down the reactor. In addition, SRP Section 15.8 states that applicants for evolutionary designs "must demonstrate that the failure probability of failing the ATWS success criteria is sufficiently small because either: (1) the criteria are met, or (2) a diverse scram system is installed that reduces significantly the probability of a failure to scram."

Based on the above, the staff finds use of the risk goal for the BWRX-300, as described in NEDC-33912, of P(ATWS) less than $1x10^{-5}$ per reactor year acceptable because it is consistent with the intent of 10 CFR 50.62 as described in SECY 83-293 and NUREG-1780. The staff concludes that an analysis that demonstrates P(ATWS) is less than $1x10^{-5}$ per reactor year [[ ]] could support an exemption from 10 CFR 50.62(c)(4). When the NRC receives an application for a BWRX-300, the staff will conduct an evaluation of reliability or probabilistic analysis that demonstrates the P(ATWS) criterion is met [[ ]], conforms with Limitation and Condition 5.1 of this SE, and confirms that special circumstances justify an exemption from 10 CFR 50.62(c)(4).

<u>10 CFR 50.62(c)(5)</u>

The regulation in 10 CFR 50.62(c)(5) requires that each BWR have equipment to automatically trip the reactor coolant recirculation pumps under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

NEDC-33912 specifies that the BWRX-300 uses natural circulation for reactor coolant flow; therefore, the action of tripping reactor recirculation pumps to limit core flow and power is not applicable. NEDC-33912 also states that the BWRX-300 [[ ]]

SRP Section 15.8 provides guidance acceptable to the staff for satisfying 10 CFR 50.62 and states the following:

> [f]or evolutionary plants, some of the equipment required to satisfy the rule may not *(sic)* apply. For example, passive BWRs do not have recirculation pumps; therefore, these designs cannot provide equipment to trip them as required by the rule. For these designs provision of an equivalent action such as reducing the vessel water level may be acceptable.

Based on the above, the staff agrees that the provisions of 10 CFR 50.62(c)(5) are not applicable to the BWRX-300 because it is a passive design and does not include reactor coolant recirculation pumps. In addition, [[ ]] provides an appropriate compensating measure and is consistent with the staff's expectations described in SRP Section 15.8. The staff will conduct a detailed evaluation to confirm 10 CFR 50.62(c)(5) is not applicable when the NRC receives an application for a BWRX-300 SMR.

4.1.2 10 CFR Part 50, Appendix A, General Design Criterion 12, "Suppression of Reactor Power Oscillations"

GDC 12 requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations that can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.

Section 4.1.2, "10 CFR 50 Appendix A, GDC 12," in NEDC-33912 states the BWRX-300 design addresses stability by using an RPV chimney that increases natural circulation core flow so that a margin to instability is maintained for all modes of operation. NEDC-33912 specifies that the BWRX-300 maintains a coupled power-flow response such that any initial perturbation that does not cause an immediate scram is naturally damped and decays quickly to steady state. In addition, NEDC-33912, Section 4.1.2, states that the relatively small core of the BWRX-300 prevents it from being susceptible to regional modes of oscillation. NEDC-33912 concludes that the BWRX-300 design will meet the requirements of GDC 12 without the need for stability detection and an associated trip system.

The staff finds the approach, as described in NEDC-33912, in combination with the analysis prescribed in Limitation and Condition 5.3 of this SE, can demonstrate compliance with GDC 12. The NRC staff will conduct a detailed evaluation of the thermal-hydraulic codes, methods, and analysis results used to demonstrate that GDC 12 is met without the need for a special stability detection and trip system for all modes of operation, including startup, when the NRC receives an application for a BWRX-300 SMR.

4.1.3 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Functions"

GDC 20 requires that the protection system be designed (1) to automatically initiate the operation of appropriate systems, including the reactivity control systems, to assure that SAFDLs are not exceeded as a result of AOOs and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Section 4.1.3, "10 CFR 50 Appendix A, GDC 20," of NEDC-33912 states, in part, the following:

> [t]he RPS provides timely and appropriate protection to provide a reactor scram for events exceeding limits. These systems ensure that SAFDLs are not exceeded. Scram trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but not be subject to spurious scrams.

GEH concludes that the BWRX-300 design will meet the requirements of GDC 20, and the analyses to demonstrate compliance will be provided during future licensing activities.

The staff finds the approach for the RPS, as described in NEDC-33912, consistent with GDC 20 and, therefore, acceptable. The NRC staff will conduct a detailed evaluation of the protection systems functions to ensure compliance with GDC 20 when the NRC receives an application for a BWRX-300 SMR.

4.1.4 10 CFR Part 50, Appendix A, General Design Criterion 21, "Protection System Reliability and Testability"

GDC 21 requires that the protection system be designed for high-functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Section 4.1.4, "10 CFR 50 Appendix A, GDC 21," of NEDC-33912 states the following:

> [t]he BWRX-300 uses Safety Class 1, safety-related equipment to ensure that high quality is achieved. The system includes redundancy to ensure that trips are reliably enforced, even in the case of a failure of a portion of the system. The ability to test and verify operability is included in the design.

GEH concludes that the BWRX-300 design will meet the requirements of GDC 21.

The staff finds the approach, as described in NEDC-33912, consistent with GDC 21 and, therefore, acceptable. The NRC staff will conduct a detailed evaluation of the protection system reliability and testability characteristics to ensure compliance with GDC 21 when the NRC receives an application for a BWRX-300 SMR.

4.1.5 10 CFR Part 50, Appendix A, General Design Criterion 22, "Protection System Independence"

GDC 22 requires that the protection system be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels, do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Section 4.1.5, "10 CFR 50 Appendix A, GDC 22," of NEDC-33912 states, in part, the following:

> RPS provides a reactor scram when pre-set limits are reached. In addition to the diversity provided by these multiple layers of defense, the appropriate redundancy is included to ensure that reliability is maintained even in the event of failures. The RPS is Safety Class 1, safety-related equipment to ensure that high quality is achieved. The RPS and associated sensors and actuation devices are protected from natural phenomena and are designed as fail-safe to ensure that the safety function is maintained.

GEH concludes that the BWRX-300 design will meet the requirements of GDC 22.

The staff finds the approach for the RPS, as described in NEDC-33912, consistent with GDC 22 and, therefore, acceptable. The NRC staff will conduct a detailed evaluation of the protection system independence characteristics to ensure compliance with GDC 22 when the NRC receives an application for a BWRX-300 SMR.

4.1.6 10 CFR Part 50, Appendix A, General Design Criterion 23, "Protection System Failure Modes"

GDC 23 requires that the protection system be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Section 4.1.6, "10 CFR 50 Appendix A, GDC 23," of NEDC-33912 states the following:

> [t]he BWRX-300 protection system, RPS, is designed such that it fails in a safe state. Upon loss of electrical power or motive force (i.e., air to the HCUs), a reactor scram occurs. The HCUs use stored energy for control rod insertion that are activated by the loss of electrical power to the actuating solenoids. This design ensures a safe state is achieved.

GEH concludes that the BWRX-300 design will meet the requirements of GDC 23.

The staff finds the approach, as described in NEDC-33912, consistent with GDC 23 and, therefore, acceptable. The NRC staff will conduct a detailed evaluation of the protection system failure modes to ensure compliance with GDC 23 when the NRC receives an application for a BWRX-300 SMR.

4.1.7 10 CFR Part 50, Appendix A, General Design Criterion 24, "Separation of Protection and Control Systems"

GDC 24 requires that the protection system be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Section 4.1.7, "10 CFR 50 Appendix A, GDC 24," of NEDC-33912 states, "[t]he BWRX-300 protection system, RPS, is separated from the control systems such as the rod control system such that the RPS effectively performs its function independent of the control systems." GEH concluded that the BWRX-300 design will meet the requirements of GDC 24.

The staff finds the approach, as described in NEDC-33912, consistent with GDC 24 and, therefore, acceptable. The NRC staff will conduct a detailed evaluation of the separation between protection and control systems to ensure compliance with GDC 24 when the NRC receives an application for a BWRX-300 SMR.

4.1.8 10 CFR Part 50, Appendix A, General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions"

GDC 25 requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Section 4.1.8, "10 CFR 50 Appendix A, GDC 25," of NEDC-33912 states that the BWRX-300 RPS, together with other mitigating design features, ensures that SAFDLs are not exceeded for reactivity control malfunctions. Design features and operating strategies that limit the effect of single failures in the reactivity control systems include redundancy in the rod control system, rod patterns, and rod blocks. The RPS provides protection for rod control system malfunctions through power-related trips, as discussed in Section 3.7.3 of this SE.

The staff finds the approach, as described in NEDC-33912, consistent with GDC 25 and, therefore, acceptable. The NRC staff will conduct a detailed evaluation of protection system adequacy to mitigate reactivity control malfunctions to ensure compliance with GDC 25 when the NRC receives an application for a BWRX-300 SMR.

4.1.9 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability"

GDC 26 requires that two independent reactivity control systems of different design principles be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, SAFDLs are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Section 4.1.9, "10 CFR 50 Appendix A, GDC 26," of NEDC-33912, states that the BWRX-300 includes two independent reactivity control systems of different design principles. The first uses control rods, which can be inserted rapidly through hydraulic scram or more slowly using the FMCRD electric motors. NEDC-33912, Section 4.1.9, cites multiple means to achieve hydraulic scram ([[   ]], RPS, and ARI) to provide high confidence that a scram will be initiated when required. The rod control system using the FMCRDs would be used to accommodate normal power changes and provide non-safety-related continuous run-in capability to achieve shutdown. In addition, NEDC-33912 states that the control rods are capable of holding the reactor core subcritical under cold conditions.

The staff has previously interpreted the GDC 26 terms "independent" and "different design principles" to indicate that no credited reactivity control systems or components can be shared and are different enough such that no common failure modes exist. In response to staff questions, GEH revised NEDC-33912, Section 4.1.9, to state that the second independent reactivity control system is the feedwater level control system. The feedwater level control system controls reactivity by controlling the downcomer water level, which affects natural circulation core flow and, therefore, core power. GEH indicated that this function is analogous to the reactor recirculation flow control that is typically credited as the second reactivity control system in forced circulation BWRs. GEH also stated that the feedwater level control system can be used to adjust the downcomer water level during normal power operation, and additional means are available to adjust the water level in other modes of operation.

GEH referenced gadolinium burnable poison in the discussion of compliance with GDC 26. The NRC staff notes that burnable poisons do not constitute a reactivity control system in the context of GDC 26; however, burnable poisons are a means of controlling reactivity and influence the reactivity requirements of the credited reactivity control systems.

The NRC staff finds this approach, as described in NEDC-33912, consistent with GDC 26 and, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm the capability of the control rods and the feedwater level control system to reliably control reactivity and prevent exceeding SAFDLs in accordance with GDC 26 when the NRC receives an application for a BWRX-300 SMR.

4.1.10 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability"

GDC 27 requires that the reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

Section 4.1.10, "10 CFR 50 Appendix A, GDC 27," of NEDC-33912 describes how the [[ ]] provides the core cooling function following a postulated accident, and, as discussed in Section 4.1.1 of this SE, the BWRX-300 design [[ ]]. In addition, NEDC-33912, Section 4.1.10, states that future licensing actions will provide the evaluation to demonstrate compliance with GDC 27 and will consider the highest worth control rod pair associated with an individual HCU to be fully withdrawn. GEH concluded that the BWRX-300 control rods, FMCRDs, and actuation systems ensure adequate shutdown margin, capability, redundancy, and diversity such that there is no need for combined reactivity control systems as required by GDC 27. GEH stated that, in these particular circumstances, GDC 27 would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. Instead, GEH proposed Principal Design Criterion (PDC) 27, "Reactivity control system capability," which states the following:

> The BWRX-300 reactivity control system shall be designed to have the capability of reliably controlling reactivity changes to assure that under postulated accident

> conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

The staff notes, as stated in SECY-18-0099, “NuScale Power Exemption Request from 10 CFR Part 50, Appendix A, General Design Criteria 27, ‘Combined Reactivity Control Systems Capability’,” that the intent of GDC 27 is to require reactor designs to achieve and maintain long-term subcriticality using only safety-related equipment following a postulated accident with margin for stuck control rods. NEDC-33912, Section 4.1.9, states that the insertion of the control blades provides the capability to hold the reactor subcritical under cold conditions. The staff finds that an analysis that demonstrates the BWRX-300 control rods alone provide adequate shutdown margin for long-term subcriticality following a postulated accident such that the ability to cool the core is maintained could support an exemption to GDC 27 and replacement with proposed PDC 27. The staff will conduct a detailed evaluation of the safety-related reactivity control system and confirm that special circumstances are present for justification of an exemption when the NRC receives an application for a BWRX-300 SMR.

4.1.11 10 CFR Part 50, Appendix A, General Design Criterion 28

GDC 28 requires the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other RPV internals to significantly impair the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold-water addition.

According to GEH, GDC 28 is satisfied by providing reactivity control system features that mitigate the postulated reactivity accidents that could damage the reactor coolant pressure boundary greater than limited local yielding or damage that significantly impairs core cooling capability. These features include the FMCRD system and rod control system, which incorporate appropriate limits on the potential amount and rate of reactivity increase, physical design of the FMCRD system, including the bayonet style coupling, FMCRD mechanism latches, and FMCRD separation switches.

In response to staff questions, GEH stated the following in its letter dated August 3, 2020 (ADAMS Accession No. ML20216A748):

> [t]he BWRX-300 uses Global Nuclear Fuel (GNF)-2 fuel, with a core design that is similar to the BWR operating fleet. The approved CRDA methodology [Licensing Topical Report NEDE-33885P-A, Revision 1, “Control Rod Drop Accident Methodology”] will be applied to the BWRX-300 to demonstrate that cladding failures do not occur for the postulated (albeit incredible) CRDA. The results of the rod drop calculations will be discussed in the Probabilistic Risk Assessment (PRA) analysis that will be summarized in a [future licensing activity].

Section 15.4.7.3 of NUREG-1666, "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design," Volume 3, issued April 2014 (ADAMS Accession No. ML14099A532), states that because of the potential consequences of an unrestricted reactivity excursion and to ensure compliance with GDC 28, analysis of a CRDA is required to demonstrate reactor coolant pressure boundary integrity and acceptable radiological consequences for the CRDA, irrespective of the probability of a CRDA. As such, the staff considers a CRDA to be a design-basis postulated reactivity accident. Absent an exemption to GDC 28 justifying a beyond-design-basis CRDA classification, the CRDA analysis should be performed in accordance with design-basis analysis assumptions, and the result of the analysis should be documented consistent with other design-basis transients and accidents (i.e., Chapter 15 of the final safety analysis report). To ensure this treatment, the staff developed Limitation and Condition 5.2 of this SE.

NEDC-33912, Section 4.1.11, states that the safety analyses to demonstrate compliance with GDC 28, including each of the specified transients and accidents, will be provided during future licensing activities. The CRDA event applied to an equilibrium cycle will be analyzed using the approved GNF CRDA methodology (NEDE-33885P-A) following confirmation of its applicability to the final BWRX-300 design. In addition, GNF has indicated plans to deviate from NEDE-33885P-A and will provide information in future licensing activities to support a conclusion that additional cycle-by-cycle CRDA evaluations are not warranted. The staff will review this information and make a finding on the acceptability of this approach at that time.

Based on the design features of the BWRX-300 and, specifically, the ability of the rod control system to implement control rod patterns and control rod blocks, along with the additional analyses described in NEDC-33912 (including a CRDA event analyzed consistent with NEDE-33885P-A and in conformance with Limitation and Condition 5.2 of this SE) that will be performed to support future licensing activities, the NRC staff finds this approach consistent with GDC 28 and, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm that GDC 28 is satisfied and cycle-by-cycle CRDA evaluations are not warranted when the NRC receives an application for a BWRX-300 SMR.

4.1.12 10 CFR Part 50, Appendix A, General Design Criterion 29, "Protection Against Anticipated Operational Occurrences"

GDC 29 requires that the protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

NEDC-33912 states that the BWRX-300 rod control system enforces control rod withdrawal limits and prevents inappropriate control rod withdrawal. Further, NEDC-33912 states that the safety-related means of quickly inserting the control rods is from the control rod drive hydraulic scram system. Section 4.1.12, "10 CFR 50 Appendix A, GDC 29," of NEDC-33912 states that the RPS hydraulic scram function is designed to high quality, and associated sensors and actuation devices are protected from natural phenomena and are designed as fail safe to ensure that the safety function is maintained. Lastly, NEDC-33912, Section 4.1.12, concludes that the BWRX-300 design will meet the requirements of GDC 29.

The staff notes that the BWRX-300 design, as described in NEDC-33912, contains an [[ ]] and NEDC-33912 does not describe the design and quality requirements applied to the system. Therefore, the staff bases its finding on the approach described above related to the safety-related RPS hydraulic scram function. The staff finds this approach consistent with GDC 29 and, therefore, acceptable. The staff will conduct a detailed evaluation of RPS and reactivity control system functional reliability, quality, separation, independence, and testability to confirm GDC 29 is satisfied when the NRC receives an application for a BWRX-300 SMR.

4.2 NUREG-0800 Guidance

The BWRX-300 employs novel design features and strategies to ensure safety at the facility. Section 4.2, "NUREG-0800 Standard Review Plan Guidance," of NEDC-33912 identified applicable guidance, clarifications, and departures from SRP Section 4.3, "Nuclear Design"; Section 7.2, "Reactor Trip System"; and Section 15.8. Currently, the NRC does not plan to update or develop staff guidance specific to BWRX-300. GEH recommends that the staff can use the existing SRPs during a future licensing review. The staff agrees that the existing SRPs are adequate, and that in the future the staff can use the design-specific information GEH included in NEDC-33912, Section 4.2, as a review aid.

4.3 Generic Issues

Section 4.3, "Generic Issues," of NEDC-33912 addresses generic issues relevant to the scope of NEDC-33912. Specifically, Section 4.3.1 of NEDC-33912 states that NUREG-1780 sets the risk goal for the probability of an ATWS at no greater than 1×10^{-5} per reactor year. GEH has committed to achieving this goal, and as discussed in Section 4.1.1 of this SE, the staff finds this approach acceptable and consistent with the underlying purpose of 10 CFR 50.62. The staff will evaluate whether this goal is met when the NRC receives an application for the BWRX-300 SMR.

5.0 Limitations and Conditions

If an applicant chooses to incorporate by reference NEDC-33912 as part of a 10 CFR Part 52 design certification application, or if a license applicant uses it for requesting a construction permit and operating license under 10 CFR Part 50 or a combined license under 10 CFR Part 52, it must provide appropriate safety analyses to demonstrate compliance with applicable regulatory requirements.

Additionally, any applicant referencing NEDC-33912 must perform and document in an application the following:

5.1 Reliability analysis or testing, considering applicable operating experience and expected load follow conditions, of the BWRX-300 diverse scram features to demonstrate the probability of an ATWS is less than 1×10^{-5} per reactor year [[ ]].

5.2 A CRDA design-basis safety analysis applied to an equilibrium cycle in accordance with an approved methodology, providing justification for any deviations (e.g., performing a one-time analysis to bound cycle-by-cycle variations), or request an exemption to justify the CRDA as a beyond-design-basis event and document the CRDA analysis results in the probabilistic risk assessment.

5.3 A stability analysis in accordance with an approved methodology to demonstrate that the BWRX-300 maintains a coupled power-flow response such that any operational perturbation, maneuver, or AOO that does not cause an immediate scram is naturally damped and decays quickly to steady state for all modes of operation; prevents SAFDLs from being exceeded; is not susceptible to regional or radial modes of oscillation; and includes necessary provisions to address cycle-specific conditions.

6.0 Conclusion

Based on the above discussion, the NRC staff concludes that the design requirements, acceptance criteria, and regulatory bases for the design functions of BWRX-300 reactivity control design functions, as described in NEDC-33912, are acceptable. In particular, NEDC-33912 describes design requirements for the reactor protection system, [[ ]], and alternate rod insertion to meet the acceptance criteria in 10 CFR 50.62 with justification to provide for a [[ ]], as well as design requirements for the [[ ]] and diverse means to insert control rods to ensure that the reactor can be shut down. If an applicant for a construction permit under 10 CFR Part 50, or a design certification or combined license under 10 CFR Part 52, is not able to demonstrate compliance with an NRC regulation when the detailed design of the BWRX-300 SMR is complete, the applicant will be expected to justify an exemption from the applicable regulatory requirement. The NRC staff will evaluate the regulatory compliance of the final design of the reactivity control design features for the BWRX-300 SMR during future licensing activities, in accordance with 10 CFR Part 50 or 10 CFR Part 52, as applicable. As discussed in this SE, GEH indicated that the detailed design of the BWRX-300 SMR is not complete at this time. The NRC staff will make a final determination of the BWRX-300 SMR's acceptability when the detailed design is completed and reviewed by the NRC staff during future licensing activities.

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LIST OF FIGURES

REVISION SUMMARY

Revision Number	Description of Change
0	Initial Issue
Supplement 1	Revised to incorporate the following responses to NRC Requests for Additional Information (eRAIs): • NRC eRAI 9761, Question NONE-4, revised Section 4.1.11 for BWRX-300 compliance to 10 CFR 50, Appendix A, GDC 28, and to add new References 5.7 and 5.8 for ESBWR Request for Additional Information (RAI) 4.6-23 (ADAMS Accession Nos. ML081090147 (Public) and ML081090148 (Non-Public) and NEDE-33885P-A, Revision 1, Global Nuclear Fuels (GNF) CRDA Application Methodology, respectively. • NRC eRAI 9761, Question NONE-5, revised Section 4.1.9 for BWRX-300 compliance to the statement "independent reactivity control systems of different design principles" in 10 CFR 50, Appendix A, GDC 26. • Revised Section 5 in response to NRC eRAI 9761, Question NONE-4, to add references to ESBWR RAIs and GNF LTR NEDE-33885P-A. • Corrected wording in the Purpose section to read: Design requirements are specified … and the backup means to automatically or manually insert control rods to ensure reactor shutdown. • Editorial correction in last sentence of Section 4.3.1. Changed wording from [["the use of an SLCS" to "the use of SLCS"]]. • Information regarding the FMCRDs has been reclassified as non-proprietary and is identified with change bars in Sections 2.2.2.1, 3.5, 3.7.1 and 4.1.1.
1	• Created -A version by adding the NRC's Final Safety Evaluation (Reference 5.9) and GEH's responses to the NRC's Requests for Additional Information (RAIs) (Reference 5.10). • Added References 5.9 and 5.10.

Acronyms and Abbreviations

Term	Definition
ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ALWR	Advanced Light-Water Reactor
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ASME	American Society of Mechanical Engineers
[[	]]
ATWS	Anticipated Transient Without Scram
B&PV	Boiler & Pressure Vessel
BL-DBA	Baseline Design Basis Analysis
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CDF	Core Damage Frequency
CN-DBA	Conservative Design Basis Analysis
COL	Combined Operating License
CP	Construction Permit
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DCA	Design Certification Application
DEC	Design Extension Condition
DL	Defense Line
ECCS	Emergency Core Cooling System
ESBWR	Economically Simplified Boiling Water Reactor
EX-DBA	Extended Design Basis Analysis
FMCRD	Fine Motion Control Rod Drive

Term	Definition
FSF	Fundamental Safety Function
GDC	General Design Criteria
GEH	GE Hitachi Nuclear Energy
HCU	Hydraulic Control Unit
HGNE	Hitachi-GE Nuclear Energy Ltd.
HVAC	Heating, Ventilation, and Air-conditioning
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICS	Isolation Condenser System
LOCA	Loss-of-Coolant Accident
LTR	Licensing Topical Report
MSIV	Main Steam Isolation Valve
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OL	Operating License
PDC	Principal Design Criterion
PIE	Postulated Initiating Event
PWR	Pressurized Water Reactor
RCPB	Reactor Coolant Pressure Boundary
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SAFDL	Specified Acceptable Fuel Design Limits
SLCS	Standby Liquid Control System
SMR	Small Modular Reactor
SRNM	Source Range Neutron Monitor
SRP	Standard Review Plan
SSC	Structure, System, and Component

1.0 INTRODUCTION

1.0 Purpose

The purpose of this report is to provide the design requirements, acceptance criteria, and regulatory basis for the BWRX-300 reactivity control design functions, specifically for the following areas:

- Design requirements are specified for the Reactor Protection System (RPS) and the [[ ]] such that they satisfy the defense-in-depth (D-in-D) and diversity requirements to protect from common cause failure (CCF) of the RPS. Design requirements are also specified for other associated functions such as Alternate Rod Insertion (ARI) to ensure that the automatic hydraulic reactor scram will meet specified reliability requirements. The design of the RPS, [[ ]] and associated D-in-D features meet the requirements of 10 CFR 50.62 with justification provided for a [[ ]]. In addition, the requirements of 10 CFR 50 Appendix A, General Design Criteria (GDC) 12, GDC 20, GDC 21, GDC 22, GDC 23, GDC 24, GDC 25, GDC 26, GDC 28, and GDC 29 are met, with justification provided for a proposed exemption to the specific requirements of GDC 27 as proposed in Principal Design Criterion (PDC) 27.
- Design requirements are specified for the [[ ]] and the backup means to automatically or manually insert control rods to ensure reactor shutdown. [[ ]] These design features meet the requirements of 10 CFR 50.62 with the [[ ]]. In addition, the requirements of 10 CFR 50 Appendix A, GDC 12 and GDC 26 are met, with justification provided for PDC 27.

1.1 Scope

The scope of this report includes the following:

- A technical description of the BWRX-300 RPS, [[ ]] and other D-in-D design features and design functions to ensure the capability to shutdown the reactor and provide for control of reactivity and reactor fuel thermal limits, including acceptance criteria, regulatory bases, and references to existing proven design concepts based upon previous Boiling Water Reactor (BWR) designs, including the Advanced Boiling Water Reactor (ABWR) and Economically Simplified Boiling Water Reactor (ESBWR).
- A regulatory review of the BWRX-300 RPS, [[ ]] and other D-in-D design features and design functions to ensure the capability to shutdown the reactor and provide for control of reactivity and reactor fuel thermal limits, including acceptance criteria, to describe compliance with regulatory requirements and to describe the bases for any exemptions to regulatory requirements or alternative approaches to regulatory guidance that may be referenced in future licensing activities either by GEH in support of a 10 CFR 52 Design Certification Application (DCA) or by a license applicant for requesting a Construction Permit (CP) and Operating License (OL) under 10 CFR 50 or a Combined Operating License (COL) under 10 CFR 52.

2.0 TECHNICAL DESCRIPTION OF REACTIVITY CONTROL

2.0 General Introduction

The BWRX-300 is an approximately 300 MWe, water-cooled, natural circulation Small Modular Reactor (SMR) utilizing simple safety systems driven by natural phenomena. It is being developed by GE Hitachi Nuclear Energy (GEH) in the USA and Hitachi-GE Nuclear Energy Ltd. (HGNE) in Japan. It is the tenth generation of the BWR. The BWRX-300 is an evolution of the U.S. NRC-licensed, 1,520 MWe ESBWR. Target applications include base load electricity generation and load following electrical generation.

The design features and functions for the mitigation of loss-of-coolant accidents (LOCAs) and small pipe breaks, and for ensuring overpressure protection requirements for the reactor pressure vessel (RPV), are delineated in Licensing Topical Report (LTR) NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 5.1].

The reactivity control design features and functions incorporate design, analysis, and operating experience from the BWR operating fleet, the ABWR, and the ESBWR. Additional D-in-D design features and functions, and design improvements, have been incorporated in the design of the BWRX-300.

2.1 Systems and Components for Control of Reactivity

The BWRX-300 uses control rods in order to change reactivity and therefore reactor power. Additionally, the reactor core is designed to include burnable poisons to allow for efficient fuel loading and management of the cycle excess reactivity.

Design Requirements:

- The core design along with the control rod negative reactivity results in ample shutdown margin in order to ensure that the reactor can remain shutdown in a cold, xenon-free condition throughout the cycle by use of control rods alone with the highest worth control rod pair associated with an individual HCU withdrawn.
- The control rods are positioned in fine increments for normal power adjustments and are also used for rapid insertion by multiple means to achieve shutdown.

2.1.1 Control Rods

The BWRX-300 employs bottom-entry, cruciform-shaped control rods as is typical of BWRs since 1961. This design concept benefits from nearly six decades of service in fleets of operating BWRs around the world.

The control rods intended for use in BWRX-300 are based on the designs used in the operational BWR fleet. This means the methods to design, evaluate and analyze the control rods, in their role as the primary means of reactivity control, are well understood. They have been exercised repeatedly, improved over time, and remain in use today.

The application of this large base of operating experience to the BWRX-300 design supports extremely high reliability of the control rods (and supporting systems) in their reactivity control role.

Design Requirements:

- Diverse sources of control rod motive force and diverse sets of control and actuation logic are provided in the design to provide extremely high confidence that the control rods can be inserted into the reactor core when necessary.
- The control rods are used for power shaping, power level adjustments, and insertion of negative reactivity to achieve shutdown.
- Control rods are the primary means of achieving shutdown in normal operations, Anticipated Operational Occurrences (AOOs), Postulated Accidents, and beyond design basis events and severe accident scenarios.
- Control rods include a bayonet style coupling to prevent inadvertent uncoupling from the FMCRD.

The RPV is a vertical, cylindrical pressure vessel fabricated with rings and rolled plate welded together, with a removable top head by use of a head flange, seals and bolting. The vessel also includes penetrations, nozzles, and reactor internals support. The reactor vessel is relatively tall which permits natural circulation driving forces to produce abundant core coolant flow.

Figure 2-1 shows a representation of BWRX-300 RPV and internals, including the location and relative placement of the control rod blades and associated fine motion control rod drives (FMCRDs).

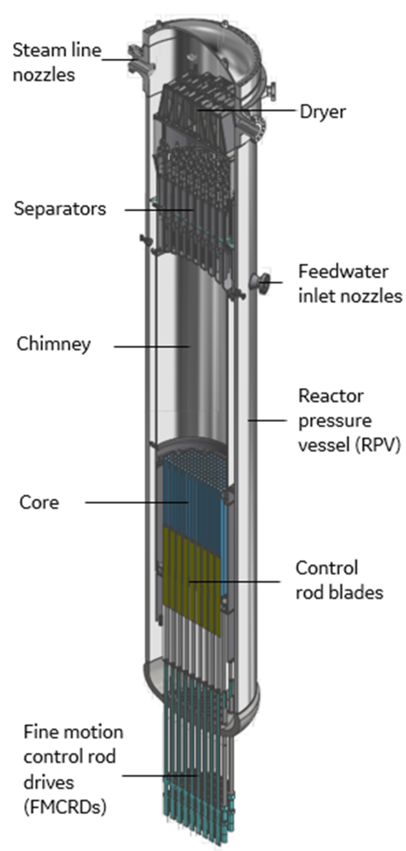

Figure 2-1: BWRX-300 Reactor Pressure Vessel and Internals

Each control rod includes a top handle, an absorber blade section, and a bottom coupling, assembled into a cruciform shape. A typical BWRX-300 control rod is shown in Figure 2-2. Each wing of an absorber blade is an array of stainless-steel tubes filled with boron carbide powder or a combination of boron carbide powder and hafnium rods. While moving vertically within the core, the absorber blade section travels through the cruciform envelope between surrounding fuel bundles. Handle pads guide the control rod along channels, and bottom connector rollers guide the control rod within a guide tube as the control rod is inserted and withdrawn from the core.

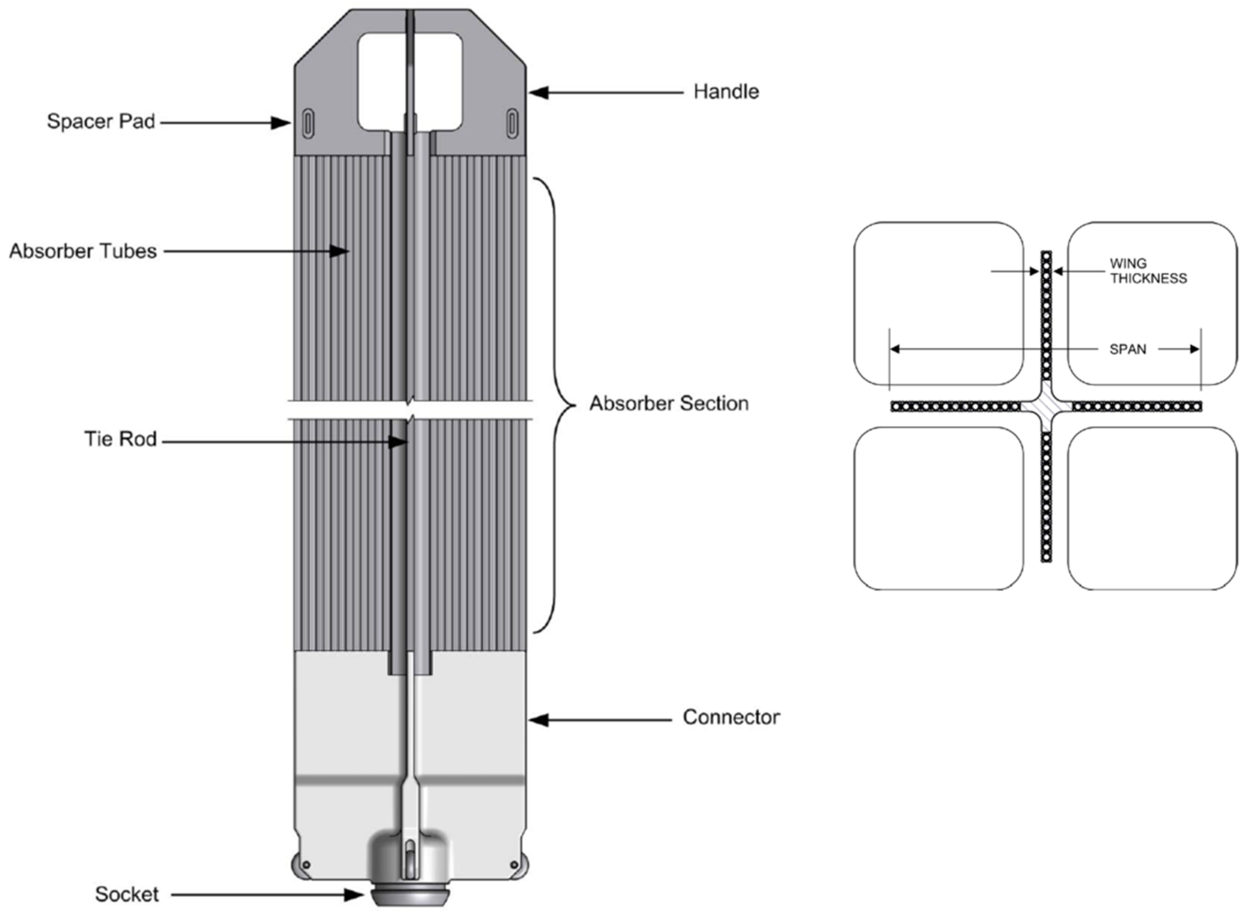

Figure 2-2: BWRX-300 Control Rod and Core Cell Arrangement

The BWRX-300 design includes 57 control rods distributed throughout the reactor core, which has 240 fuel bundle assemblies, as shown in Figure 2-3. The BWRX-300 core uses an N-Lattice configuration. This lattice design provides a larger dimension between fuel bundle centerlines than predecessor designs such as the BWR-6. This dimension includes the space for the control rod. A limited number of the control rods are used for normal power changes. The associated fuel cells are designated as control cells. The other control rods are primarily repositioned during significant power changes including startups and shutdowns. The control rods provide ample shutdown margin when inserted into the core during all conditions including when cold and xenon-free when the pair of control rods of the highest worth controlled by a single Hydraulic

Control Unit (HCU) is assumed to remain in the fully withdrawn position. A reactor core reload analysis is performed prior to every fuel cycle to define the core operating strategy for that cycle.

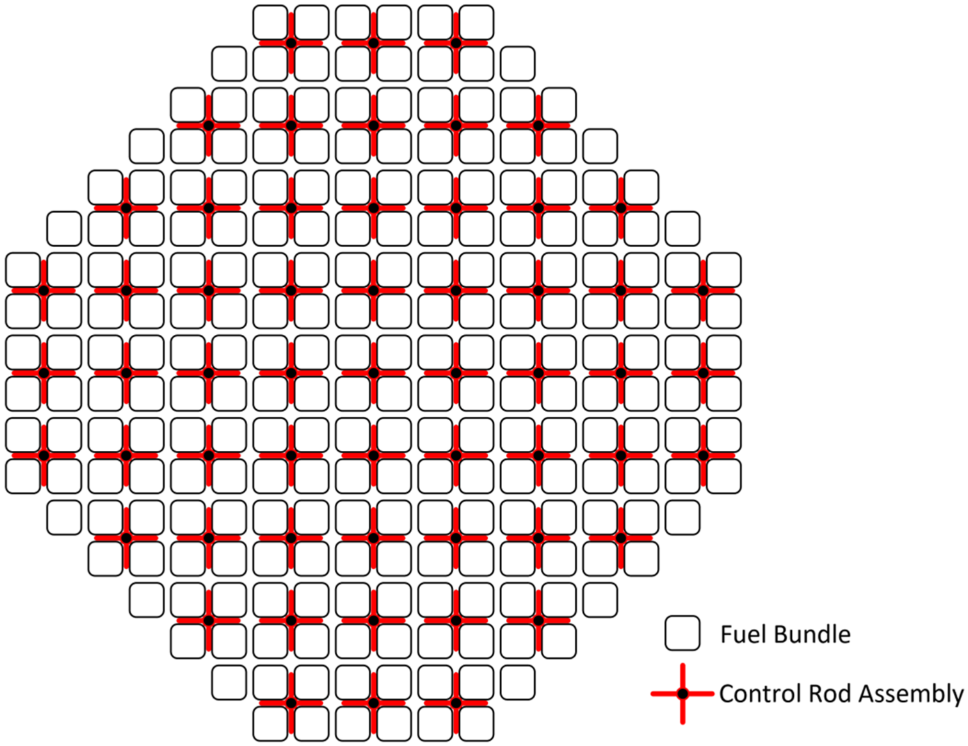

Figure 2-3: BWRX-300 Control Rod Locations Within Core

2.1.2 Control Rod Drives

Each Control Rod Assembly (i.e., control rod) is coupled to a Control Rod Drive (CRD) which is used to position the control rod.

Design Requirements:

- There are two diverse motive forces for the CRD and the associated control rod.
  - The control rods are normally positioned with an electric motor drive.
  - When a rapid shutdown is desired, the control rods are inserted hydraulically by use of high-pressure water.

The CRDs that are used for BWRX-300 are called FMCRDs to indicate the dual diverse means of movement as opposed to the predecessor locking piston hydraulically driven CRDs.

2.1.2.1 Fine Motion Control Rod Drive System

Design Requirements:

- During power operation, changes in core reactivity are controlled by movement and positioning of the control rods within the core, in fine increments, using FMCRD electric motors (one motor per control rod).
- The FMCRD motors also provide continuous run-in functionality to achieve shutdown.

- In the event of a postulated initiating event (PIE) that requires a rapid reactor shutdown, and if the reactor scram fails or is delayed, then the reactor is shut down by the electric motor run-in of FMCRDs function.
- [[

  ]]
- The FMCRDs include separation detection sensors that sense that the hollow piston along with the associated control rod are resting on the ball nut. These separation detection sensors actuate a control rod block signal.
- Rod withdrawal block signals prevent control rod withdrawal when required to enforce established control rod patterns.
- A rod withdrawal block signal prevents withdrawal of FMCRDs based upon an SRNM high period signal during startup.

The fine positioning and shutdown capabilities are achieved with a ball-nut and ball-screw arrangement driven by the FMCRD motor. The ball-nut is keyed to a guide tube to prevent its rotation and traverses through the guide tube vertically as the ball-screw is rotated. A hollow piston, connected to a control rod, rests on the ball-nut. The weight of the control rod keeps the hollow piston and ball-nut in contact during positioning in both insert and withdraw positioning. A schematic of the FMCRD including motor is illustrated in Figure 2-4.

[[

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Figure 2-4: BWRX-300 FMCRD Schematic

2.1.2.2 Control Rod Drive Hydraulic Scram System

In BWR designs, a fast insertion of control rods using stored hydraulic energy is referred to as a scram or hydraulic scram. The force required for hydraulic scram is provided by 29 HCUs that include nitrogen-charged accumulators. A hydraulic scram is initiated by opening 29 scram valves, one on each accumulator water discharge path.

Design Requirements:

- Scram valves open by spring action and are normally held closed by pressurized control air.
- To cause hydraulic scram, a de-energizing reactor trip signal is provided to solenoid-operated pilot valves that vent the control air from the scram valves for opening.
- The design also includes a [[ ]].
- The scram valves are "fail safe" in that loss of either electrical power to the solenoid pilot valve or loss of control air pressure results in opening the scram valve, and hydraulic insertion of all control rods.
- When an accumulator water discharge path through a scram valve is opened, high-pressure nitrogen raises a piston within the accumulator forcing water through the scram piping at high pressure.
- A single accumulator's water is directed to a scram inlet connection on each of two CRDs, with exception of the one CRD located in the center of the core which has its own dedicated accumulator.
- Inside each CRD, the high-pressure water bypasses the ball-nut and lifts the hollow piston, driving the control rod into the core.
  - The scram water is discharged directly into the reactor vessel via clearances between CRD components.
  - A spring washer buffer assembly stops the hollow piston at the end of its stroke.
- Departure from the ball-nut releases spring-loaded latches in the hollow piston to engage slots in the guide tube.
  - These latches support the hollow piston in the fully inserted position.
  - Following a hydraulic scram insertion, the control rod cannot be withdrawn until the ball-nut is driven up, re-engaged, and the hollow piston de-latched from the guide tube.
- The design also includes ARI pilot valves on the control air header, which serves all 29 scram valves.
  - The ARI pilot valves are energized-to-actuate and provide an alternate path to vent control air and open all scram valves resulting in hydraulic insertion of all control rods.
- Any time an automatic hydraulic scram is initiated, a "scram follow" signal is generated such that each FMCRD motor drives the ball-nut to a position just below the fully inserted hollow piston.

– This establishes a second means (in addition to the spring-loaded latches described above) to prevent any control rod from dropping out of its fully inserted position.

Figure 2-5 shows a simplified view of an FMCRD depicting hydraulic scram.

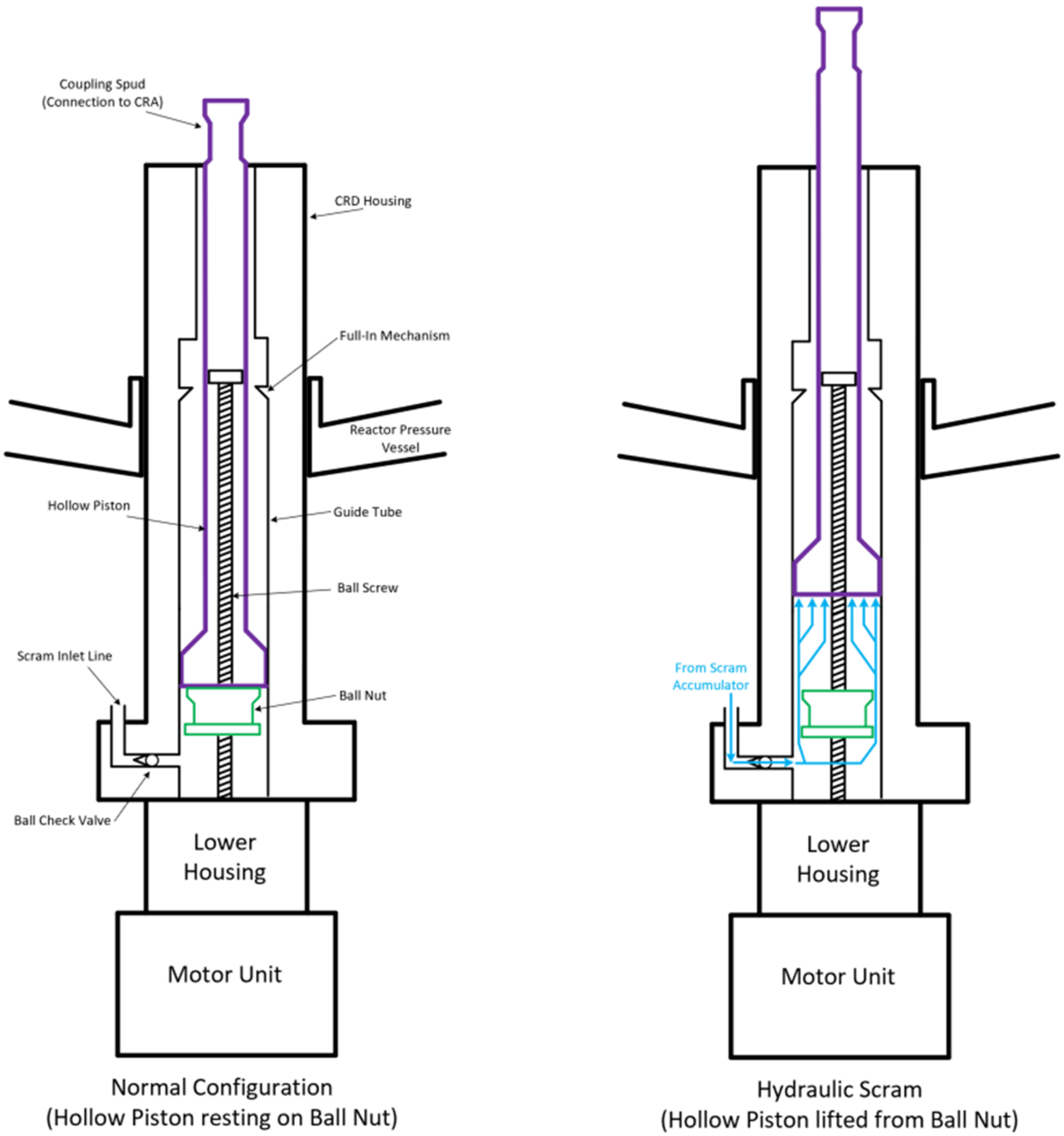

Figure 2-5: BWRX-300 Simplified View of FMCRD with Hydraulic Scram

Figure 2-6 shows hydraulic scram and ARI operation.
[[

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Figure 2-6: BWRX-300 Hydraulic Scram and ARI Schematic

The scram functions are categorized as defense line (DL) functions as further described in Section 3.0 to provide D-in-D for scram. Hydraulic scram by the RPS is a DL3 Safety Category 1 safety-related function, ARI hydraulic scram is a DL4a Safety Category 2 function, [[ ]]. The scram follow is designated as a Safety Category 3 function and is not needed for success of hydraulic scram. [[ ]]

2.2 BWRX-300 Associated Mitigating Systems

Design Requirements:

- In the case of a PIE that requires a rapid reactor shutdown, and if the reactor scram fails or is delayed, then the [[ ]].

The ICS is described in LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 5.1]. A brief description of these features and the associated effects are described below.

2.2.1 Isolation Condenser System

The arrangement of one IC heat exchanger situated in an IC pool is shown in Figure 2-7. The [[ ]] is discussed in NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 5.1].

[[

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Figure 2-7: BWRX-300 Isolation Condenser System (Only One Train Shown)

Design Requirements:

- The ICS passively removes heat from the reactor (i.e., heat transfer from the IC heat exchanger tubes to the surrounding IC pool water is accomplished by condensation and natural circulation, and no forced circulation equipment is required) when the normal heat removal system is unavailable following any of the following events:
  - Sudden reactor isolation at power operating conditions
  - During station blackout (i.e., unavailability of all alternate current (AC) power)
  - Anticipated Transient Without Scram (ATWS)
  - LOCA
- The ICS consists of three independent trains, each containing an IC heat exchanger that condenses steam on the tube side and transfers heat by heating and evaporating water in the IC pool, which is vented to the atmosphere.
- [[
  ]]
- To start an IC train, the IC condensate return valve is opened whereupon the standing condensate drains into the reactor. [[

]]

- The IC pools have a total installed capacity that provides approximately seven days of reactor decay heat removal capability.
  - The heat rejection process can be continued by replenishing the IC pool inventory.
- The [[
  ]].

The BWRX-300 ICS is based on the ESBWR ICS design [Reference 5.2]. The ICS is designed as a safety-related system to remove decay heat passively [[

]]

The ICS contains IC heat exchangers that condense steam on the tube side and transfer heat to the IC pool. The IC heat exchangers, connected by piping to the RPV, are placed at an elevation above the source of steam (RPV) and, when the steam is condensed, the condensate is returned to the RPV via a condensate return pipe.

The steam side connections between the RPV and the IC heat exchangers are normally open, and the condensate lines are normally closed. This allows the IC heat exchangers and drain piping to fill with condensate, which is maintained at a subcooled temperature by the IC pool water during normal reactor operation.

The ICS is placed into operation by opening condensate return valves and draining the condensate to the RPV, thus causing steam from the reactor to fill the tubes which transfer heat to the cooler IC pool water.

3.0 DEFENSE-IN-DEPTH OF REACTIVITY CONTROL FUNCTIONS

3.0 General Overview of Defense-in-Depth (D-in-D) Concept

A plant-level D-in-D concept is applied to the BWRX-300 design. The BWRX-300 design is arranged in defense lines, consistent with the levels of defense defined in International Atomic Energy Agency (IAEA) SSR-2/1 [Reference 5.3]. The D-in-D concept defines the design and analysis rules governing that arrangement such that the defense lines have good alignment with the safety assessments defined in a BWRX-300 safety assessment framework used to demonstrate plant safety. The D-in-D concept is applicable to the BWRX-300 systems and equipment responsible for performing functions assigned to each DL. This includes both the primary systems directly acting on the nuclear and heat generation processes and their supporting systems. The functions are assigned to DLs based on their roles in the layered, deterministic design basis analyses including Baseline Design Basis Analysis (BL-DBA), Conservative Design Basis Analysis (CN-DBA), and Extended Design Basis Analysis (EX-DBA).

The BWRX-300 uses a D-in-D integrated [[

]] This is further described in LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 5.1].

Multiple defense lines provide layered protection against unacceptable releases of radiation. The defense lines include engineering and operational practices, plant features, and plant functions. These features, functions and practices are designed such that:

- The existence of pre-conditions which could lead to accident scenarios are minimized.
- The normal operation of the plant is monitored and controlled such that PIEs can be mitigated before evolving into accident scenarios.
- If an accident scenario does develop, the consequences are limited.
- Multiple defense lines are independently capable of performing the plant's fundamental safety functions (FSFs).

Five defense lines are adopted for the BWRX-300 D-in-D concept, consistent with the IAEA lines of defense approach. The first and fifth defense lines do not include performance of plant functions. The first line minimizes potential for accidents to occur in the first place and minimizes potential for failures to occur in subsequent defense lines by applying high-quality and conservatism in design, construction and operation. The fifth line involves off-site emergency preparedness to protect the public in case a substantial radioactive release occurs. The second, third, and fourth lines comprise plant functions which act to prevent PIEs from leading to significant radioactive releases.

Among the second, third and fourth defense lines, two independent and diverse lines can mitigate any PIE caused by a single failure or by a single operator error.

Among the second, third and fourth defense lines, at least one line can mitigate any PIE caused by a CCF in another line, with the mitigation means being independent from the effects of the initiating CCF.

The fourth defense line additionally contains independent provisions for prevention and/or mitigation of severe accidents.

The adequacy of the defense lines is assessed and demonstrated using layered, deterministic safety analyses which are designed to exercise the different defense lines. The PIEs to be considered in these analyses are selected based on rigorous and systematic failure modes and effects analyses of the plant systems, as well as internal and external hazard evaluations, and human operation hazards.

Functional and design requirements are derived from the deterministic safety analyses, and from the D-in-D concept itself, to ensure that the defense line functions are implemented in the design consistent with their role in the D-in-D concept, and the credit taken for them in the safety analyses.

The above concept states that for any PIE due to a CCF, at least one diverse and independent defense line must be able to mitigate the PIE's effects. This approach is consistent with the general IAEA guidance to provide "several" layers of protection. When DL1 and DL4b are explicitly considered, several layers of protection are provided:

- DL1 is provided to protect against occurrence of the CCF in the first place,
- An additional defense line, among DL2-4a, is provided to mitigate the effects of the CCF if it were to occur, and
- DL4b is provided as further protection against the sequence becoming a severe accident.

Figure 3-1 shows a representation of the D-in-D concept. The BWRX-300 functions to support D-in-D associated with reactivity control are discussed below. The combination of these functions ensures that the BWRX-300 adequately controls reactivity during normal, abnormal, design basis, and CCF events. The functions associated with reactivity management and associated events are listed for each defense line in the sections below.

[[

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Figure 3-1: BWRX-300 Simplified Defense-in-Depth Concept

3.1 Defense Line 1

DL1 includes the quality measures taken to minimize potential for failures and initiating events to occur in the first place and to minimize potential for failures to occur in subsequent lines of defense. These quality measures cover the design, construction, operation and maintenance of the plant. DL1 also includes the use of appropriate conservatism in design and analyses. DL1 does not include plant functionality; the plant functions themselves belong to subsequent lines of defense. However, the DL1 quality and conservatism measures are inherent in the design performance, and reliability of subsequent defense line functions, support the effectiveness of those functions, and provide assurance that opportunities to defeat multiple defense lines are minimized.

DL1 measures result in fewer challenges to the RPS. By incorporating operating experience into the design and operation as well as including appropriate quality measures, the plant experiences fewer trips. The BWRX-300 design benefits from the experience gained from the prior generations of BWRs and other power plants for select features. The continual learning incorporated into the design and operation results in elimination of some events and the reduction of frequency and severity of other events.

Although there are many DL1 activities that are applicable to reactivity control, some examples include the following:

- Technical Specification operational controls
- N-Lattice core less likely to experience control rod binding
- Advances in channel materials and core design/operation minimize probability of channel bow
- Normal power changes are with control rods – continuous observation of normal function
- Reliability measures included in design minimize probability of PIEs and failure of mitigation
- [[ ]]
- Seismic qualification ensures core geometry maintained
- FMCRDs similar to ABWR and ESBWR
- Control Rod Blades
  - Same as ABWR
  - Almost identical to latest design for BWR fleet
- ABWR fleet has 22+ years of operating experience with control rod blades and FMCRDs
- Bayonet style coupling of control rod to FMCRD to prevent inadvertent uncoupling

3.2 Defense Line 2

DL2 contains plant functions designed to control or initiate responses to PIEs, especially AOOs, before any parameters reach a DL3 actuation setpoint. The effectiveness of DL2 functions is assessed in the BL-DBA, with the DL2 functions, and equipment performing those functions, subject to functional and design requirements derived from the BL-DBA.

Those functions which normally operate to control the plant parameters on a continuous basis are part of DL2. Other functions such as blocking control rod motion and anticipatory plant trips are also part of DL2. The functions in DL2 are assigned to Safety Category 3 and performed by (at least) Safety Class 3 equipment as defined in IAEA guidance. These functions are considered non-safety related in the US, but the appropriate quality and reliability measures are applied to ensure functional performance as a D-in-D measure. The functions in DL2 must be performed independently from DL3 functions, and any portion of DL2 functions subject to CCF must be performed diversely from corresponding portions of functions in DL3.

The DL2 functions are used as normal plant controls and therefore operate regularly and with a high degree of reliability. Because of their importance to plant operation, the D-in-D functions are supported by operational reliability measures. These features control or prevent many PIEs. They minimize the probability that a PIE will challenge DL3.

DL2 features that are important for reactivity control include the following:

- [[

  ]]
- The rod control system provides normal control of control rods within established movement patterns
- Control rod blocks mitigate incorrect rod withdrawals

3.3 Defense Line 3

DL3 contains plant functions which act to mitigate a PIE by preventing fuel damage when possible, assuring the integrity of the barriers to release, and placing the plant in a safe state. DL3 also includes functions credited to maintain the plant in a safe condition following mitigation of PIEs, until normal operations are resumed. The effectiveness of DL3 functions is assessed in the CN-DBA, with the DL3 functions, and equipment performing those functions, subject to functional and design requirements derived from the CN-DBA.

DL3 functions typically include reactor scram and actuation of engineered safety features. The DL3 functions are needed when DL2 is not effective at intercepting a PIE or when a PIE is simply beyond the capabilities of the DL2 functions. The functions in DL3 are assigned to Safety Category 1 and performed by Safety Class 1 equipment as defined by IAEA guidance. These features are treated as safety-related in the US.

The systems and equipment involved in performance of DL3 functions are as simple as possible. Examples of the desired simplicity include eliminating the need for active support systems (e.g., power supply; heating, ventilation, and air-conditioning (HVAC); and cooling water) and minimizing the need for active control functions (pumps, motors, and actively controlled valve positioners).

DL3 features that are important for reactivity control include the following:

- RPS hydraulic scram
- [[

]]

3.4 Defense Line 4

DL4 includes two subsets of functions, designated as DL4a and DL4b functions.

DL4a functions are those which can place and maintain the plant in a safe state in case of PIEs with failure of the DL3 functions. The DL4a functions should prevent the progression of accidents or radioactive release to the public. The need for DL4a functions generally arises from specific, postulated CCFs occurring in DL3. The effectiveness of DL4a functions is assessed in the EX-DBA, with the DL4a functions, and equipment performing those functions, subject to functional and design requirements derived from the EX-DBA.

DL4a functions are assigned to Safety Category 2 and performed by (at least) Safety Class 2 equipment as defined by IAEA guidance. These features are considered non-safety related in the US, but the appropriate reliability and quality control measures are included to ensure that they can be relied upon for D-in-D.

DL4b functions are those explicitly provided to prevent or mitigate an accident involving substantial melting of the nuclear fuel (i.e., a severe accident) while keeping radioactive releases to acceptable levels. DL4b also protects for events that exceed DL1 assumptions regarding PIEs as a result of extreme events, multiple events, or multiple failures.

DL4 features that are important for reactivity control include the following:

- ARI provides hydraulic scram in event of HCU actuation failure
- Electric motor run-in of FMCRDs

3.5 Defense Line 5

DL5 includes emergency preparedness measures to cope with potential unacceptable releases in case the first four defense lines are not effective. These are largely off-site measures taken to protect the public in a scenario involving substantial release of radiation.

DL5 measures are not in the scope of this LTR.

3.6 Specific Reactivity Control Events Considered in Defense-in-Depth Concept

The D-in-D approach that has been applied to the BWRX-300 results in elimination of select events from previous designs, reduction in the frequency of other events, and improved mitigation of events. This section provides a summary of select events. It is not considered an all-inclusive list of BWRX-300 events sequences.

3.6.1 Anticipated Transient Without Scram (ATWS)

As defined in 10 CFR 50.62, an ATWS is an AOO followed by a failure of the reactor trip system. Section 4.0 of this report includes the regulatory evaluation of compliance with the regulatory requirements of 10 CFR 50.62. This section describes the D-in-D features of the BWRX-300 that prevent or mitigate these specific reactivity control events.

As described in Section 3.2, DL1 activities minimize potential for failures and initiating events to occur in the first place that require actuation of the RPS. Industry data supports the fact that the operating nuclear fleet has effectively reduced the challenges to plant operation that require RPS actuation. BWRX-300 benefits from the predecessor BWR's operating experience.

For BWRX-300 [[

]] A hydraulic scram signal with scram follow is actuated to quickly effect a shutdown. Additionally, the [[

]]

For a case of an event such as described above, if the [[ ]] were to fail to perform the reactor shutdown, the RPS which is in DL3 will sense a [[ ]], such as high reactor pressure, and cause a shutdown through a hydraulic scram signal with scram follow. Additionally, for [[

]]

If a scram signal is applied to the HCUs to cause the control rods to insert, but does not release the stored energy, a backup DL4 ARI signal is applied to the ARI solenoids causing the air header to depressurize, thereby resulting in a release of the HCU stored energy and insertion of control rods.

As further D-in-D against a failure to scram, the [[

]]

Because the BWRX-300 uses natural circulation for reactor coolant flow, the action of tripping reactor recirculation pumps to limit core flow and power is not applicable. [[

]]

The combined effects of the features described above provide multiple layers of defense to ensure an effective scram when needed and to control reactor conditions while the shutdown is being completed by any of the methods listed. These diverse shutdown methods and mitigating features result in a [[ ]].

The BWRX-300 acceptance criteria for evaluating the effectiveness of the reactivity control diverse shutdown methods and mitigating features include the following:

- Pressures in the reactor coolant system and main steam system are maintained below 120% of the reactor coolant pressure boundary (RCPB) design pressure (ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Service Level C limit);
- Peak cladding temperature is within the 10 CFR 50.46 limit of 1204ºC (2200°F);

- Peak cladding oxidation is within the requirements of 10 CFR 50.46;
- Peak containment pressure and temperature do not exceed containment design pressure and temperature;
- A coolable geometry is maintained; and
- Radiological releases are maintained within 10 CFR 100 allowable limits.

The BWRX-300 design requirements associated with ATWS prevention and mitigation include the following:

- [[

  ]]
- RPS initiates a reactor scram based on signals and setpoints needed to support safety analysis credited trips.
- [[

  ]]
- ARI provides a diverse means to actuate the HCUs upon sensing a failure to scram.
- FMCRDs receive an electric motor run-in signal upon sensing a parameter requiring a scram.
- [[
  ]]
- FMCRD insertion time limits are established based on meeting the acceptance criteria of the safety analyses.

3.6.2 Control Rod Drop Accident

As with the ESBWR, the BWRX-300 has features to prevent a Control Rod Drop Accident (CRDA). The FMCRDs have a different coupling design to attach the drive assembly to the control rod than was used for the locking piston control rod drives. The FMCRD uses a bayonet style coupling that requires a 45-degree rotation to uncouple it. Since the FMCRD is firmly bolted into its position under the reactor vessel and the control rod is constrained from rotation by the fuel assemblies, it is not possible for the control rod to become uncoupled from the FMCRD during reactor operation. The hollow piston is the component within the FMCRD that is coupled to the control rod. The hollow piston normally rests on the ball nut internal to the FMCRD. There are dual separation detection devices that sense that the hollow piston along with the associated control rod are resting on the ball nut. If the sensor detects that the hollow piston is no longer on the ball nut, then control rod withdrawal is blocked. Additionally, the hollow piston has latches that prevent inadvertent withdrawal of the assembly when not attached to the ball nut.

These BWRX-300 features prevent inadvertent uncoupling of the control rod from the FMCRD, block withdrawal of the drive assembly if separation of the hollow piston and ball nut occurs and latches a hollow piston that is not resting on the ball nut. Since these items do not allow a separation distance to occur between the FMCRD and the control rod or from the ball nut to an unlatched hollow piston, it is not possible for a control rod drop accident to occur.

The BWRX-300 design requirements associated with CRDA prevention and mitigation include the following:

- FMCRDs include bayonet style coupling to prevent inadvertent uncoupling.
- FMCRDs include separation detection sensors that actuate a control rod block.

3.6.3 Rod Withdrawal Error

The protection system along with other mitigating design features assure that specified acceptable fuel design limits (SAFDLs) are not exceeded for rod withdrawal error events. In order to prevent a rod withdrawal error, the rod control system has redundancy to limit the effect of single failures. Additionally, the rod patterns are enforced for rod withdrawals. If there is a malfunction of the rod control system that results in a rod withdrawal error during startup, a rod block is initiated based upon a startup range neutron monitor (SRNM) high signal. If the rod withdrawal error were to result in a further increase to the SRNM based setpoint, a reactor scram occurs. If a rod withdrawal error were to occur at higher power, the average power range monitor (APRM) scram will terminate the event if it were to continue to its setpoint.

The BWRX-300 design requirements associated with rod withdrawal error prevention and mitigation include the following:

- The SRNMs provide a period based rod block function during startup.
- The SRNMs provide a period based reactor scram signal during startup.
- The APRMs provide a high flux reactor scram signal during power operations.

4.0 REGULATORY EVALUATION

4.0 10 CFR 50 Regulations

4.0.1 10 CFR 50.62

10 CFR 50.62, Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants, addresses an AOO as defined in 10 CFR 50, Appendix A, followed by the failure of the reactor trip portion of the protection system specified in 10 CFR 50 Appendix A, GDC 20. The following requirements are included for traditional BWRs:

- Regulatory Requirement: 10 CFR 50.62(c)(3) requires that each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

  Statement of Compliance: The BWRX-300 includes an ARI system as described in Section 2.2.2.2. The ARI system provides a diverse means of depressurizing the scram air header to ensure that the HCU stored energy is released to cause a reactor scram. Therefore, the BWRX-300 design will meet this requirement of 10 CFR 50.62(c)(3).

- Regulatory Requirement: 10 CFR 50.62(c)(4) requires that each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.

  Statement of Compliance: [[

  ]] Acceptance criteria as described in NUREG-0800, Standard Review Plan (SRP) 15.8, states that for evolutionary plants like the BWRX-300 some of the equipment required to satisfy the rule may not apply. For example, passive BWRs do not have recirculation pumps; therefore, these designs cannot provide equipment to trip them as required by the rule. For these designs provision of an equivalent action such as reducing the vessel water level may be acceptable.

  As discussed in Section 3.7.1, the BWRX-300 includes a comprehensive D-in-D approach to ensure that the reactor is effectively shut down when specified conditions are reached. The BWRX-300 includes a [[

]]. Additionally, the reactor can be shutdown by using the diverse FMCRD electric motor run-in. [[

]]

The BWRX-300 will meet the stated risk goal of 10 CFR 50.62 by providing a [[

]]. 10 CFR 50.62 was primarily issued to address a generic safety issue potentially affecting currently operating plants. The basis for ATWS rule requirements, which are outlined in SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," [Reference 5.4], which concluded that additional ATWS safety requirements were justified and included the stipulation to reduce the risk of core damage because of ATWS to be less than 10^{-5} per reactor year. [[

]]

Requirements for evolutionary plant designs beyond the original 10 CFR 50.62 regulatory requirements are addressed in more detail in SECY-93-087, "Policy, Technical, And Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR Designs)" [Reference 5.5] and addressed in SRP 15.8. The BWRX-300 design will meet the diversity and D-in-D guidelines for ATWS described in SECY-93-087, Item II.Q, Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems, without the use of an SLCS.

The effectiveness of the ATWS Rule requirements was evaluated in NUREG-1780 [Reference 5.6], which states that during the ATWS rulemaking the NRC staff set a goal that P(ATWS) should be no more than 1.0E-05/RY. P(ATWS) was defined as the annual frequency of an ATWS leading to plant conditions that exceed certain design parameters that can result in core melt, containment failure, and the release of radioactivity and can be viewed as the expected core damage frequency (CDF) of an unmitigated ATWS. Updating the original generic ATWS regulatory analysis, using operating data since the ATWS rule was implemented, found that on a generic basis, all four reactor types achieved the ATWS rule risk goal. The risk of core damage from a single CCF to scram is further reduced by reducing challenges to the RPS. NUREG-1780 notes that the initiating event frequency has been reduced by a factor of eight demonstrating that the Commission's recommendation to reduce the number of automatic reactor scrams has been very effective in reducing P(ATWS) (the probability of an ATWS). [[

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This statement of compliance may be used as the bases for the necessary exemption in future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant for requesting a CP and OL under 10 CFR 50 or a COL under 10 CFR 52.

- Regulatory Requirement: 10 CFR 50.62(c)(5) requires that each boiling water reactor must have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

  Statement of Compliance: Because the BWRX-300 uses natural circulation for reactor coolant flow, the action of tripping reactor recirculation pumps to limit core flow and power is not applicable. [[

  ]]

  Therefore, this requirement is not applicable to the BWRX-300. The above statement of compliance may be used as the bases for this conclusion in future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant for requesting a CP and OL under 10 CFR 50 or a COL under 10 CFR 52.

4.0.2 10 CFR 50 Appendix A, GDC 12

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 12, Suppression of reactor power oscillations, requires that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.

  Statement of Compliance: The BWRX-300 design addresses stability utilizing an RPV chimney that increases natural circulation core flow so that a margin to instability is maintained for all modes of operation. In accordance with GDC 12, it is required for the operating BWRs and the BWRX-300 that the power response to a sudden insertion of reactivity (typically a pressurization event) results in a coupled power and core flow response such that SAFDLs are not exceeded. For some reduced flow scenarios in forced circulation BWRs, a scram is required when the power and flow response is not damped to prevent SAFDLs related to critical power ratio (CPR) from being exceeded. Additionally, forced circulation BWRs have enforced operational exclusion zones to avoid regions of high power and low flow. The BWRX-300 maintains a coupled power and flow response such that any initial perturbation that does not cause an immediate scram decays quickly to steady state even at the limiting point of the cycle. Also, the relatively small core of the BWRX-300 causes it not to be susceptible to regional modes of oscillation. Therefore, a special stability detection and trip system is not required. The natural circulation driving head provided by the BWRX-300 chimney along with the core orifice design causes core flow and power perturbations to be naturally damped.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 12.

4.0.3 10 CFR 50 Appendix A, GDC 20

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 20, Protection system functions, requires that the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that SAFDLs are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

  Statement of Compliance: The BWRX-300 uses a D-in-D approach to prevent and mitigate events. In addition to the prevention features included in the rod control system, the [[

  ]]. The RPS provides timely and appropriate protection to provide a reactor scram for events exceeding limits. These systems ensure that SAFDLs are not exceeded. Scram trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but not be subject to spurious scrams. The analyses to demonstrate compliance will be provided during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 20.

4.0.4 10 CFR 50 Appendix A, GDC 21

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 21, Protection system reliability and testability, requires that the protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

  Statement of Compliance: The BWRX-300 uses Safety Class 1, safety-related equipment to ensure that high quality is achieved. The system includes redundancy to ensure that trips are reliably enforced, even in the case of a failure of a portion of the system. The ability to test and verify operability is included in the design.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 21.

4.0.5 10 CFR 50 Appendix A, GDC 22

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 22, Protection system independence, requires that the protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design

techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Statement of Compliance: The BWRX-300 uses a D-in-D approach to ensure that safety is maintained. The reactivity control system prevents inappropriate reactivity additions from the control rods. [[

]] RPS provides a reactor scram when pre-set limits are reached. In addition to the diversity provided by these multiple layers of defense, the appropriate redundancy is included to ensure that reliability is maintained even in the event of failures. The RPS is Safety Class 1, safety-related equipment to ensure that high quality is achieved. The RPS and associated sensors and actuation devices are protected from natural phenomena and are designed as fail-safe to ensure that the safety function is maintained.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 22.

4.0.6 10 CFR 50 Appendix A, GDC 23

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 23, Protection system failure modes, requires that the protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

  Statement of Compliance: The BWRX-300 protection system, RPS, is designed such that it fails in a safe state. Upon loss of electrical power or motive force (i.e., air to the HCUs), a reactor scram occurs. The HCUs use stored energy for control rod insertion that are activated by the loss of electrical power to the actuating solenoids. This design ensures a safe state is achieved.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 23.

4.0.7 10 CFR 50 Appendix A, GDC 24

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 24, Separation of protection and control systems, requires that the protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

  Statement of Compliance: The BWRX-300 protection system, RPS, is separated from the control systems such as the rod control system such that the RPS effectively performs its function independent of the control systems.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 24.

4.0.8 10 CFR 50 Appendix A, GDC 25

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 25, Protection system requirements for reactivity control malfunctions, requires that the protection system shall be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

  Statement of Compliance: The BWRX-300 protection system along with other mitigating design features assure that SAFDLs are not exceeded for reactivity control malfunctions. In order to prevent a rod withdrawal error, the rod control system has redundancy to limit the effect of single failures. Additionally, the rod patterns are enforced for rod withdrawals. If there is a malfunction of the rod control system that results in a rod withdrawal error during startup, a rod block is initiated based upon a SRNM high signal. If the rod withdrawal error were to result in a further increase to the SRNM based setpoint, a reactor scram will occur. If a rod withdrawal error were to occur at higher power, the APRM scram terminates the event if it were to continue to its setpoint. The analyses to demonstrate compliance will be provided during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 25.

4.0.9 10 CFR 50 Appendix A, GDC 26

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 26, Reactivity control system redundancy and capability, requires that two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, SAFDLs are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

  Statement of Compliance:

  Two independent reactivity control systems of different design principles are provided.

  Like previous BWR designs, including the ABWR and ESBWR, one of the two required independent reactivity control systems of different design principles is the hydraulic insertion of the control rods. A positive means for inserting the control rods is the highly reliable HCUs in both operating BWRs and the BWRX 300. In addition, the BWRX-300 has redundant and independent systems ([[ ]]), RPS, and ARI) to ensure that a hydraulic scram is initiated when required.

Additionally, BWRX-300 has the capability of electric motor-driven control rod movement that was not present in operating BWRs with only hydraulic control rod drive systems. This capability also exists for ABWR and ESBWR. Normal power changes in the BWRX-300 are made by the rod control system using fine motion control of specific control rods using the electric motor-driven control of the FMCRDs and by burnable poison, gadolinium, that is included in the fuel pellets in an axial and radial distribution within the core. The BWRX-300 rod control system and associated motor-driven control rod movement is a means of reactivity control that uses an independent control and motive force other than the hydraulic HCU insertion system.

The control blades are capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. Burnable gadolinium in the fuel and the FMCRDs provide for slow reactivity adjustments. The ganged movement of blades via the motors and the scram function provide protection when needed for AOOs and design basis accidents (DBAs).

In BWRs, the second means of reactivity control is the ability to respond to power shape changes by changing core flow and recirculation ratio. In the operating BWR fleet, the recirculation flow is forced using either internal or external recirculation pumps. The BWRX-300, like the ESBWR, does not employ forced circulation, but the core flow does change naturally in response to changes in reactivity and axial power shape in the same way as a jet pump BWR that is operating in natural circulation. For natural circulation, the core flow depends on the downcomer water level, and reducing the water level reduces core flow and thus core power. The BWRX-300 has the ability to reduce power via this mechanism. The Feedwater Level Control System is used to control feedwater flow and therefore controls reactor water level. This system can be used to make adjustments to water level when in normal power operation, and additional means are available for water level adjustments when in other modes of operation.

One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

The requirement that one of the systems shall be capable of holding the reactor core subcritical under cold conditions is satisfied in operating BWRs and the BWRX-300 by the insertion of the control blades. The use of gadolinium burnable poison in the fuel pellets is a diverse means of reactivity control that controls the power profile at the beginning of core life and enables the control rods alone to have adequate shutdown margin. The evaluation to demonstrate compliance considers the highest worth control rod pair associated with an individual HCU to be fully withdrawn and will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 26.

4.0.10 10 CFR 50 Appendix A, GDC 27

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 27, Combined reactivity control systems capability, requires that the reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling

system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Statement of Compliance: As described in LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 5.1], the required emergency core cooling system (ECCS) design functions of the [[

]] to meet the criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(4). Another design function of the [[
]] which is further required to maintain a coolable geometry to meet the criteria in 10 CFR 50.46(b)(4). The worst-case single failure affecting the [[
]] does not prevent fulfillment of the required ECCS design functions. Because of the relatively large volume of reactor coolant above the reactor core during normal operation, there is no need for [[
]] following the worst-case postulated LOCA assuming failure of [[
]]. Following the worst-case postulated LOCA, the [[ ]] continues to provide long-term cooling to meet the requirements of 10 CFR 50.46(b)(5) and only requires operator action to [[ ]] after approximately seven days. The evaluation to demonstrate compliance considers the highest worth control rod pair associated with an individual HCU to be fully withdrawn and will be provided during future licensing activities. [[
]] The control rods, FMCRDs, and actuation systems ensure adequate shutdown margin, capability, redundancy, and diversity such that there is no need for combined reactivy control systems as required by GDC 27. [[

]]

Based on the above discussions, the special circumstance as specified in 10 CFR 50.12(a)(2)(ii) is present justifying an exemption to these specific requirements of 10 CFR 50 Appendix A, GDC 27. The application of the regulation in these particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. Instead, the following PDC 27 is proposed:

> PDC 27, Reactivity control system capability, the BWRX-300 reactivity control system shall be designed to have the capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

These statements of compliance and proposed PDC 27 may be used as the bases for the necessary exemption in future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant for requesting a CP and OL under 10 CFR 50 or a COL under 10 CFR 52.

4.0.11 10 CFR 50 Appendix A, GDC 28

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 28, Reactivity limits, requires that the reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity

accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Statement of Compliance: The combined features of the CRD system and the rod control system incorporate appropriate limits on the potential amount and rate of reactivity increase. The fine motion movement capability of the FMCRD allows reactivity additions from rod withdrawal to be limited. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worth. The BWRX-300 design prevents rod drop and rod ejection events through positive design means. Control rod drop is prevented by use of a bayonet style coupling, CRD mechanism latches, and CRD separation switches. Control rod ejection is prevented by physical constraints including the attachment of the control rod guide tube to the core plate and the CRD connection to the control rod guide tube. The FMCRD includes a brake that further prevents inadvertent rod withdrawal. The FMCRD also includes an internal ball check valve, which reduces the chances of rapid rod withdrawal. The ball check valve functions as a safety related component because it prevents reverse flow from the scram inlet port against the pressure and flow conditions caused by a break of the scram line, preventing the loss of pressure from the underside of the hollow piston and the generation of loads on the drive that could cause a rapid rod withdrawal and associated reactivity insertion. Normal rod movement and the rod withdrawal rate are limited by the FMCRD.

The rod control system controls rod patterns and provides control rod blocks to limit the rate and amount of reactivity addition for control rod movement. Compliance with GDC 28 was demonstrated in ESBWR Design Control Document Section 15.4.7.3 by analysis of the consequences of a postulated CRDA [Reference 5.2]. ESBWR Request for Additional Information (RAI) 4.6-23 (ADAMS Accession Nos. ML081090147 (Public) and ML081090148 (Non-Public)) summarized the results of the analysis [Reference 5.7]. In NUREG-1966, Volume 3, Section 15.4.7.3, the Staff Evaluation noted significant conservatism in the ESBWR analysis. There are no significant differences between the ESBWR approved analysis and the design and response of the BWRX-300.

The safety analyses to demonstrate compliance will be provided during future licensing activities and will evaluate PIEs and resulting transients or accidents including the steam line break and events that could change the reactor coolant temperature and pressure including cold water additions. The reactivity control systems include appropriate mitigating features for these events. The BWRX-300 CRD coupling is the same as the ESBWR. The CRDA event applied to an equilibrium cycle will be analyzed as a Special Event during future licensing activities using the approved Global Nuclear Fuels (GNF) CRDA Methodology (NEDE-33885P-A) [Reference 5.8]. Future licensing activities will confirm the expected similarities and support a conclusion that additional cycle-by-cycle CRDA evaluations are not warranted.

The design meets the requirements of GDC 28 by providing reactivity control systems features that mitigate postulated reactivity accidents that could result in damage to the

RCPB greater than limited local yielding or damage that impairs core cooling capability significantly.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 28.

4.0.12 10 CFR 50 Appendix A, GDC 29

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 29, Protection against anticipated operational occurrences, requires that the protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

  Statement of Compliance: The BWRX-300 rod control system enforces control rod withdrawal limits and prevents inappropriate control rod withdrawal. For events that result in AOOs such as a turbine trip, the [[ ]]. If the event reaches the RPS trip setpoints, a scram is enforced by the RPS. The combination of these system provides D-in-D protection for AOOs and more significant events. The RPS is Safety Class 1, safety-related equipment to ensure that high quality is achieved. The RPS and associated sensors and actuation devices are protected from natural phenomena and are designed as fail-safe to ensure that the safety function is maintained.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 29.

4.1 NUREG-0800 Standard Review Plan Guidance

4.1.1 Standard Review Plan 4.3

SRP 4.3, Nuclear Design, Rev. 3, states that the areas of review include confirmation that design bases are established as required by the appropriate GDC. Areas concerning core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality of the reactor during refueling, stability, analytical methods, and pressure vessel irradiation are to be reviewed. GDC specified in the SRP 4.3 acceptance criteria relevant to this LTR include GDC 12, GDC 20, GDC 25, GDC 26, GDC 27, and GDC 28. The BWRX-300 will meet the requirements of 10 CFR 50 Appendix A, GDC 12, GDC 20, GDC 25, GDC 26, and GDC 28 as described in Section 4.1, except for the exemption justification provided for GDC 27 and proposed PDC 27 in Section 4.1.10.

The areas of review, review interfaces, acceptance criteria, review procedures, evaluation findings, and references specified for evolutionary plants are applicable to the BWRX-300, [[

]] Therefore, GEH recommends that the existing SRP may be used during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

4.1.2 Standard Review Plan 7.2

SRP 7.2, Reactor Trip System, Rev. 6, states that the areas of review include confirming that the reactor trip system satisfies the requirements of the acceptance criteria and guidelines applicable to the protection system and performs its safety functions for all plant conditions under which the safety functions are required. GDC specified in the SRP 7.2 acceptance criteria relevant to this LTR include GDC 20, GDC 21, GDC 22, GDC 23, GDC 24, GDC 25, and GDC 29. The BWRX-300 will meet the requirements of 10 CFR 50 Appendix A, GDC 20, GDC 21, GDC 22, GDC 23, GDC 24, GDC 25, and GDC 29 as described in Section 4.1.

Therefore, the areas of review, review interfaces, acceptance criteria, review procedures, evaluation findings, and references are applicable to the BWRX-300 based on the design description and design requirements discussed in Sections 2.0 and 3.0 of this LTR for the RPS. Therefore, GEH recommends that the existing SRP may be used during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

4.1.3 Standard Review Plan 15.8

SRP 15.8, Anticipated Transients Without Scram, Rev. 2, states that certain light-water-cooled plants have prescribed systems and equipment that have been determined to reduce the risks attributable to ATWS events, for each of the nuclear steam supply system (NSSS) vendor's designs, to an acceptably low level. The rule also requires applicants to demonstrate the adequacy of their plants' prescribed systems and equipment.

10 CFR 50.62 was issued to address a generic safety issue potentially affecting current operating plants. The basis for the ATWS rule requirements, which are outlined in SECY-83-293, include the stipulation to reduce the risk of core damage due to an ATWS to be less than 10^{-5} per reactor year. The BWRX-300 will meet the stated risk goal of 10 CFR 50.62 by [[

]].

During DCA reviews of evolutionary plant designs, the NRC developed additional requirements criteria (i.e., [[ ]] or to demonstrate that the consequences of ATWS events are acceptable). The BWRX-300 is an evolutionary plant, and SRP 15.8 has designated acceptance criteria for evolutionary plants. The specific acceptance criteria for evolutionary plants is included in the guidance provided in SRP 15.8 (Items 3.A. through 3.C.).

SRP Acceptance Criteria:

A. For evolutionary plants where the ATWS rule does not explicitly require a diverse scram system, the applicant may provide either of the following:

   i. A diverse scram system satisfying the design and quality assurance criteria specified in SRP Section 7.2.

   ii. Demonstrate that the consequences of an ATWS event are within acceptable values.

BWRX-300 discussion:

The BWRX-300 includes an [[ ]]. Therefore, the BWRX-300 design conforms to this guidance.

SRP Acceptance Criteria:

B. For evolutionary plants, some of the equipment required to satisfy the rule may not apply. For example, passive BWRs do not have recirculation pumps; therefore, these designs cannot provide equipment to trip them as required by the rule. For these designs provision of an equivalent action such as reducing the vessel water level may be acceptable.

BWRX-300 discussion:

The specific equipment requirements described in 10 CFR 50.62, and the way the BWRX-300 addresses these requirements, are discussed in Section 4.1.1. This includes 10 CFR 50.62(c)(3) requiring an ARI system which is applicable to the BWRX-300 design, [[ ]] and 10 CFR 50.62(c)(5) requiring equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS which is not applicable to the BWRX-300 design. Therefore, the BWRX-300 design conforms to this guidance where applicable.

SRP Acceptance Criteria:

C. Applicants must demonstrate that the failure probability of failing the ATWS success criteria is sufficiently small because either: (1) the criteria are met, or [[ ]]. The analysis leading to the ATWS rule in NUREG-0460 used the following ATWS success criteria, which have their bases in the Commission regulations and GDC listed above. Applicant's design shall maintain:

i. Coolable geometry for the reactor core. If fuel and clad damage were to occur following a failure to scram, GDC 35 requires that this condition should not interfere with continued effective core cooling. 10 CFR 50.46 defines three specific core-coolability criteria: (1) Peak clad temperature shall not to exceed 1221°C (2200°F), (2) Maximum cladding oxidation shall not to exceed 17% the total cladding thickness before oxidation, and (3) Maximum hydrogen generation shall not to exceed 1% of the maximum hypothetical amount if all the fuel clad had reacted to produce hydrogen.

ii. Maintain reactor coolant pressure boundary integrity. Appendix A to WASH-1270 states that in evaluating the reactor coolant system boundary for ATWS events, "the calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the 'emergency conditions' as defined in the ASME Nuclear Power Plant Components Code, Section III." The acceptance criteria for reactor coolant pressure, based upon the ASME Service Level C limits, are approximately 10.3 MPa (1500 psig) for BWRs and approximately 22MPa (3200 psig) for Pressurized Water Reactors (PWRs).

iii. Maintain containment integrity. Following a failure to scram, the containment pressure and temperature must be maintained at acceptably low levels based on GDC 16 and 38. The containment pressure and temperature limits are design dependent; but to satisfy GDC 50, those limits must ensure that containment design leakage rates are not exceeded when subjected to the calculated pressure and temperature conditions resulting from any ATWS event.

BWRX-300 discussion:

> The BWRX-300 will meet the stated risk goal of 10 CFR 50.62 to reduce the risk of core damage due to an ATWS to be less than 10^{-5} per reactor year [[
>
>]]. As discussed in Section 3.7.1 and Section 4.1.1, these acceptance criteria are met through the defense in depth measures to insert control rods through hydraulically-driven or electric motor-driven means in combination with the [[
>]]. Therefore, the BWRX-300 design conforms to this guidance.

The areas of review, review interfaces, acceptance criteria, review procedures, evaluation findings, and references specified for evolutionary plants are applicable to the BWRX-300, with the exception of requiring [[ ]] and equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS which are not applicable to the BWRX-300 design, based on the design description and design requirements discussed in Sections 2.0 and 3.0 of this LTR. Therefore, GEH recommends that the existing SRP may be used during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

4.2 Generic Issues

The following generic issues do not represent the total listing required to support a 10 CFR 52 design certification application if pursued or for future 10 CFR 50 license applications but are provided based on their relevance to the scope of this LTR.

4.2.1 NUREG-1780

NUREG-1780, Regulatory Effectiveness of the Anticipated Transient Without Scram Rule, states that during the ATWS rulemaking the NRC staff set a goal that P(ATWS) should be no more than 1.0E-05/RY. As described in Section 4.1.1, the BWRX-300 will meet the stated risk goal of 10 CFR 50.62 as further defined in NUREG-1780 by [[

]].

5.0 REFERENCES

5.1 NEDC-33910P "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," June 2020.

5.2 26A66412AR, Rev 10, "ESBWR Design Control Document, Tier 2, Chapter 5 Reactor Coolant System and Connected Systems", GE Hitachi Nuclear Energy, April 2014.

5.3 International Atomic Energy Agency, "Safety of Nuclear Power Plants: Design," Specific Safety Requirements No. SSR-2/1, Rev. 1.

5.4 SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," USNRC, Washington, DC, July 19, 1983.

5.5 SECY-93-087, "Policy, Technical, And Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR Designs)," ADAMS Accession Number ML003708021, USNRC, Washington, DC, July 21, 1993.

5.6 NURGEG-1780, "Regulatory Effectiveness of the Anticipated Transient Without Scram Rule," USNRC, Washington, DC, September 2003.

5.7 General Electric Hitachi Letter, "Response to NRC Request for Additional Information Letters No. 115 and No. 137 – Related to ESBWR Design Certification Application – RAI Numbers 4.6-23 Supplement 2 and 4.6-38, Respectively," April 14, 2008 (ADAMS Accession Nos. ML081090147 (Public) and ML081090148 (Non Public)).

5.8 NEDE-33885P-A, Revision 1, Global Nuclear Fuels (GNF) CRDA Application Methodology, March 2020.

5.9 Letter from Rani Franovich (NRC) to Michelle Catts (GEH), "Final Safety Evaluation for GE-Hitachi Licensing Topical Report NEDC-33912P, Revision 0", "BWRX-300 Reactivity Control," ADAMS Accession Number ML21006A167, January 12, 2021.

5.10 General Electric Hitachi Letter, "Response to Request for Additional Information (eRAI) 9761 for Licensing Topical Report NEDC-33912P, Revision 0, BWRX-300 Reactivity Control," M200099, August 3, 2020.

Appendix A
GEH Response to NRC RAIs on NEDC-33912P-A Revision 0

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eRAI No.: 9761
Date of eRAI Issue: 07/17/2020

NONE-4

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (GDC) 28, "*Reactivity limits,*" states:

> The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Licensing Topical Report (LTR) NEDC-33912P, Revision 0, "BWRX-300 Reactivity Control," Section 4.1.11, "10 CFR 50 Appendix A, GDC 28," states:

> The BWRX-300 design prevents rod drop and rod ejection events through positive design means. Control rod drop is prevented by use of a bayonet style coupling, CRD mechanism latches, and CRD separation switches. Control rod ejection is prevented by physical constraints including the attachment of the control rod guide tube to the core plate and the CRD connection to the control rod guide tube.

While the regulation clearly includes the qualifying statement "unless prevented by positive means" in relation to the consideration of rod ejection, that same qualifier is not included for the consideration of rod dropout. The BWRX-300 statement of compliance is worded as if the "unless prevented by positive means" qualifier applied to both rod ejection and rod dropout, which is not the case in GDC 28.

Furthermore, GEH states in Section 3.7.2, "Control Rod Drop Accident:"

These BWRX-300 features prevent inadvertent uncoupling of the control rod from the FMCRD, block withdrawal of the drive assembly if separation of the hollow piston and ball nut occurs and latches a hollow piston that is not resting on the ball nut. Since these items do not allow a separation distance to occur between the FMCRD and the control rod or from the ball nut to an unlatched hollow piston, it is not possible for a control rod drop accident to occur.

The combination of this and the language from Section 4.1.11 above implies that GEH does not plan to analyze the control rod drop accident for the BWRX-300. This approach does not appear to be in compliance with GDC 28. Analysis of a control rod drop accident is safety significant to ensure the BWRX-300 design provides adequate reactivity control and prevents reactor power excursions.

Does GEH plan to request an exemption to GDC 28 under 10 CFR 50.12, "Specific exemptions," supported by a failure modes and effects analysis (FMEA) of the mechanisms and probabilities of the possible suite of scenarios that could lead to a control rod drop (including normal operation.

and during maintenance)? The NRC staff notes that an FMEA would also identify the next most limiting event sequence, currently bounded by the analysis of a control rod dropout, that would need to be analyzed for the BWRX-300.

Please confirm GEH's plans regarding the analysis of the control rod drop accident for the BWRX-300. If GEH plans to seek an exemption, please provide changes to LTR Section 4.1.11 to indicate that, and provide, or commit to provide during future licensing activities, an FMEA for the event and analysis of the most limiting event sequence that is credible.

GEH Response to NRC Question NONE-4

The Global Nuclear Fuels – Americas, LLC (GNF) Licensing Topical Report NEDE-33885-P-A, Revision 1, Control Rod Drop Accident Methodology (EPID L-2018-TOP-2006) was approved by the NRC on January 16, 2020, and accepted for use on March 27, 2020. This LTR demonstrates operating BWR compliance with the requirements of GDC 28 using the specified control rod drop accident (CRDA) methodology, and this methodology also applies to the BWRX-300. The BWRX-300 uses GNF-2 fuel, with a core design that is similar to the BWR operating fleet. The approved CRDA methodology will be applied to the BWRX-300 to demonstrate that cladding failures do not occur for the postulated (albeit incredible) CRDA. The results of the rod drop calculations will be discussed in the Probabilistic Risk Assessment (PRA) analysis that will be summarized in a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications. NEDC-33885P-A methodology provides additional assurance that the BWRX-300 design meets the requirements of GDC 28 by demonstrating that the CRDA acceptance criteria are met by the BWRX-300 core and fuel design.

Even if a rod drop were to occur, with the control rod decoupled and suspended because binding combined with a sheared coupling or hollow piston separated from the ball nut, the rod would only drop to the next latch position or in the case of a failed latch then to the ball nut position as determined by the rod control system and limited by the allowed withdrawal sequence. Note that any withdrawal after separation of the rod/hollow piston and ball nut would only occur if there was a failure of the control rod withdrawal block system. The control rod drive (CRD) design has been reviewed and approved for Advanced Boiling Water Reactor (ABWR) and Economically Simplified Boiling Water Reactor (ESBWR). The BWRX-300 CRD bayonet style coupling is the same as the ESBWR.

ESBWR Request for Additional Information (RAI) 4.6-23 ((ADAMS Accession Nos. ML081090147 (Public)/ML081090148 (Non-Public) summarized the results of the analysis in part:

> *Compliance with GDC 28 was demonstrated for the initial and an equilibrium ESBWR core by showing that the conservatively-calculated fuel enthalpy rises during CRDAs remain well below the lower bound clad failure limits in Appendix B of Revision 3 to SRP Section 4.2.*

Below these limits the integrity of the cladding can be assumed so assessment of hypothesized fuel dispersal and consequential energetic pressure pulses in the fluid are not required. The calculated enthalpy rises are conservative because they are based on adiabatic calculations that do not credit the reduction in enthalpy increase due to heat transfer or from the negative feedback caused by voiding in the coolant.

The ESBWR Final Safety Evaluation Report (FSER) NUREG-1966, Volume 3, Section 15.4.7.3, included analysis of the consequences of a postulated CRDA, noting significant conservatism in the analysis. The analyses demonstrated that specified acceptable fuel design limits are not exceeded. Because there is no fuel damage, the rod drop events do not result in any radiological release or degradation of the reactor coolant pressure boundary (RCPB).

No significant differences are expected between the BWRX-300 analyses and the analyses previously performed for either the ESBWR or for the operating BWRs using the approved methodology in NEDE-33885P-A. The methodology described in NEDE-33885P-A was designed to be flexible and accommodate all BWRs. The GNF-2 fuel used in the BWRX-300 is identical to fuel previously analyzed in BWR cores similar to the BWRX-300 core design. Results from a CRDA analysis for a BWRX-300 equilibrium cycle will be performed using the approved methodology in NEDE-33885P-A and documented in a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications, as a Special Event. These future licensing activities will confirm the expected similarities and support a conclusion that additional cycle-by-cycle CRDA evaluations are not warranted because of the significant safety margins in the BWRX-300 equilibrium cycle and initial cycle. The very low probability of a CRDA provides additional assurance that checks for future cycles are not necessary.

The safety analyses to demonstrate compliance of the BWRX-300 for other reactivity transients will be provided during future licensing activities and will evaluate postulated initiating event (PIEs) and resulting transients or accidents including the steam line break and events that could change the reactor coolant temperature and pressure, including cold water additions. The reactivity control systems include appropriate mitigating features for these events.

Based on the above discussions, the design meets the requirements of GDC 28 by providing reactivity control system features that mitigate postulated reactivity accidents that could result in damage to the RCPB greater than limited local yielding or damage that impairs core cooling capability significantly.

Proposed Changes to NEDC-33912P, Revision 0

NEDC-33912P, Revision 0, will be revised to reflect the following changes for BWRX-300 compliance to 10 CFR 50, Appendix A, GDC 28, and to add references to applicable ESBWR RAIs and the GNF LTR NEDE-33885P-A:

4.1.11 10CFR 50 Appendix A, GDC 28

Statement of Compliance: The combined features of the CRD system and the rod control system incorporate appropriate limits on the potential amount and rate of reactivity increase. The fine motion movement capability of the FMCRD allows limited reactivity additions from rod withdrawal. Control rod withdrawal sequences and patterns are selected to achieve

optimum core performance and low individual rod worth. The BWRX-300 design prevents rod drop and rod ejection events through positive design means. Control rod drop is prevented by use of a bayonet style coupling, CRD mechanism latches, and CRD separation switches. Control rod ejection is prevented by physical constraints including the attachment of the control rod guide tube to the core plate and the CRD connection to the control rod guide tube. The FMCRD includes a brake that further prevents inadvertent rod withdrawal. The FMCRD also includes an internal ball check valve, which reduces the chances of rapid rod withdrawal. The ball check valve functions as a safety-related component because it prevents reverse flow from the scram inlet port against the pressure and flow conditions caused by a break of the scram line, preventing the loss of pressure from the underside of the hollow piston and the generation of loads on the drive that could cause a rapid rodwithdrawal and associated reactivity insertion. Normal rod movement and the rod withdrawal rate are limited by the FMCRD.

The rod control system ~~enforces the~~ controls rod patterns and provides control rod blocks to limit the rate and amount of reactivity addition for control rod movement. Compliance with GDC 28 was demonstrated in ESBWR Design Control Document Section 15.4.7.3 by analysis of the consequences of a postulated CRDA [Reference 5.2]. ESBWR Request for Additional Information (RAI) 4.6-23 ((ADAMS Accession Nos. ML081090147 (Public)/ML081090148 (Non-Public) summarized the results of the analysis [Reference 5.7]. In NUREG-1966, Volume 3, Section 15.4.7.3, the Staff Evaluation noted significant conservatism in the ESBWR analysis. There are no significant differences between the ESBWR approved analysis and the design and response of the BWRX-300.

The safety analyses to demonstrate compliance will be provided during future licensing activities and will evaluate PIEs and resulting transients or accidents, including the steam line break and events that could change the reactor coolant temperature and pressure, such as cold water additions. The reactivity control systems include appropriate mitigating features for these events. The BWRX-300 CRD coupling is the same as the ESBWR. The CRDA event applied to an equilibrium cycle will be analyzed as a Special Event during future licensing activities using the approved GNF CRDA Methodology (NEDE-33885P-A) [Reference 5.8]. Future licensing activities will confirm the expected similarities and support a conclusion that additional cycle-by-cycle CRDA evaluations are not warranted.

The design meets the requirements of GDC 28 by providing reactivity control systems features that mitigate postulated reactivity accidents that could result in damage to the RCPB greater than limited local yielding or damage that impairs core cooling capability significantly.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 28.

5.0 REFERENCES

5.7 General Electric Hitachi Letter, "Response to NRC Request for Additional Information Letters No. 115 and No. 137 – Related to ESBWR Design Certification Application – RAI Numbers 4.6-23 Supplement 2 and 4.6-38, Respectively," April 14, 2008 (ADAMS Accession Nos. ML081090147 (Public)/ML081090148 (Non-Public)).

5.8 Global Nuclear Fuels – Americas, LLC, Licensing Topical Report NEDE-33885-P-A, Revision 1, Control Rod Drop Accident Methodology (EPID L-2018-TOP-2006), January 16, 2020.

eRAI No.: 9761
Date of eRAI Issue: 07/17/20

NRC Question NONE-5

10 CFR Part 50, GDC 26, "Reactivity control system redundancy and capability," requires:

> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Section 4.1.9, "10 CFR 50 Appendix A, GDC 26," of NEDC-33912P, Revision 0, states that the two BWRX-300 reactivity control systems credited for meeting GDC 26 are the hydraulic control units (HCUs) and the fine-motion control rod drives (FMCRDs). NEDC-33912P further states that the electrically-motor-driven control rod movement via the FMCRDs is a means of reactivity control that is independent of the HCU insertion system. The scram function of the HCUs is relied upon to satisfy the second sentence of GDC 26, while the fine motion of control rods using the FMCRDs is relied upon for the normal, planned power changes identified in the third sentence of GDC 26.

The NRC staff interprets "independent reactivity control systems of different design principles" in GDC 26 as meaning the systems would be independent and diverse; would not contain shared components; and would not be subject to common failure modes. However, the proposed BWRX-300 reactivity control systems rely on shared components (control blades) and therefore may also be subject to common failure modes. Therefore, it appears the BWRX-300 design would not comply with GDC 26 as written. Independence and diversity in the design and functionality of reactivity control systems is safety significant to ensure the BWRX-300 design provides adequate reactivity control and prevents reactor power excursions.

Based on a plain language read and the historical implementation of GDC 26, the staff requests GEH to provide justification why an exemption to GDC 26 is not needed for the BWRX-300. In addition, GEH has the option to specify in the LTR that an exemption would be sought in future licensing activities and propose an alternate principal design criterion in the LTR that the BWRX-300 design can satisfy.

GEH Response to NRC Question NONE-5

In BWRs, the second means of reactivity control is the ability to respond to power shape changes by changing core flow and recirculation ratio. The BWRX-300 does not employ forced circulation, but the core flow does change naturally in response to changes in reactivity and axial power shape in the same way as a jet-pump BWR that is operating in natural circulation. For

natural circulation, the core flow depends on the downcomer water level, and reducing the water level reduces core flow and thus core power. The BWRX-300 has the ability to reduce power via this mechanism. The Feedwater Level Control System is used to control feedwater flow and therefore controls reactor water level. This system can be used to make adjustments to water level when in normal power operation, and additional means are available for water level adjustments when in other modes of operation.

Proposed Changes to NEDC-33912P, Revision 0

NEDC-33912P, Revision 0, will be revised to describe the BWRX-300 compliance to the statement “independent reactivity control systems of different design principles” in 10 CFR 50, Appendix A, GDC 26:

4.1.9 10 CFR 50 Appendix A, GDC 26

. . .

Statement of Compliance: ~~Previous BWR designs used forced circulation, and controlled reactor power and reactivity during normal operations including normal startup and shutdown and to account for fuel burnup by means of establishing a set pattern for the hydraulic locking piston control rod drives and then controlling reactor recirculation pump speed or forced reactor recirculation flow. The BWRX-300 design uses natural circulation, and controls reactor power during normal operations including normal startup and shutdown and to account for fuel burnup by establishing a set pattern for specific rods using the electrically driven FMCRDs, and then by using fine motion control of specific rods using the electrically-driven control of the FMCRDs. Unlike the prior hydraulic locking piston control rod drives, the BWRX-300 FMCRDs are able to electrically move in very fine increments and therefore serve the same function of reactivity control as was performed by recirculation flow control. This fine movement is performed by a rod control system that includes rod pattern enforcement, control rod blocks, and an ability to monitor and control core thermal limits. Like the prior locking piston control rod drives, the BWRX-300 FMCRDs also include a hydraulic scram function that is a different design principle than the FMCRD electrically-driven normal control and electric motor run-in functions.~~

~~Like previous BWR designs, including the ABWR and ESBWR, the first of the two required independent reactivity control systems of different design principles is the hydraulic insertion of the control rods. A positive means for inserting the control rods is the highly reliable HCUs in both operating BWRs and the BWRX-300. In addition, the BWRX-300 has redundant and independent systems (~~[[ ]]~~, RPS, and ARI) to ensure that a hydraulic scram is initiated when required.~~

~~The control rod blades are capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, SAFDLs are not exceeded.~~

~~The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. This requirement is for~~

~~normal power changes (including xenon burnout), so the requirement is not referring to accident or shutdown scenarios. In PWRs, the second system is typically soluble boron which is slowly changed to control reactivity changes and manage axial power shape changes due to exposure and xenon effects, while the second system for BWRs has typically been the recirculation flow control system. BWRX-300 has the capability of electric motor-driven control rod movement that was not present in BWRs with only hydraulic control rod drive systems. Normal power changes in the BWRX-300 are made by the rod control system using fine motion control of specific control rods using the electric motor-driven control of the FMCRDs and by burnable poison, gadolinium, that is included in the fuel pellets in an axial and radial distribution within the core. The BWRX-300 rod control system and associated motor-driven control rod movement is a means of reactivity control that is independent of the hydraulic HCU insertion system. The BWRX-300 rod control system and associated motor-driven movement is capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded and therefore serves as the second means of reactivity control.~~

~~The requirement that one of the systems shall be capable of holding the reactor core subcritical under cold conditions is satisfied in operating BWRs and the BWRX-300 by the insertion of the control rod blades. The use of gadolinium burnable poison in the fuel pellets enables the control rods alone to have adequate shutdown margin. The evaluation to demonstrate compliance considers the highest worth control rod pair associated with an individual HCU to be fully withdrawn and will be provided during future licensing activities.~~

Two independent reactivity control systems of different design principles are provided.

Like previous BWR designs, including the ABWR and ESBWR, one of the two required independent reactivity control systems of different design principles is the hydraulic insertion of the control rods. A positive means for inserting the control rods is the highly reliable HCUs in both operating BWRs and the BWRX-300. In addition, the BWRX-300 has redundant and independent systems ([[ ]], RPS, and ARI) to ensure that a hydraulic scram is initiated when required.

Additionally, BWRX-300 has the capability of electric motor-driven control rod movement that was not present in operating BWRs with only hydraulic control rod drive systems. This capability also exists for ABWR and ESBWR. Normal power changes in the BWRX-300 are made by the rod control system using fine motion control of specific control rods using the electric motor-driven control of the FMCRDs and by burnable poison, gadolinium, that is included in the fuel pellets in an axial and radial distribution within the core. The BWRX-300 rod control system and associated motor-driven control rod movement is a means of reactivity control that uses an independent control and motive force than the hydraulic HCU insertion system.

The control blades are capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. Burnable gadolinium in the fuel and the FMCRDs provide for

slow reactivity adjustments. The ganged movement of blades via the motors and the scram function provide protection when needed for AOOs and design basis accidents (DBAs).

In BWRs, the second means of reactivity control is the ability to respond to power shape changes by changing core flow and recirculation ratio. In the operating BWR fleet, the recirculation flow is forced using either internal or external recirculation pumps. The BWRX-300 like the ESBWR does not employ forced circulation, but the core flow does change naturally in response to changes in reactivity and axial power shape in the same way as a jet-pump BWR that is operating in natural circulation. For natural circulation, the core flow depends on the downcomer water level, and reducing the water level reduces core flow and thus core power. The BWRX-300 has the ability to reduce power via this mechanism. The Feedwater Level Control System is used to control feedwater flow and therefore controls reactor water level. This system can be used to make adjustments to water level when in normal power operation, and additional means are available for water level adjustments when in other modes of operation.

One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

The requirement that one of the systems shall be capable of holding the reactor core subcritical under cold conditions is satisfied in operating BWRs and the BWRX-300 by the insertion of the control blades. The use of gadolinium burnable poison in the fuel pellets is a diverse means of reactivity control that controls the power profile at the beginning of core life and enables the control rods alone to have adequate shutdown margin. The evaluation to demonstrate compliance considers the highest worth control rod pair associated with an individual HCU to be fully withdrawn and will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 26.

Appendix B
Replaced Pages from NEDO-33912 Revision 0, Supplement 1

REVISION SUMMARY

Revision Number	Description of Change
0	Initial Issue
Supplement 1	Revised to incorporate the following responses to NRC Requests for Additional Information (eRAIs): • NRC eRAI 9761, Question NONE-4, revised Section 4.1.11 for BWRX-300 compliance to 10 CFR 50, Appendix A, GDC 28, and to add new References 5.7 and 5.8 for ESBWR Request for Additional Information (RAI) 4.6-23 (ADAMS Accession Nos. ML081090147 (Public) and ML081090148 (Non-Public) and NEDE-33885P-A, Revision 1, Global Nuclear Fuels (GNF) CRDA Application Methodology, respectively. • NRC eRAI 9761, Question NONE-5, revised Section 4.1.9 for BWRX-300 compliance to the statement "independent reactivity control systems of different design principles" in 10 CFR 50, Appendix A, GDC 26. • Revised Section 5 in response to NRC eRAI 9761, Question NONE-4, to add references to ESBWR RAIs and GNF LTR NEDE-33885P-A. • Corrected wording in the Purpose section to read: Design requirements are specified … and the backup means to automatically or manually insert control rods to ensure reactor shutdown. • Editorial correction in last sentence of Section 4.3.1. Changed wording from "[[ ]]" to "[[ ]]". • Information regarding the FMCRDs has been reclassified as non-proprietary and is identified with change bars in Sections 2.2.2.1, 3.5, 3.7.1 and 4.1.1.

Acronyms and Abbreviations

Term	Definition
ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ALWR	Advanced Light-Water Reactor
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ASME	American Society of Mechanical Engineers
[[	]]
ATWS	Anticipated Transient Without Scram
B&PV	Boiler & Pressure Vessel
BL-DBA	Baseline Design Basis Analysis
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CDF	Core Damage Frequency
CN-DBA	Conservative Design Basis Analysis
COL	Combined Operating License
CP	Construction Permit
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DCA	Design Certification Application
DEC	Design Extension Condition
DL	Defense Line
ECCS	Emergency Core Cooling System
ESBWR	Economically Simplified Boiling Water Reactor
EX-DBA	Extended Design Basis Analysis
FMCRD	Fine Motion Control Rod Drive

1.0 INTRODUCTION

1.1 Purpose

The purpose of this report is to provide the design requirements, acceptance criteria, and regulatory basis for the BWRX-300 reactivity control design functions, specifically for the following areas:

- Design requirements are specified for the Reactor Protection System (RPS) and the [[ ]] such that they satisfy the defense-in-depth (D-in-D) and diversity requirements to protect from common cause failure (CCF) of the RPS. Design requirements are also specified for other associated functions such as Alternate Rod Insertion (ARI) to ensure that the automatic hydraulic reactor scram will meet specified reliability requirements. The design of the RPS, [[ ]] and associated D-in-D features meet the requirements of 10 CFR 50.62 with justification provided for a [[ ]]. In addition, the requirements of 10 CFR 50 Appendix A, General Design Criteria (GDC) 12, GDC 20, GDC 21, GDC 22, GDC 23, GDC 24, GDC 25, GDC 26, GDC 28, and GDC 29 are met, with justification provided for a proposed exemption to the specific requirements of GDC 27 as proposed in Principal Design Criterion (PDC) 27.
- Design requirements are specified for the [[ ]] and the backup means to automatically or manually insert control rods to ensure reactor shutdown. [[ ]] These design features meet the requirements of 10 CFR 50.62 with the [[ ]]. In addition, the requirements of 10 CFR 50 Appendix A, GDC 12 and GDC 26 are met, with justification provided for PDC 27.

1.2 Scope

The scope of this report includes the following:

- A technical description of the BWRX-300 RPS, [[ ]] and other D-in-D design features and design functions to ensure the capability to shutdown the reactor and provide for control of reactivity and reactor fuel thermal limits, including acceptance criteria, regulatory bases, and references to existing proven design concepts based upon previous Boiling Water Reactor (BWR) designs, including the Advanced Boiling Water Reactor (ABWR) and Economically Simplified Boiling Water Reactor (ESBWR).
- A regulatory review of the BWRX-300 RPS, [[ ]] and other D-in-D design features and design functions to ensure the capability to shutdown the reactor and provide for control of reactivity and reactor fuel thermal limits, including acceptance criteria, to describe compliance with regulatory requirements and to describe the bases for any exemptions to regulatory requirements or alternative approaches to regulatory guidance that may be referenced in future licensing activities either by GEH in support of a 10 CFR 52 Design Certification Application (DCA) or by a license applicant for requesting a Construction Permit (CP) and Operating License (OL) under 10 CFR 50 or a Combined Operating License (COL) under 10 CFR 52.

Control Unit (HCU) is assumed to remain in the fully withdrawn position. A reactor core reload analysis is performed prior to every fuel cycle to define the core operating strategy for that cycle.

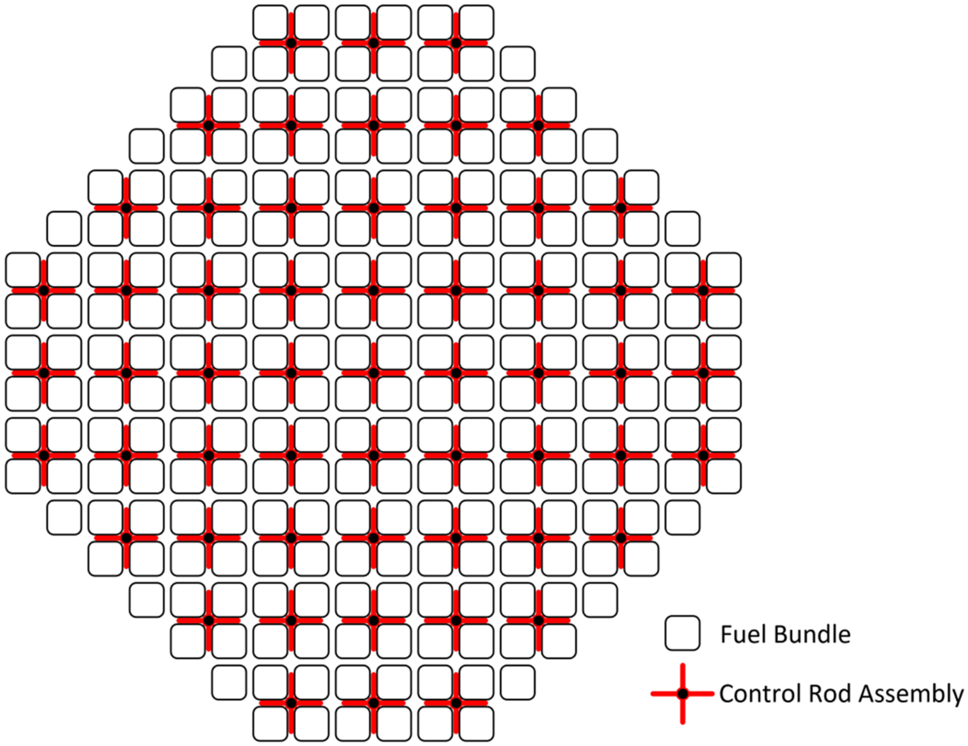

Figure 2-3: BWRX-300 Control Rod Locations Within Core

2.2.2 Control Rod Drives

Each Control Rod Assembly (i.e., control rod) is coupled to a Control Rod Drive (CRD) which is used to position the control rod.

Design Requirements:

- There are two diverse motive forces for the CRD and the associated control rod.
  - The control rods are normally positioned with an electric motor drive.
  - When a rapid shutdown is desired, the control rods are inserted hydraulically by use of high-pressure water.

The CRDs that are used for BWRX-300 are called FMCRDs to indicate the dual diverse means of movement as opposed to the predecessor locking piston hydraulically driven CRDs.

2.2.2.1 Fine Motion Control Rod Drive System

Design Requirements:

- During power operation, changes in core reactivity are controlled by movement and positioning of the control rods within the core, in fine increments, using FMCRD electric motors (one motor per control rod).
- The FMCRD motors also provide continuous run-in functionality to achieve shutdown.

- In the event of a postulated initiating event (PIE) that requires a rapid reactor shutdown, and if the reactor scram fails or is delayed, then the reactor is shut down by the electric motor run-in of FMCRDs function.
- [[

  ]]
- The FMCRDs include separation detection sensors that sense that the hollow piston along with the associated control rod are resting on the ball nut. These separation detection sensors actuate a control rod block signal.
- Rod withdrawal block signals prevent control rod withdrawal when required to enforce established control rod patterns.
- A rod withdrawal block signal prevents withdrawal of FMCRDs based upon an SRNM high period signal during startup.

The fine positioning and shutdown capabilities are achieved with a ball-nut and ball-screw arrangement driven by the FMCRD motor. The ball-nut is keyed to a guide tube to prevent its rotation and traverses through the guide tube vertically as the ball-screw is rotated. A hollow piston, connected to a control rod, rests on the ball-nut. The weight of the control rod keeps the hollow piston and ball-nut in contact during positioning in both insert and withdraw positioning. A schematic of the FMCRD including motor is illustrated in Figure 2-4.

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Figure 2-4: BWRX-300 FMCRD Schematic

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3.5 Defense Line 4

DL4 includes two subsets of functions, designated as DL4a and DL4b functions.

DL4a functions are those which can place and maintain the plant in a safe state in case of PIEs with failure of the DL3 functions. The DL4a functions should prevent the progression of accidents or radioactive release to the public. The need for DL4a functions generally arises from specific, postulated CCFs occurring in DL3. The effectiveness of DL4a functions is assessed in the EX-DBA, with the DL4a functions, and equipment performing those functions, subject to functional and design requirements derived from the EX-DBA.

DL4a functions are assigned to Safety Category 2 and performed by (at least) Safety Class 2 equipment as defined by IAEA guidance. These features are considered non-safety related in the US, but the appropriate reliability and quality control measures are included to ensure that they can be relied upon for D-in-D.

DL4b functions are those explicitly provided to prevent or mitigate an accident involving substantial melting of the nuclear fuel (i.e., a severe accident) while keeping radioactive releases to acceptable levels. DL4b also protects for events that exceed DL1 assumptions regarding PIEs as a result of extreme events, multiple events, or multiple failures.

DL4 features that are important for reactivity control include the following:

- ARI provides hydraulic scram in event of HCU actuation failure
- Electric motor run-in of FMCRDs

3.6 Defense Line 5

DL5 includes emergency preparedness measures to cope with potential unacceptable releases in case the first four defense lines are not effective. These are largely off-site measures taken to protect the public in a scenario involving substantial release of radiation.

DL5 measures are not in the scope of this LTR.

3.7 Specific Reactivity Control Events Considered in Defense-in-Depth Concept

The D-in-D approach that has been applied to the BWRX-300 results in elimination of select events from previous designs, reduction in the frequency of other events, and improved mitigation of events. This section provides a summary of select events. It is not considered an all-inclusive list of BWRX-300 events sequences.

3.7.1 Anticipated Transient Without Scram (ATWS)

As defined in 10 CFR 50.62, an ATWS is an AOO followed by a failure of the reactor trip system. Section 4.0 of this report includes the regulatory evaluation of compliance with the regulatory requirements of 10 CFR 50.62. This section describes the D-in-D features of the BWRX-300 that prevent or mitigate these specific reactivity control events.

- Peak cladding oxidation is within the requirements of 10 CFR 50.46;
- Peak containment pressure and temperature do not exceed containment design pressure and temperature;
- A coolable geometry is maintained; and
- Radiological releases are maintained within 10 CFR 100 allowable limits.

The BWRX-300 design requirements associated with ATWS prevention and mitigation include the following:

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  ]]
- RPS initiates a reactor scram based on signals and setpoints needed to support safety analysis credited trips.
- [[
-
-
  ]]
- ARI provides a diverse means to actuate the HCUs upon sensing a failure to scram.
- FMCRDs receive an electric motor run-in signal upon sensing a parameter requiring a scram.
- [[
  ]]
- FMCRD insertion time limits are established based on meeting the acceptance criteria of the safety analyses.

3.7.2 Control Rod Drop Accident

As with the ESBWR, the BWRX-300 has features to prevent a Control Rod Drop Accident (CRDA). The FMCRDs have a different coupling design to attach the drive assembly to the control rod than was used for the locking piston control rod drives. The FMCRD uses a bayonet style coupling that requires a 45-degree rotation to uncouple it. Since the FMCRD is firmly bolted into its position under the reactor vessel and the control rod is constrained from rotation by the fuel assemblies, it is not possible for the control rod to become uncoupled from the FMCRD during reactor operation. The hollow piston is the component within the FMCRD that is coupled to the control rod. The hollow piston normally rests on the ball nut internal to the FMCRD. There are dual separation detection devices that sense that the hollow piston along with the associated control rod are resting on the ball nut. If the sensor detects that the hollow piston is no longer on the ball nut, then control rod withdrawal is blocked. Additionally, the hollow piston has latches that prevent inadvertent withdrawal of the assembly when not attached to the ball nut.

]]. Additionally, the reactor can be shutdown by using the diverse FMCRD electric motor run-in. [[

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The BWRX-300 will meet the stated risk goal of 10 CFR 50.62 by providing a [[

]]. 10 CFR 50.62 was primarily issued to address a generic safety issue potentially affecting currently operating plants. The basis for ATWS rule requirements, which are outlined in SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," [Reference 5.4], which concluded that additional ATWS safety requirements were justified and included the stipulation to reduce the risk of core damage because of ATWS to be less than 10^{-5} per reactor year. [[

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Requirements for evolutionary plant designs beyond the original 10 CFR 50.62 regulatory requirements are addressed in more detail in SECY-93-087, "Policy, Technical, And Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR Designs)" [Reference 5.5] and addressed in SRP 15.8. The BWRX-300 design will meet the diversity and D-in-D guidelines for ATWS described in SECY-93-087, Item II.Q, Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems, without the use of an SLCS.

The effectiveness of the ATWS Rule requirements was evaluated in NUREG-1780 [Reference 5.6], which states that during the ATWS rulemaking the NRC staff set a goal that P(ATWS) should be no more than 1.0E-05/RY. P(ATWS) was defined as the annual frequency of an ATWS leading to plant conditions that exceed certain design parameters that can result in core melt, containment failure, and the release of radioactivity and can be viewed as the expected core damage frequency (CDF) of an unmitigated ATWS. Updating the original generic ATWS regulatory analysis, using operating data since the ATWS rule was implemented, found that on a generic basis, all four reactor types achieved the ATWS rule risk goal. The risk of core damage from a single CCF to scram is further reduced by reducing challenges to the RPS. NUREG-1780 notes that the initiating event frequency has been reduced by a factor of eight demonstrating that the Commission's recommendation to reduce the number of automatic reactor scrams has been very effective in reducing P(ATWS) (the probability of an ATWS). [[

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Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 24.

4.1.8 10 CFR 50 Appendix A, GDC 25

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 25, Protection system requirements for reactivity control malfunctions, requires that the protection system shall be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

  Statement of Compliance: The BWRX-300 protection system along with other mitigating design features assure that SAFDLs are not exceeded for reactivity control malfunctions. In order to prevent a rod withdrawal error, the rod control system has redundancy to limit the effect of single failures. Additionally, the rod patterns are enforced for rod withdrawals. If there is a malfunction of the rod control system that results in a rod withdrawal error during startup, a rod block is initiated based upon a SRNM high signal. If the rod withdrawal error were to result in a further increase to the SRNM based setpoint, a reactor scram will occur. If a rod withdrawal error were to occur at higher power, the APRM scram terminates the event if it were to continue to its setpoint. The analyses to demonstrate compliance will be provided during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 25.

4.1.9 10 CFR 50 Appendix A, GDC 26

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 26, Reactivity control system redundancy and capability, requires that two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, SAFDLs are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

  Statement of Compliance:

  Two independent reactivity control systems of different design principles are provided.

  Like previous BWR designs, including the ABWR and ESBWR, one of the two required independent reactivity control systems of different design principles is the hydraulic insertion of the control rods. A positive means for inserting the control rods is the highly reliable HCUs in both operating BWRs and the BWRX 300. In addition, the BWRX-300 has redundant and independent systems ([[ ]]), RPS, and ARI) to ensure that a hydraulic scram is initiated when required.

Additionally, BWRX-300 has the capability of electric motor-driven control rod movement that was not present in operating BWRs with only hydraulic control rod drive systems. This capability also exists for ABWR and ESBWR. Normal power changes in the BWRX-300 are made by the rod control system using fine motion control of specific control rods using the electric motor-driven control of the FMCRDs and by burnable poison, gadolinium, that is included in the fuel pellets in an axial and radial distribution within the core. The BWRX-300 rod control system and associated motor-driven control rod movement is a means of reactivity control that uses an independent control and motive force other than the hydraulic HCU insertion system.

The control blades are capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. Burnable gadolinium in the fuel and the FMCRDs provide for slow reactivity adjustments. The ganged movement of blades via the motors and the scram function provide protection when needed for AOOs and design basis accidents (DBAs).

In BWRs, the second means of reactivity control is the ability to respond to power shape changes by changing core flow and recirculation ratio. In the operating BWR fleet, the recirculation flow is forced using either internal or external recirculation pumps. The BWRX-300, like the ESBWR, does not employ forced circulation, but the core flow does change naturally in response to changes in reactivity and axial power shape in the same way as a jet pump BWR that is operating in natural circulation. For natural circulation, the core flow depends on the downcomer water level, and reducing the water level reduces core flow and thus core power. The BWRX-300 has the ability to reduce power via this mechanism. The Feedwater Level Control System is used to control feedwater flow and therefore controls reactor water level. This system can be used to make adjustments to water level when in normal power operation, and additional means are available for water level adjustments when in other modes of operation.

One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

The requirement that one of the systems shall be capable of holding the reactor core subcritical under cold conditions is satisfied in operating BWRs and the BWRX-300 by the insertion of the control blades. The use of gadolinium burnable poison in the fuel pellets is a diverse means of reactivity control that controls the power profile at the beginning of core life and enables the control rods alone to have adequate shutdown margin. The evaluation to demonstrate compliance considers the highest worth control rod pair associated with an individual HCU to be fully withdrawn and will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 26.

4.1.10 10 CFR 50 Appendix A, GDC 27

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 27, Combined reactivity control systems capability, requires that the reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling

accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Statement of Compliance: The combined features of the CRD system and the rod control system incorporate appropriate limits on the potential amount and rate of reactivity increase. The fine motion movement capability of the FMCRD allows reactivity additions from rod withdrawal to be limited. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worth. The BWRX-300 design prevents rod drop and rod ejection events through positive design means. Control rod drop is prevented by use of a bayonet style coupling, CRD mechanism latches, and CRD separation switches. Control rod ejection is prevented by physical constraints including the attachment of the control rod guide tube to the core plate and the CRD connection to the control rod guide tube. The FMCRD includes a brake that further prevents inadvertent rod withdrawal. The FMCRD also includes an internal ball check valve, which reduces the chances of rapid rod withdrawal. The ball check valve functions as a safety related component because it prevents reverse flow from the scram inlet port against the pressure and flow conditions caused by a break of the scram line, preventing the loss of pressure from the underside of the hollow piston and the generation of loads on the drive that could cause a rapid rod withdrawal and associated reactivity insertion. Normal rod movement and the rod withdrawal rate are limited by the FMCRD.

The rod control system controls rod patterns and provides control rod blocks to limit the rate and amount of reactivity addition for control rod movement. Compliance with GDC 28 was demonstrated in ESBWR Design Control Document Section 15.4.7.3 by analysis of the consequences of a postulated CRDA [Reference 5.2]. ESBWR Request for Additional Information (RAI) 4.6-23 (ADAMS Accession Nos. ML081090147 (Public) and ML081090148 (Non-Public)) summarized the results of the analysis [Reference 5.7]. In NUREG-1966, Volume 3, Section 15.4.7.3, the Staff Evaluation noted significant conservatism in the ESBWR analysis. There are no significant differences between the ESBWR approved analysis and the design and response of the BWRX-300.

The safety analyses to demonstrate compliance will be provided during future licensing activities and will evaluate PIEs and resulting transients or accidents including the steam line break and events that could change the reactor coolant temperature and pressure including cold water additions. The reactivity control systems include appropriate mitigating features for these events. The BWRX-300 CRD coupling is the same as the ESBWR. The CRDA event applied to an equilibrium cycle will be analyzed as a Special Event during future licensing activities using the approved Global Nuclear Fuels (GNF) CRDA Methodology (NEDE-33885P-A) [Reference 5.8]. Future licensing activities will confirm the expected similarities and support a conclusion that additional cycle-by-cycle CRDA evaluations are not warranted.

The design meets the requirements of GDC 28 by providing reactivity control systems features that mitigate postulated reactivity accidents that could result in damage to the

RCPB greater than limited local yielding or damage that impairs core cooling capability significantly.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 28.

4.1.12 10 CFR 50 Appendix A, GDC 29

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 29, Protection against anticipated operational occurrences, requires that the protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

  Statement of Compliance: The BWRX-300 rod control system enforces control rod withdrawal limits and prevents inappropriate control rod withdrawal. For events that result in AOOs such as a turbine trip, the [[ ]]. If the event reaches the RPS trip setpoints, a scram is enforced by the RPS. The combination of these system provides D-in-D protection for AOOs and more significant events. The RPS is Safety Class 1, safety-related equipment to ensure that high quality is achieved. The RPS and associated sensors and actuation devices are protected from natural phenomena and are designed as fail-safe to ensure that the safety function is maintained.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 29.

4.2 NUREG-0800 Standard Review Plan Guidance

4.2.1 Standard Review Plan 4.3

SRP 4.3, Nuclear Design, Rev. 3, states that the areas of review include confirmation that design bases are established as required by the appropriate GDC. Areas concerning core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality of the reactor during refueling, stability, analytical methods, and pressure vessel irradiation are to be reviewed. GDC specified in the SRP 4.3 acceptance criteria relevant to this LTR include GDC 12, GDC 20, GDC 25, GDC 26, GDC 27, and GDC 28. The BWRX-300 will meet the requirements of 10 CFR 50 Appendix A, GDC 12, GDC 20, GDC 25, GDC 26, and GDC 28 as described in Section 4.1, except for the exemption justification provided for GDC 27 and proposed PDC 27 in Section 4.1.10.

The areas of review, review interfaces, acceptance criteria, review procedures, evaluation findings, and references specified for evolutionary plants are applicable to the BWRX-300, [[

]] Therefore, GEH recommends that the existing SRP may be used during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

BWRX-300 discussion:

> The BWRX-300 will meet the stated risk goal of 10 CFR 50.62 to reduce the risk of core damage due to an ATWS to be less than 10^{-5} per reactor year [[
>
>]]. As discussed in Section 3.7.1 and Section 4.1.1, these acceptance criteria are met through the defense in depth measures to insert control rods through hydraulically-driven or electric motor-driven means in combination with the [[ ]]. Therefore, the BWRX-300 design conforms to this guidance.

The areas of review, review interfaces, acceptance criteria, review procedures, evaluation findings, and references specified for evolutionary plants are applicable to the BWRX-300, with the exception of requiring [[ ]] and equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS which are not applicable to the BWRX-300 design, based on the design description and design requirements discussed in Sections 2.0 and 3.0 of this LTR. Therefore, GEH recommends that the existing SRP may be used during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

4.3 Generic Issues

The following generic issues do not represent the total listing required to support a 10 CFR 52 design certification application if pursued or for future 10 CFR 50 license applications but are provided based on their relevance to the scope of this LTR.

4.3.1 NUREG-1780

NUREG-1780, Regulatory Effectiveness of the Anticipated Transient Without Scram Rule, states that during the ATWS rulemaking the NRC staff set a goal that P(ATWS) should be no more than 1.0E-05/RY. As described in Section 4.1.1, the BWRX-300 will meet the stated risk goal of 10 CFR 50.62 as further defined in NUREG-1780 by [[

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5.0 REFERENCES

5.1 NEDC-33910P, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection"

5.2 26A66412AR, Rev 10, "ESBWR Design Control Document, Tier 2, Chapter 5 Reactor Coolant System and Connected Systems", GE Hitachi Nuclear Energy, April 2014

5.3 International Atomic Energy Agency, "Safety of Nuclear Power Plants: Design," Specific Safety Requirements No. SSR-2/1, Rev. 1

5.4 SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," USNRC, Washington, DC, July 19, 1983

5.5 SECY-93-087, "Policy, Technical, And Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR Designs)," ADAMS Accession Number ML003708021, USNRC, Washington, DC, July 21, 1993

5.6 NURGEG-1780, "Regulatory Effectiveness of the Anticipated Transient Without Scram Rule," USNRC, Washington, DC, September 2003

5.7 General Electric Hitachi Letter, "Response to NRC Request for Additional Information Letters No. 115 and No. 137 – Related to ESBWR Design Certification Application – RAI Numbers 4.6-23 Supplement 2 and 4.6-38, Respectively," April 14, 2008 (ADAMS Accession Nos. ML081090147 (Public) and ML081090148 (Non Public))

5.8 NEDE-33885P-A, Revision 1, Global Nuclear Fuels (GNF) CRDA Application Methodology, March 2020