ENCLOSURE 1

M230080

NEDO-33910-A, Revision 2,
BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection

Non-Proprietary Information

GE Hitachi Nuclear Energy

NEDO-33910-A
Revision 2
June 2021

Non-Proprietary Information

Licensing Topical Report

BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection

INFORMATION NOTICE

This is a non-proprietary version of the GE Hitachi Nuclear Energy (GEH) document NEDC-33910P-A Revision 2, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

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

November, 18,2020

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-33910P, REVISION 0, "BWRX-300 REACTOR PRESSURE VESSEL ISOLATION AND OVERPRESSURE PROTECTION"

Dear Ms. Catts:

By letter dated December 30, 2019, GE-Hitachi Nuclear Energy Americas, LLC (GEH), submitted Licensing Topical Report (TR) NEDC-33910P, Revision 0, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19364A209), 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 June 4, 2020 (ADAMS Accession No. ML20156A270), GEH submitted Revision 0, Supplement 1 of the TR. By letter dated June 20, 2020 (ADAMS Accession No ML20174A574), GEH submitted Revision 0, Supplement 2 of the TR.

The NRC staff has found TR NEDC-33910P, Revision 0, as updated by the June 20, 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 TR.

In accordance with the guidance provided on the NRC's TR website (http://www.nrc.gov/about-nrc/regulatory/licensing/topical-reports.html), the NRC requests that GEH publish an accepted version of this TR 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 TR 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 TR is acceptable) is invalidated, GEH and/or an applicant referencing the TR will be expected to revise and resubmit its respective documentation or submit justification for the continued applicability of the TR 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-33910P ListServ

SUBJECT: FINAL SAFETY EVALUATION FOR GE-HITACHI LICENSING TOPICAL REPORT NEDC-33910P, REVISION 0, “BWRX-300 REACTOR PRESSURE VESSEL ISOLATION AND OVERPRESSURE PROTECTION”
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT NEDC-33910P, REVISION 0, SUPPLEMENT 2

BWRX-300 REACTOR PRESSURE VESSEL ISOLATION AND OVERPRESSURE PROTECTION

GE-HITACHI NUCLEAR ENERGY

DOCKET NUMBER 99900003

1.0 Introduction

The purpose of GE-Hitachi Nuclear Energy Americas, LLC (GEH) Licensing Topical Report (LTR) NEDC-33910 (Revision 0, Supplement 2), "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," submitted June 22, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20174A574), is to describe the design requirements, acceptance criteria, and regulatory bases for the design functions of the reactor pressure vessel (RPV) isolation and overpressure protection for the BWRX-300 small modular reactor (SMR). Specifically, the LTR addresses the following:

- Design requirements are specified by GEH for the RPV isolation valves and configuration with the function of closing to limit the loss of coolant from large and medium pipe breaks, and design requirements are specified for automatic actuation of the isolation condenser system (ICS) to remove decay heat from large, medium, and small pipe breaks, to meet the acceptance criteria in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46(b).

- Design requirements are specified by GEH for the reactor protection system (RPS) and ICS for overpressure protection.

NEDC-33910 also includes the following:

- A technical evaluation of the BWRX-300 RPV isolation and overpressure protection design features and design functions, acceptance criteria, regulatory bases, and references to existing proven design concepts based on previous boiling water reactor (BWR) designs.

- A discussion of the BWRX-300 RPV isolation and overpressure protection design features and design functions that focuses on compliance with regulatory requirements and describes the bases for any regulatory requirements, as well as alternative approaches to methods described in regulatory guidance that may be referenced in future licensing activities.

In this safety evaluation, the U.S. Nuclear Regulatory Commission (NRC) staff describes its review of NEDC-33910 and the acceptability of LTR provisions for the design of the RPV isolation and overpressure protection for the BWRX-300 SMR. In response to NRC staff requests for additional information, GEH submitted a response dated April 20, 2020 (ADAMS Accession No. ML20111C944), and a response dated May 4, 2020 (ADAMS Accession

No. ML20125D893). The NRC staff will evaluate the compliance of the final design of the RPV isolation and overpressure protection features for the BWRX-300 SMR during future licensing activities in accordance with 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 safety evaluation, double brackets indicate potentially proprietary information.

2.0 Technical Evaluation of Reactor Pressure Vessel Isolation

2.1 General Introduction

2.1.1 Reactor Pressure Vessel

Section 2.1.1, “Reactor Pressure Vessel,” in NEDC-33910 describes the RPV for the BWRX-300 SMR. The RPV is a vertical, cylindrical pressure vessel fabricated with rings and rolled plate welded together. The top head is removable by use of a head flange, seals, and bolting. The RPV design permits natural circulation driving forces to produce reactor core coolant flow.

2.1.2 Isolation Condenser System

Section 2.1.2, “Isolation Condenser System,” in NEDC-33910 describes the ICS for the BWRX-300 SMR. The ICS includes three trains with two parallel isolation condenser (IC) condensate return valves in each train.

In the letter dated April 20, 2020, GEH stated that the detailed design of the IC condensate return valves was not yet complete. However, GEH indicated that it expects the design functions and features of the IC condensate return valves to be similar to applicable valves in the Economic Simplified Boiling Water Reactor (ESBWR). The NRC certified the ESBWR design in 10 CFR Part 52, Appendix E, “Design Certification Rule for the ESBWR Design.” GEH specified that compliance with the requirements of 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion (GDC) 1, “Quality standards and records,” GDC 2, “Design bases for protection against natural phenomena,” GDC 4, “Environmental and dynamic effects design bases,” and GDC 37, “Testing of emergency core cooling system,” for the IC condensate return valves is anticipated to be the same as described in the ESBWR Design Control Document (DCD). At this time, GEH proposed that the limited design requirements specific for the ICS, including the IC condensate return valves, be found acceptable for ensuring that the ICS can perform the functions in demonstrating compliance with 10 CFR Part 50, Appendix A, GDC 35, “Emergency core cooling.” GEH described the specific functions of the IC condensate return valves. For example, GEH specified that qualification of the IC condensate return valves, such as compliance with American Society of Mechanical Engineers (ASME) Standard QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants” (or a later edition), as accepted in NRC Regulatory Guide (RG) 1.100, Revision 3, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” issued September 2009, will be addressed in the detailed design of the valves. In addition, GEH specified a design requirement that the actuators for the IC condensate return valves will be designed to maintain the valves in their open position by positive mechanical means.

The NRC staff finds that the IC condensate return valve concept as described in NEDC-33910 is consistent with the requirements in 10 CFR Part 50, Appendix A, GDC 35, and therefore is acceptable. The NRC staff will perform a detailed evaluation to confirm that the final design of the RPV isolation valves satisfies regulatory requirements in 10 CFR Part 50, Appendix A, GDC 35 when an application for a BWRX-300 SMR is received.

2.2 General Overview of the Reactor Pressure Vessel Isolation Concept

Section 2.2, “General Overview of the Reactor Pressure Vessel Isolation Concept,” in NEDC-33910 provides a general overview of the RPV isolation concept, and Section 2.5, “Reactor Pressure Vessel Isolation Valve Design Requirements,” specifies the RPV isolation valve design requirements for the BWRX-300 SMR. They indicate that there will be two RPV isolation valves in series in the applicable RPV piping. These sections specify that [[

]], will make up the series RPV isolation valves in each piping line. According to these sections, all RPV isolation valves will be designed to have a proven low-leakage potential. These sections state that the RPV isolation valves for main steam, feedwater, shutdown cooling, and the reactor water cleanup system will be designed to fail closed. These sections indicate that the RPV isolation valves for IC steam supply and condensate return will be designed to fail as-is. Further, the RPV isolation valves with automatic closure requirements will be designed to rely on Class 1E battery-backed direct current power.

In response to NRC staff questions, GEH stated in its letters dated April 20, 2020, and May 4, 2020, that the design of the [[

]]. GEH stated that NEDC-33911P, “BWRX-300 Containment Performance,” submitted March 31, 2020 (ADAMS Accession No. ML20091S367), presents further details on the [[

]]. GEH included a revised Section 2.4.1, “Connection of Reactor Pressure Vessel Isolation Valves to Reactor Vessel,” in NEDC-33910 to provide additional details on this connection.

The NRC staff finds that, based on the description in NEDC-33910, together with its reference to NEDC-33911, the RPV isolation valve concept is consistent with 10 CFR Part 50, Appendix A, GDC 30, “Quality of reactor coolant pressure boundary,” and GDC 35 and is therefore acceptable. The NRC staff will perform a detailed evaluation to confirm that the final design of the RPV isolation valves satisfies regulatory requirements when an application for a BWRX-300 SMR is received.

2.3 Reactor Pressure Vessel Design Requirements

Section 2.3, “Reactor Pressure Vessel Design Requirements,” in NEDC-33910 specifies that the BWRX-300 RPV is designed using the same codes and standards as the ESBWR RPV and with similar selection of design-code-accepted material specifications as described in the ESBWR DCD. Section 2.3 states that changes as a result of newer editions or published revisions of codes and standards used in the BWRX-300 design are subject to appropriate regulatory review and approval. The section also reports that the full details of the material specifications and codes and standards for the BWRX-300 SMR will be provided during future licensing activities. The NRC staff will perform a detailed evaluation to verify the final design of the RPV satisfies applicable regulatory requirements when an application for a BWRX-300 SMR is received.

2.4 Reactor Pressure Vessel Nozzle Design Requirements

Section 2.4.1 of NEDC-33910 states that the [[ ]]. The section also states that [[ ]]. The NRC staff will review the specific features of the RPV nozzle design during future licensing activities for the BWRX-300 SMR.

2.4.1 Connection of Reactor Pressure Vessel Isolation Valves to Reactor Vessel

Section 2.4.1 of NEDC-33910 states that [[ ]], extending to the containment wall, the BWRX-300 design requirements include identifying postulated pipe rupture locations and configurations inside containment as specified in NRC Branch Technical Position (BTP) 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment,” Part B, Item 1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2). Where break locations are selected without the benefit of stress calculations, breaks are postulated at the piping welds to each fitting, valve, or welded attachment.

In response to NRC staff questions, GEH revised Section 2.4.1 to specify that [[ ]]. Section 2.4.1 also indicates that these [[ ]] will be established during detailed design of the [[ ]] and provided during future licensing activities.

The NRC staff will review the specific aspects of the connection of the RPV isolation valves to the reactor vessel during future licensing activities of the BWRX-300 SMR.

2.5 Reactor Pressure Vessel Isolation Valve Design Requirements

In its letter dated April 20, 2020, GEH stated that the detailed design of the RPV isolation valve assemblies has not yet been completed. GEH indicated that compliance with the requirements of 10 CFR Part 50, Appendix A, GDC 1, 2, and 4 for the RPV isolation valve assemblies is expected to be the same as for other ASME *Boiler and Pressure Vessel Code* (BPV Code) Class 1 valves described in the ESBWR DCD. GEH specified that it is proposing that the design requirements for the RPV isolation valve assemblies be found acceptable for ensuring that the functions are met in demonstrating compliance with 10 CFR Part 50, Appendix A, GDC 33, “Reactor coolant makeup,” and GDC 35. GEH described the design features for the RPV isolation valves. For example, GEH specified that the detailed design of the valves will address qualification of the RPV isolation valves, such as compliance with ASME Standard QME-1-2007 (or a later edition) as accepted in RG 1.100.

NEDC-33910 describes instrumentation and control (I&C) information related to ICS RPV isolation actuation logic. The acceptability of specific I&C signals and logic described in NEDC-33910 is outside the scope of this LTR and will be reviewed during future licensing activities.

The NRC staff finds that the RPV isolation valve design features as described in NEDC-33910 are consistent with 10 CFR Part 50, Appendix A, GDC 33 and 35, and are therefore acceptable. The NRC staff will perform a detailed evaluation to confirm that the final design of the RPV isolation valves satisfies 10 CFR Part 50, Appendix A, GDC 33 and 35, when an application for a BWRX-300 SMR is received.

2.6 Design Requirements for the Reactor Pressure Vessel Isolation Valve Actuator

Section 2.6, "Reactor Pressure Vessel Isolation Valve Actuator Design Requirements," in NEDC-33910 specifies the RPV isolation valve actuator design requirements for the BWRX-300 SMR. This section specifies that the valve and actuator designs will be qualified using ASME Standard QME-1. This section refers to several aspects to be considered as part of the design requirements for the RPV isolation valve actuator.

In its letter dated April 20, 2020, GEH stated that the detailed design of the RPV isolation valve actuators is not complete. GEH indicated that compliance with the requirements of 10 CFR Part 50, Appendix A, GDC 1, 2, and 4, for the RPV isolation valve actuators is expected to be the same as for actuators for other ASME BPV Code Class 1 valves described in the ESBWR DCD. GEH specified that it is proposing that the design requirements for the RPV isolation valve actuators be found acceptable for ensuring that the functions are met in demonstrating compliance with 10 CFR Part 50, Appendix A, GDC 33 and 35. GEH described the design features for the RPV isolation valve actuators. For example, GEH specified that the detailed design of the valves will address qualification of the RPV isolation valve actuators, such as compliance with ASME Standard QME-1-2007 (or a later edition) as accepted in RG 1.100. In addition, GEH specified design requirements for the RPV isolation valves that the valve actuators will maintain their valves in the applicable positions by positive mechanical means.

The NRC staff finds that the RPV isolation valve actuators concept, as described in NEDC-33910, is consistent with 10 CFR Part 50, Appendix A, GDC 33 and 35, and is therefore acceptable. The NRC staff will perform a detailed evaluation to confirm that the final design of the RPV isolation valve actuator satisfies 10 CFR Part 50, Appendix A, GDC 33 and 35, when an application for a BWRX-300 SMR is received.

2.7 Categories of Pipe Breaks

Section 2.7, "Categories of Pipe Breaks," specifies the two categories of steam and liquid line breaks that will be considered in the safety analysis and the associated design requirements for the BWRX-300.

The categories described in the topical report cover the full range of postulated break sizes. In addition, the design requirements are consistent with the Commission's regulations related to loss-of-coolant accident (LOCA) mitigation and emergency core cooling system (ECCS) design. The topical report specifies that the BWRX-300 ECCS will be demonstrated to respond in the event of a pipe break to maintain the appropriate design requirements. The analyses supporting these statements will be provided during future licensing activities related to the design.

The two categories of break sizes specified in the topical report differentiate between larger pipes with isolation valves and smaller pipes without isolation valves (e.g., the differential pressure instrument lines). The key concept related to showing these design requirements are met for the larger isolated lines is a [[

]], to maintain reactor water level at or above the top of active fuel (TAF) or fuel cladding temperature within normal operating temperature range.

There are two key concepts used in the topical report to ensure the design requirements are met for the [[

]], such that the reactor water level is maintained at or above the TAF or the fuel cladding temperature is maintained within normal operating temperature range.

Reactor water level and fuel cladding temperature are maintained for at least [[ ]] for both categories of break sizes through operation of the BWRX-300 ECCS. The NRC staff considers that inclusion of the small unisolated breaks within the LOCA spectrum specified in the LTR for analysis in the absence of injection or makeup flow ensures that the largest unisolated mass and energy release from a pipe break will be analyzed as part of the LOCA break spectrum and demonstrated to meet the LOCA criteria specified for at least [[ ]].

Small leaks that are within the capability of the nonsafety-related high-pressure control rod drive (CRD) injection system and therefore not considered LOCAs by the definition in 10 CFR Part 50, will be evaluated to ensure that the specified acceptable fuel design limits (SAFDLs) are not exceeded, consistent with GDC 33.

The NRC staff finds that the categories of pipe breaks and the associated design requirements for the BWRX-300, as described in NEDC-33910, are consistent with 10 CFR 50.46(b). The NRC staff will perform a detailed evaluation to confirm 10 CFR 50.46(b) is met when an application for a BWRX-300 SMR is received.

2.8 Loss-of-Coolant Accident Acceptance Criteria

Section 2.8, "LOCA Acceptance Criteria," specifies that the BWRX-300 SMR will show that the provisions of 10 CFR 50.46 are met by ensuring that the more stringent acceptance criteria of reactor water level maintained at or above the TAF or fuel cladding temperature maintained within normal operating temperature range. Either of these acceptance criteria will ensure that the prescriptive criteria outlined in 10 CFR 50.46(b) are not violated; Section 4.1.1 of this safety evaluation report provides more detail.

The NRC staff finds this approach acceptable to demonstrate that 10 CFR 50.46(b) will be met because GEH intends to ensure that the more stringent acceptance criteria for reactor water levels are maintained. The NRC staff will perform a detailed evaluation to confirm 10 CFR 50.46(b) is met when an application for a BWRX-300 SMR is received.

3.0 Technical Evaluation of Overpressure Protection

3.1 General Overview of the Overpressure Protection Concept

Section 3.1, "General Overview of the Overpressure Protection Concept," specifies that the BWRX-300 integrated overpressure protection during operation at power is ensured by application of the RPS to shut down the reactor, in combination with heat removal through the ICS to control RPV pressure. Section 3.1 indicates that [[

]].

3.1.1 Reactor Protection System Design Requirements

Section 3.1.1, “Reactor Protection System Design Requirements,” explains at a high level the function of the BWRX-300 reactor protection system (RPS). The design is based on the previously approved ESBWR and is a safety-related system to control reactivity for overpressure protection. The topical report denotes specific design requirements for the system that, combined with regulations and guidance prescriptive to the design of the RPS, will be implemented to ensure the safe operation of the BWRX-300 design.

The NRC staff finds that the RPS for overpressure protection, as described in NEDC-33910 and previously approved for ESBWR, is consistent with the requirements of GDC 15, “Reactor coolant system design,” and is therefore acceptable. The NRC staff will perform a detailed evaluation to confirm GDC 15 is met when an application for a BWRX-300 SMR is received.

3.1.2 Isolation Condenser System Design Requirements

Section 3.1.2, “Isolation Condenser System Design Requirements,” describes the design and function of the BWRX-300 isolation condenser system (ICS) as a safety-related system to remove decay heat passively following a reactor shutdown and isolation. The topical report specifies design requirements for the ICS to confirm that the reactor core can be adequately cooled, ensuring that overpressure protection design requirements are met and that the ICS will be single-failure proof.

The NRC staff finds that the ICS for overpressure protection, as described in NEDC-33910, is consistent with the requirements of GDC 30, “Quality of reactor coolant pressure boundary,” and GDC 31, “Fracture prevention of reactor coolant pressure boundary,” and is therefore acceptable. The NRC staff will perform a detailed evaluation to confirm GDCs 30 and 31 are met when an application for a BWRX-300 SMR is received.

3.2 ASME Requirements for Overpressure Protection

Section 3.2, “ASME Requirements for Overpressure Protection,” in NEDC-33910 specifies that overpressure protection for the reactor coolant pressure boundary (RCPB) complies with ASME BPV Code, Section III, “Rules for Construction of Nuclear Facility Components,” Article NB-7000, “Overpressure Protection.” Section 3.2 characterizes paragraph NB-7120, “Integrated Overpressure Protection,” as requiring that overpressure protection of the components shall be provided by any of the following as integrated overpressure protection:

- The use of pressure relief devices and associated pressure sensing elements.
- The use of reactor shutdown system.
- A design without pressure relief devices such that for each component in the protected system
    - The overpressure does not exceed 1.1 times design pressure for the expected system pressure transient condition.
    - The calculated stress intensity and other design limitations for Service Level C are not exceeded for the unexpected system excess pressure transient condition.

Section 3.2 of NEDC-33910 states that overpressure protection for the BWRX-300 SMR is provided in accordance with ASME BPV Code, Section III, Article NB-7000, subparagraphs [[

]].

The NRC staff finds that providing overpressure protection for the RCPB of the BWRX-300 SMR by implementing the provisions in ASME BPV Code, Section III, Article NB-7000, as incorporated by reference in 10 CFR 50.55a, "Codes and standards," is acceptable. Compliance with the latest edition and addenda of the ASME BPV Code incorporated by reference in 10 CFR 50.55a will need to be demonstrated as part of future licensing activities for the BWRX-300 SMR.

4.0 Regulatory Evaluation

4.1 10 CFR Part 50 Regulations

Section 4.1, "10 CFR 50 Regulations," of NEDC-33910 provides statements of compliance for the regulations in 10 CFR Part 50 applicable to the RPV isolation and overpressure protection features of the BWRX-300 SMR.

In response to the NRC staff's questions, GEH revised NEDC-33910 to include additional regulations applicable to the BWRX-300 SMR. For each regulation, NEDC-33910 describes the intent to meet those design requirements for the BWRX-300 SMR. In some instances, NEDC-33910 indicates that specific design requirements for the BWRX-300 components will be provided during future licensing activities.

The subsections below provide the staff's evaluation of the preliminary design information related to each regulation, and additional evaluation will be provided during future licensing activities, if needed.

4.1.1 10 CFR 50.34(f) Additional Requirements Related to Three Mile Island

In evaluating whether the items in 10 CFR 50.34(f) are technically relevant to the BWRX-300 reactor design, the NRC staff consulted NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," issued November 1980 (ADAMS Accession No. ML051400209). This NUREG contains additional information and background on the requirements, which assisted the staff in understanding their basis and intent in the evaluation of their technical relevancy as applied to the BWRX-300 design.

4.1.1.1 10 CFR 50.34(f)(1)(v)

This regulation requires performance of an evaluation of the safety effectiveness of separating the high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high-pressure core spray systems in lieu of HPCI systems, substitute the words, "high-pressure core spray" for "high-pressure coolant injection" and "HPCS" for "HPCI.") (Applicable to BWRs only.)

According to GEH, the BWRX-300 design does not include any safety-related high-pressure injection systems as specified in the regulation as their functions are unnecessary given the design's unique approach to mitigating LOCAs. In addition, the initiation signals for the safety systems that mitigate the effects of LOCAs are separate and use different process variables. GEH asserts that an exemption from 10 CFR 50.34(f) is justified because the design meets the special circumstances specified in 10 CFR 50.12(a)(2)(ii). Additionally, GEH indicates that the statements in the TR "may be used as the bases for the necessary partial or full exemptions in future licensing activities."

The intent of this requirement, as described in NUREG-0737, is for BWR plant designs to modify their ECCS initiation levels such that RCIC initiates at a higher water level than HPCI. The logic behind this is to minimize the potential number of challenges due to system initiation on HPCI and to reduce the stress on the reactor vessel due to HPCI's cold water injection.

GEH states that the BWRX-300 design does not contain HPCI and RCIC systems, nor does it rely on any safety-related injection systems as part of its design basis. In addition, the BWRX-300 uses [[ ]], and the ICS does not contain moving parts other than one-time actuation of valves and does not inject cold water into the vessel (except during the sort time needed to obtain steady IC loop flow).

Therefore, subject to confirmation during licensing activities for a final design, the NRC staff finds that this requirement, based on the plain language reading of the regulation combined with the additional information in NUREG-0737, may not be technically relevant to the BWRX-300 design. The information GEH has provided in NEDC-33910 could support an exemption request when an application is submitted.

4.1.1.2 10 CFR 50.34(f)(1)(vi)

This regulation requires performance of a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWRs only.)

According to GEH, the BWRX-300 unique design obviates the need to perform this study.

The intent of this requirement, as described in NUREG-0737, is for BWR plant designs to implement actions to reduce the challenges for relief valves, as operating experience at the time showed high failure rates of these valves.

GEH states that the BWRX-300 design [[ ]] uses the ICS to reduce system pressure during design-basis events.

Therefore, subject to confirmation during licensing activities for a final design, the NRC staff finds that this requirement, based on the plain language reading of the regulation combined with the additional information in NUREG-0737, may not be technically relevant to the BWRX-300 design. The information GEH has provided in NEDC-33910 could support an exemption request when an application is submitted.

4.1.1.3 10 CFR 50.34(f)(1)(vii)

This regulation requires the performance of a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWRs only.) GEH does not plan to credit any manual actuation of the BWRX-300 safety systems to mitigate the effects of a LOCA, therefore obviating the need to perform this assessment.

The intent of this requirement, as described in NUREG-0737, is for BWR plant designs to modify ADS actuation logic to eliminate the need for manual actuation to ensure adequate core cooling. The NUREG advises that additional actuation logic be added to the ADS to complement the existing actuation signals.

GEH states that the BWRX-300 design [[ ]]. The systems that ensure adequate core cooling actuate automatically, do not rely on manual activation, and are single-failure proof, ensuring that the core will be adequately cooled during design-basis events.

Therefore, subject to confirmation during licensing activities for a final design, the NRC staff finds that this requirement, based on the plain language reading of the regulation combined with the additional information in NUREG-0737, may not be technically relevant to the BWRX-300 design. The information GEH has provided in NEDC-33910 could support an exemption request when an application is submitted.

4.1.1.4 10 CFR 50.34(f)(1)(viii)

This regulation requires performance of a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low-pressure coolant injection (LPCI) systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (Applicable to BWRs only.)

According to GEH, the BWRX-300 design does not include core spray or any safety-related LPCI systems; other systems are credited to mitigate the effects of a LOCA. Actuation of these safety-related systems (e.g., the ICS) is a one-time action and does not rely on the use of active pumps. Therefore, the study is not required.

The intent of this requirement, as described in NUREG-0737, is for BWR plant designs to modify the core spray and LPCI system logic such that the systems will restart automatically, if needed to ensure adequate core cooling, following a manual stoppage by the operators.

GEH also states that the BWRX-300 design does not contain core spray or any other safety-related LPCI systems as described or envisioned by the rule. However, the ICS is the safety-related system that ensures adequate core cooling. The system actuates on a one-time opening of valves, flow is determined by natural circulation through the system, and the system is capable of maintaining core cooling without operator action for at least [[ ]] following an event.

Therefore, subject to confirmation during licensing activities for a final design, the NRC staff finds that this requirement, based on the plain language reading of the regulation combined with the additional information in NUREG-0737, may not be technically relevant to the BWRX-300 design. The information GEH has provided in NEDC-33910 could support an exemption request when an application is submitted.

4.1.1.5 10 CFR 50.34(f)(1)(ix)

This regulation requires performance of a study to determine the need for additional space cooling to ensure reliable long-term operation of the RCIC and HPCI systems, following a complete loss of offsite power to the plant for at least 2 hours. (For plants with high-pressure core spray systems in lieu of HPCIs, substitute the words “high-pressure core spray” for “high-pressure coolant injection” and “HPCS” for “HPCI.”) (Applicable to BWRs only.)

According to GEH, the BWRX-300 does not include RCIC and HPCI systems; other systems are sufficient to mitigate the effects of a LOCA. During accident response, these systems have only a one-time action to open or close appropriate valves that are environmentally qualified to operate under post-accident conditions and do not rely on active pumps or other equipment requiring space cooling. Therefore, this study is not required.

GEH states that the BWRX-300 does not include RCIC and HPCI systems or systems that are similar in design and function. The systems that mitigate the effects of LOCA in place of RCIC and HPCI in a traditional BWR do not need space cooling to perform their functions, as a one-time action early in the accident progression is all that is needed.

The intent of this requirement, as described in NUREG-0737, is for BWR plant designs to ensure that pump-room temperatures for the RCIC and HPCI systems remain within allowable limits during long-term operation. The NUREG specifies that the systems need to be designed such that they can withstand a loss of alternating current power to their support systems for at least 2 hours.

The BWRX-300 design does not contain RCIC or HPCI systems, nor any other safety-related injection system. The ICS is responsible for cooling the core during design-basis events and will be environmentally qualified to operate in those conditions. In addition, the ICS does not include active pumps that would be subject to cooling requirements. This design negates the need for support systems to cool the equipment responsible for providing emergency core cooling.

Therefore, subject to confirmation during licensing activities for a final design, the NRC staff finds that this requirement, based on the plain language reading of the regulation combined with the additional information in NUREG-0737, may not be technically relevant to the BWRX-300 design. The information GEH has provided in NEDC-33910 could support an exemption request when an application is submitted.

4.1.1.6 10 CFR 50.34(f)(1)(x)

This regulation requires the performance of a study to ensure that the ADS, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident, taking no credit for nonsafety-related equipment or instrumentation and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWRs only.)

According to GEH, the BWRX-300 [[ ]] other systems are sufficient to mitigate the effects of a LOCA. These systems have only a one-time action to open or close appropriate valves during accident response, taking no credit for nonsafety-related equipment or instrumentation. The valves and actuators are environmentally qualified to operate under post-accident conditions. Therefore, this study is not required.

The intent of this requirement, as described in NUREG-0737, is for BWR plant designs to include ADSs with accumulators capable of cycling the valves open five times at design pressures and maintaining their ability to perform their function for 100 days following an accident, while considering normal leakage.

GEH states that the BWRX-300 design [[ ]] maintains pressure in the BWRX-300 design at an appropriate level during and following an accident applies only a one-time repositioning of valves with safety-related equipment and onsite Class 1E direct current battery power. In addition, there is no concern about the valves being functional for 100 days as the one-time operation occurs immediately following an event, and then pressure and core cooling are maintained through natural circulation.

In describing compliance with 10 CFR 50.34(f)(1)(x), NEDC-33910 indicates that the ICS and RPV isolation are one-time actuation systems. In response to the NRC staff's questions, GEH revised Sections 2.5 and 3.1.2 in NEDC-33910 to specify that the RPV isolation valves and the IC condensate return valves, respectively, will maintain their appropriate positions by positive mechanical means.

Therefore, subject to confirmation during licensing activities for a final design, the NRC staff finds that this requirement, based on the plain language reading of the regulation combined with the additional information in NUREG-0737, may not be technically relevant to the BWRX-300 design. The information GEH has provided in NEDC-33910 could support an exemption request when an application is submitted.

4.1.1.7 10 CFR 50.34(f)(2)(x)

This regulation requires a test program and associated model development and conduct of tests to qualify reactor coolant system (RCS) relief and safety valves and, for pressurized-water reactors, power-operated relief block valves for all fluid conditions expected under operating conditions, transients, and accidents. The test program shall consider anticipated transients without scram (ATWS) conditions. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed.

According to GEH, the design features of the BWRX-300 RCPB include the use of the RPS and associated safety systems for overpressure protection. Qualification tests of the system are performed to verify proper operation under all fluid conditions expected under operating conditions, transients, and accidents, and operation under ATWS conditions is evaluated. Based on the required qualification tests and evaluation, and because the [[

]], this test program and associated model development, including conduct of tests, are not required.

In response to the NRC staff's questions, GEH revised NEDC-33910 to specify that the requirement of 10 CFR 50.34(f)(2)(x) concerning an ATWS event is not technically relevant to the BWRX-300 SMR, because the RCS [[

]]. NEDC-33910 specifies that the IC condensate return valves perform a similar function during an ATWS event in providing overpressure protection and will be qualified by testing to perform their safety-related design function for fluid conditions expected under operating conditions, transients, and accidents, including ATWS events. Therefore, subject to confirmation during licensing activities for a final design, the NRC staff finds that this requirement, based on the plain language reading of the regulation combined with the additional information in NUREG-0737, may not be technically relevant to the BWRX-300 design. The information GEH has provided in NEDC-33910 could support an exemption request when an application is submitted.

4.1.1.8 10 CFR 50.34(f)(2)(xi)

This regulation requires direct indication of relief and safety valve position (open or closed) in the control room.

According to GEH, the design features of the BWRX-300 RCPB include the use of the RPS and associated safety systems for overpressure protection. The intent of this requirement is to provide indication to the operator of the inadvertent operation of relief and safety valves to minimize the potential for loss of reactor coolant, [[

]]. Therefore, direct indication of relief and safety valve position (open or closed) provided in the control room is not required. However, the BWRX-300 design does provide direct position indication of the [[

]].

The intent of this requirement, as described in NUREG-0737, is for plant designs to include direct indication in the control room of RCS relief and safety valves through the use of reliable valve-position detection devices or reliable indication of flow in the discharge pipe. This ensures that the operators are aware of any potential loss of reactor coolant inventory through the valves.

The BWRX-300 design [[ ]]. The systems that maintain pressure do so in a closed-loop fashion with the RPV following their initiation. This means that the BWRX-300 systems do not present an opportunity for an uncontrolled loss of coolant following actuation or operation [[

]].

Therefore, subject to confirmation during licensing activities for a final design, the NRC staff finds that this requirement, based on the plain language reading of the regulation combined with the additional information in NUREG-0737, may not be technically relevant to the BWRX-300 design. The information GEH has provided in NEDC-33910 could support an exemption request when an application is submitted.

4.1.2 10 CFR 50.46 Acceptance Criteria for Emergency Core Cooling System

4.1.2.1 10 CFR 50.46(a)(1)(i)

Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS). The ECCS must be designed so that its calculated cooling performance following postulated LOCAs conforms to the criteria given in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to ensure that the most severe postulated LOCAs are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. Comparisons to applicable experimental data must be made, and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, “ECCS Evaluation Models,” to 10 CFR Part 50, Part II, “Required Documentation,” sets forth the documentation requirements for each evaluation model. This section does not apply to a nuclear power reactor facility for which the certifications required under 10 CFR 50.82(a)(1) have been submitted.

According to GEH, the design features of the BWRX-300 used to comply with this requirement include [[ ]] for all postulated LOCA break sizes, in conjunction with the [[

]] for postulated LOCA break sizes [[

]]. BWRX-300 SMR specific acceptance criteria of maintaining reactor water level at or above TAF or maintaining fuel cladding temperature within normal operating temperature range bound the acceptance criteria in 10 CFR 50.46(b).

The BWRX-300 design is consistent with the applicability requirements in 10 CFR 50.46(a)(1)(i) of being a boiling light-water reactor (LWR) fueled with uranium dioxide pellets within cylindrical Zircaloy cladding; therefore, the design must include an ECCS to provide cooling to the core consistent with the requirements in 10 CFR 50.46(b). [[

]] perform this function. The NRC staff finds the systems' designs, as described in NEDC-33910, maintain reactor water level at or above TAF or maintain fuel cladding temperature within the normal operating temperature range, consistent with the requirements in 10CFR 50.46(b), and are therefore acceptable. The NRC staff will perform a detailed evaluation to confirm the requirements in 10 CFR 50.46(b) are met when an application for a BWRX-300 SMR is received.

4.1.2.2 10 CFR 50.46(b)(1)

The calculated maximum fuel element cladding temperature shall not exceed 2,200 degrees Fahrenheit (F).

According to GEH, the BWRX-300 acceptance criteria in response to a LOCA are that reactor water level is maintained at or above TAF, or fuel cladding temperature is maintained within normal operating temperature range, such that no significant fuel cladding heatup occurs. Therefore, no alternative approach, exception, or exemption from these requirements is required. Analyses to demonstrate compliance with the BWRX-300 acceptance criteria will be provided during future licensing activities.

GEH has proposed to follow the regulation as written. The NRC staff agrees that this is an appropriate approach as the reactor and fuel design proposed for the BWRX-300 aligns with the basis of the regulation to ensure the safety of LWRs using a fuel matrix of zirconium alloy cladding with uranium dioxide pellets and is therefore acceptable. Furthermore, BWRX-300 LOCA acceptance criteria specify that reactor water level be maintained at or above TAF, or fuel cladding temperature be maintained within the normal operating temperature range, and therefore, fuel temperatures would not exceed 2,200 degrees F. This will be confirmed in licensing a final design as part of the NRC staff's normal review process.

4.1.2.3 10 CFR 50.46(b)(2)

The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph, "total oxidation" means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding, the circumference does not include the rupture opening.

According to GEH, the BWRX-300 acceptance criteria in response to a LOCA are that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range such that no significant fuel cladding oxidization occurs. Therefore, no alternative approach, exception, or exemption from these requirements is required. Analyses to demonstrate compliance with the BWRX-300 acceptance criteria will be provided during future licensing activities.

GEH has proposed to follow the regulation as written. The NRC staff agrees that this is an appropriate approach as the reactor and fuel design proposed for the BWRX-300 aligns with the basis of the regulation to ensure the safety of LWRs using a fuel matrix of zirconium alloy cladding with uranium dioxide pellets and is therefore acceptable. Furthermore, BWRX-300 LOCA acceptance criteria specify that during a LOCA, the reactor water level be maintained at or above TAF, or fuel cladding temperature be maintained within normal operating temperature range, thereby preventing the calculated total oxidation of the fuel cladding to exceed 0.17 times the total cladding thickness before oxidation. This will be confirmed in licensing a final design as part of the NRC staff's normal review process.

4.1.2.4 10 CFR 50.46(b)(3)

The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

According to GEH, the BWRX-300 acceptance criteria in response to a LOCA are that the reactor water level is maintained at or above TAF, or fuel cladding temperature is maintained within normal operating temperature range, such that no significant fuel cladding hydrogen generation occurs. Therefore, no alternative approach, exception, or exemption from these requirements is required. Analyses to demonstrate compliance with the BWRX-300 acceptance criteria will be provided during future licensing activities.

GEH has proposed to follow the regulation as written. The NRC staff agrees that this is an appropriate approach as the reactor and fuel design proposed for the BWRX-300 aligns with the basis of the regulation to ensure the safety of LWRs using a fuel matrix of zirconium alloy cladding with uranium dioxide pellets and is therefore acceptable. Furthermore, BWRX-300 LOCA acceptance criteria specify that during a LOCA, the reactor water level be maintained at or above TAF, or fuel cladding temperature be maintained within the normal operating temperature range, such that the calculated total amount of hydrogen generation could not exceed the regulatory requirement. This will be confirmed in licensing a final design as part of the NRC staff's normal review process.

4.1.2.5 10 CFR 50.46(b)(4)

Calculated changes in core geometry shall be such that the core remains amenable to cooling.

According to GEH, the BWRX-300 acceptance criteria in response to a LOCA are that the reactor water level is maintained at or above TAF, or fuel cladding temperature is maintained within normal operating temperature range, so that no significant changes in core geometry occur. Therefore, no alternative approach, exception, or exemption from these requirements is required. Analyses to demonstrate compliance with the BWRX-300 acceptance criteria will be provided during future licensing activities.

GEH has proposed to follow the regulation as written. The NRC staff agrees that this is an appropriate approach as the reactor and fuel design proposed for the BWRX-300 aligns with the basis of the regulation to ensure the safety of LWRs using a fuel matrix of zirconium alloy cladding with uranium dioxide pellets and is therefore acceptable. Furthermore, BWRX-300 LOCA acceptance criteria specify that during a LOCA, the reactor water level be maintained at or above TAF, or fuel cladding temperature be maintained within normal operating temperature range, so that a coolable geometry is maintained during the event mitigation and for long-term cooling of the core post-accident. This will be confirmed in licensing a final design as part of the NRC staff's normal review process, including the impacts of dynamic LOCA loads on the fuel assemblies.

4.1.2.6 10 CFR 50.46(b)(5)

After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

According to GEH, the BWRX-300 acceptance criteria in response to a LOCA are that the reactor water level is maintained at or above TAF, or fuel cladding temperature is maintained within the normal operating temperature range, such that no significant fuel cladding heatup occurs for at least [[ ]] while relying on only automatic actuation of passive safety-related systems. Therefore, no alternative approach, exception, or exemption from these requirements is required. Analyses to demonstrate compliance with the BWRX-300 acceptance criteria will be provided during future licensing activities.

GEH has proposed to follow the regulation as written. The NRC staff agrees that this is an appropriate approach as the reactor and fuel design proposed for the BWRX-300 aligns with the basis of the regulation to ensure the safety of LWRs using a fuel matrix of zirconium alloy cladding with uranium dioxide pellets and is therefore acceptable. Furthermore, GEH indicates that no significant fuel cladding heatup occurs during the postulated LOCA for at least [[ ]] without operator action and using only safety-related equipment. This approach, as described in NEDC-33910, is consistent with Commission policy related to passive plant design described in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994 (ADAMS Accession No. ML003708068), and specifications in the Electric Power Research Institute's "Advanced Light Water Reactor Utility Requirements Document," Revision 8, issued March 1999. Therefore, the approach is acceptable. GEH will provide additional analysis, which may include the use of nonsafety-related equipment and operator actions, to show that long-term cooling is provided beyond [[ ]]. The NRC staff will perform a detailed evaluation of the additional analysis to confirm that the design satisfies applicable regulatory requirements when an application for a BWRX-300 SMR is received.

4.1.3 10 CFR 50.55a

The regulation in 10 CFR 50.55a(a) incorporates by reference specific editions and addenda of consensus codes and standards with conditions to establish requirements for the design, fabrication, erection, construction, testing, and inspection of certain components of nuclear power plants, except where the NRC grants relief from or authorizes alternatives to those requirements.

In Section 4.1.3, "10 CFR 50.55a," NEDC-33910 specifies that the BWRX-300 RPV isolation valves and overpressure protection design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within 6 months of any license application, including any application for a construction permit under 10 CFR Part 50, or design certification application under 10 CFR Part 52. In response to the NRC staff's questions, GEH revised NEDC-33910 to specify that these requirements are to be implemented during the detailed design of the safety-related components of the ICS and RPV isolation valves. NEDC-33910 specifies that the BWRX-300 design will meet the requirements of 10 CFR 50.55a.

Section 4.1.3 in NEDC-33910 specifies that the requirements of 10 CFR 50.55a will be satisfied through use of the standards approved in that section. Therefore, the NRC staff finds this approach, as described in NEDC-33910, acceptable. The NRC staff will conduct a detailed evaluation to confirm 10 CFR 50.55a is met when an application for a BWRX-300 SMR is received.

4.1.4 10 CFR Part 50, Appendix A, General Design Criterion 1

In 10 CFR Part 50, Appendix A, GDC 1 requires that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

In Section 4.1.4, "10 CFR 50 Appendix A, GDC 1," NEDC-33910 states that the BWRX-300 RPV isolation and overpressure protection design features, [[ ]], are to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed in accordance with generally recognized codes and standards and under an approved quality assurance program with approved control of records. In response to the NRC staff's questions, GEH revised NEDC-33910 to specify that the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 1.

Section 4.1.4 in NEDC-33910 specifies that the requirements of 10 CFR Part 50, Appendix A, GDC 1, will be satisfied. The NRC staff finds this approach, as described in NEDC-33910, consistent with 10 CFR Part 50, Appendix A, GDC 1, and therefore acceptable. The NRC staff will conduct a detailed evaluation to confirm 10 CFR Part 50, Appendix A, GDC 1 is satisfied when an application for a BWRX-300 SMR is received.

4.1.5 10 CFR Part 50, Appendix A, General Design Criterion 2

In 10 CFR Part 50, Appendix A, GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

In Section 4.1.5, “10 CFR 50 Appendix A, GDC 4,” NEDC-33910 states that the BWRX-300 RPV isolation and overpressure protection design features, [[ ]], are to be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. Specific design requirements for the [[ ]] used to verify the capability to perform their safety functions, and the natural phenomena and effects evaluated, will be provided during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 2.

In response to the NRC staff’s question, GEH clarified in its letters dated April 20, 2020, and May 4, 2020, that NEDC-33911 provides details on the piping connected to the RPV isolation valve assemblies. In particular, GEH indicated that a terminal end break would be postulated [[

]] as specified in BTP 3-4, Part B, Item 1(iii)(2). GEH stated that qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as accepted in RG 1.100, will be addressed in the detailed design of the valves and will be specified during future licensing activities. Also in its April 20 and May 4, 2020, letters, GEH noted that the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 4, by designing the valves to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs, and the valves will be appropriately protected against dynamic effects that may result from equipment failures and from events and conditions outside the nuclear power unit.

Section 4.1.5 in NEDC-33910 specifies that the requirements of 10 CFR Part 50, Appendix A, GDC 2, will be satisfied. Therefore, the NRC staff finds this approach, as described in NEDC-33910, acceptable. The NRC staff will conduct a detailed evaluation to confirm 10 CFR Part 50, Appendix A, GDC 2 is satisfied when an application for a BWRX-300 SMR is received.

4.1.6 10 CFR Part 50, Appendix A, General Design Criterion 4

In 10 CFR Part 50, Appendix A, GDC 4 requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

In Section 4.1.6, “10 CFR Appendix A, GDC 4,” NEDC-33910 specifies that a design requirement of the BWRX-300 SMR is that the SSCs required to mitigate a LOCA shall be operable in the environmental conditions (primary containment vessel pressure, temperature, radiation, etc.) following a LOCA. In addition, the dynamic effects of postulated pipe breaks are to be considered in the BWRX-300 design. As described in this LTR, [[

]] the BWRX-300 design requirements consider the acceptable criteria to identify postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, as discussed in Section 2.4.1 of this LTR. Therefore, the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 4.

In response to the NRC staff's questions, GEH clarified in its letters dated April 20, 2020, and May 4, 2020, that as described in detail in NEDC-33911, [[

]], extending to the containment wall, the BWRX-300 design requirements include identifying postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, Part B, Item 1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2). In particular, GEH stated that a terminal end break would be postulated [[ ]] as specified in BTP 3-4, Part B, Item 1(iii)(2). GEH indicated that the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 4, by designing the valves to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs, and the valves will be appropriately protected against dynamic effects that may result from equipment failures and from events and conditions outside the nuclear power unit. GEH indicated that qualification, such as compliance with ASME Standard QME-1-2007 (or later edition), as accepted in RG 1.100, will be addressed in the detailed design of the valves and will be specified during future licensing activities.

The NRC staff finds this approach, as described in NEDC-33910, consistent with 10 CFR Part 50, Appendix A, GDC 4, and, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm 10 CFR Part 50, Appendix A, GDC 4 is satisfied when an application for a BWRX-300 SMR is received.

4.1.7 10 CFR Part 50, Appendix A, General Design Criterion 14

In 10 CFR Part 50, Appendix A, GDC 14, "Reactor coolant pressure boundary," requires that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

In Section 4.1.7, "10 CFR 50 Appendix A, GDC 14," NEDC-33910 states that the RPV nozzles, [[
]] are designed, fabricated, erected, and tested as ASME BPV Code, Section III, Class 1 components. [[

]]. This results in an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture, in compliance with this criterion. For piping that has [[
]]. In response to the NRC staff's questions, GEH revised NEDC-33910 to indicate that further design details are to be described during future licensing activities. The revised NEDC-33910 specifies that the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 14.

Section 4.1.7 in NEDC-33910 describes compliance with 10 CFR Part 50, Appendix A, GDC 14, concerning the RCPB. The NRC staff finds this approach, as described in NEDC-33910, consistent with 10 CFR Part 50, Appendix A, GDC 14, and therefore acceptable. The NRC staff will conduct a detailed evaluation to confirm 10 CFR Part 50, Appendix A, GDC 14 is satisfied when an application for a BWRX-300 SMR is received.

4.1.8 10 CFR Part 50, Appendix A, GDC 15

In 10 CFR Part 50, Appendix A, GDC 15 requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.

In Section 4.1.8, "10 CFR 50 Appendix A, GDC 15," NEDC-33910 states that overpressure protection for the BWRX-300 SMR is provided in accordance with ASME BPV Code, Section III, paragraph NB-7120, subparagraphs [[ ]]. The combination of RPS and [[ ]] design features ensures that the acceptance criteria for each component in the protected system are met, which includes ensuring that (1) the overpressure does not exceed 1.1 times design pressure for the expected system pressure transient condition, and (2) the calculated stress intensity and other design limitations for Service Level C are not exceeded for the unexpected system excess pressure transient condition. In response to the NRC staff's questions, GEH revised NEDC-33910 to indicate that further design details will be described during future licensing activities. The revised NEDC-33910 specifies that the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 15. Section 4.1.8 in NEDC-33910 describes compliance with 10 CFR Part 50, Appendix A, GDC 15, with respect to overpressure protection for the BWRX-300 SMR. The NRC staff finds this approach, as described in NEDC-33910, consistent with 10 CFR Part 50, Appendix A, GDC 15, and therefore acceptable. The NRC staff will conduct a detailed evaluation to confirm 10 CFR Part 50, Appendix A, GDC 15 is satisfied when an application for a BWRX-300 SMR is received.

4.1.9 10 CFR Part 50, Appendix A, General Design Criterion 30

In 10 CFR Part 50, Appendix A, GDC 30, "Quality of reactor coolant pressure boundary," requires that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

In Section 4.1.9, "10 CFR 50 Appendix A, GDC 30," NEDC-33910 states that the components of the RCPB, including the ICS and RPV isolation valves, and the overpressure protection equipment [[ ]] are to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed in accordance with generally recognized codes and standards, and under an approved quality assurance program with approved control of records, as required by 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1. In addition, means are to be provided to detect and identify the location of the source of reactor coolant leakage, including the components of the ICS and RPV isolation valves, for components of the RCPB. In response to the NRC staff questions, GEH revised NEDC-33910 to indicate that further design details will be described during future licensing activities. The revised NEDC-33910 specifies that the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 30.

Section 4.1.9 in NEDC-33910 describes compliance with 10 CFR Part 50, Appendix A, GDC 30, with respect to quality of the RCPB for the BWRX-300 SMR. The NRC staff finds this approach, as described in NEDC-33910, consistent with 10 CFR Part 50, Appendix A, GDC 30, and therefore acceptable. The NRC staff will conduct a detailed evaluation to confirm 10 CFR Part 50, Appendix A, GDC 30 is satisfied when an application for a BWRX-300 SMR is received.

4.1.10 10 CFR Part 50, Appendix A, General Design Criterion 31

In 10 CFR Part 50, Appendix A, GDC 31, "Fracture prevention of reactor coolant pressure boundary," requires that the RCPB be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.

According to GEH, In Section 4.1.10, "10 CFR 50 Appendix A, GDC 31," NEDC-33910 states that the components of the RCPB, including the ICS and RPV isolation valves, are to be designed with sufficient margin to ensure that these requirements are met. In response to the NRC staff's questions, GEH revised NEDC-33910 to indicate that further design details will be described during future licensing activities. The revised NEDC-33910 specifies that the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 31.

Section 4.1.10 in NEDC-33910 describes compliance with 10 CFR Part 50, Appendix A, GDC 31, with respect to fracture prevention of the RCPB for the BWRX-300 SMR. The NRC staff finds this approach, as described in NEDC-33910, consistent with 10 CFR Part 50, Appendix A, GDC 31, and therefore acceptable. The NRC staff will conduct a detailed evaluation to confirm 10 CFR Part 50, Appendix A, GDC 31 is satisfied when an application for a BWRX-300 SMR is received.

4.1.11 10 CFR Part 50, Appendix A, General Design Criterion 33

In 10 CFR Part 50, Appendix A, GDC 33 requires that a system to supply reactor coolant makeup for protection against small breaks in the RCPB be provided. The system safety function shall be to ensure that SAFDLs are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components that are part of the boundary. The system shall be designed to ensure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

According to GEH, GDC 33 applies to small leaks in the RCPB that are [[

]] (e.g., leakage from flanges or cracks in piping or other components) and do not exceed the capability of the nonsafety-related high-pressure CRD system used as normal reactor coolant makeup during power operations. The plant technical specifications stipulate the maximum allowed leakage rate for continuing power operation. For leakage greater than the maximum allowed leakage rate, automatic reactor scram and automatic actuation of the [[ ]] is not anticipated for most of these small leaks because the normal means of makeup from the high-pressure CRD system and feedwater maintain the level in the normal operating range. However, these small leaks that do not exceed the capability of the nonsafety-related high-pressure CRD system are evaluated using SAFDLs rather than the BWRX-300 acceptance criteria in response to a LOCA. The high-pressure CRD system pumps can operate using either onsite electric power system operation (assuming offsite power is not available) backed up by nonsafety-related standby diesel generators and offsite electric power system operation (assuming onsite power is not available). GEH states that the BWRX-300 design will therefore meet the requirements of 10 CFR Part 50, Appendix A, GDC 33.

The NRC staff considers the BWRX-300 high-pressure CRD system, as described in the LTR, to be important to safety because it will be relied on to satisfy the reactor coolant makeup function required by GDC 33 in order to protect against small breaks in the RCPB. GEH indicates that the system, including the pumps, will be designed to ensure that its safety function of maintaining SAFDLs can be performed for onsite electric power operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available).

The NRC staff finds this approach, as described in NEDC-33910, consistent with 10 CFR Part 50, Appendix A, GDC 33, and therefore acceptable. The NRC staff will conduct a detailed evaluation to confirm 10 CFR Part 50, Appendix A, GDC 33 is satisfied when an application for a BWRX-300 SMR is received.

4.1.12 10 CFR Part 50, Appendix A, GDC 35

In 10 CFR Part 50, Appendix A, GDC 35 requires that a system to provide abundant emergency core cooling be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system can accomplish its safety function, assuming a single failure.

According to GEH, the combined design features of the [[ ]] meet the definition of an ECCS that has a calculated cooling performance following postulated LOCAs in full compliance with the criteria in 10 CFR 50.46(b). The [[ ] has the capability to provide sufficient emergency core cooling, which is assured for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS through the use of [[

]].

The BWRX-300 acceptance criteria in response to a LOCA are that the performance of the [[

]] is sufficient to ensure that reactor water level is maintained above TAF, or fuel cladding temperature is maintained within normal operating temperature range, such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

[[

]]

Analyses to demonstrate compliance with the BWRX-300 acceptance criteria will be provided during future licensing activities.

The combination of the [[ ]] as described in NEDC-33910, along with the stated design requirements in the LTR, is consistent with 10 CFR Part 50, Appendix A, GDC 35, and is therefore acceptable. This is accomplished [[

]] to ensure that the water level in the RPV is maintained while cooling is provided [[ ]], to ensure that the BWRX-300 LOCA acceptance criteria are met. [[

]].

For small breaks [[ ]], GEH states that the depressurization of the RPV [[

,]] mitigates the loss of inventory, such that reactor water level is maintained at or above TAF, or fuel cladding temperature is maintained within normal operating temperature range. This will be confirmed in a future licensing review of a final design as part of the NRC staff's normal review process.

In addition, the initial actions to actuate the systems that are part of the ECCS are powered by Class 1E batteries such that the systems can accomplish their safety function during a loss of onsite or offsite electric power. The systems as described in NEDC-33910 are also consistent with 10 CFR Part 50, Appendix A, GDC 35, because they are designed to ensure that a single failure will not prevent them from meeting their safety functions of maintaining core inventory and providing adequate core cooling.

The BWRX-300 design, as described in NEDC-33910 is consistent with 10 CFR Part 50, Appendix A, GDC 35, because it provides sufficient cooling of the core [[ ]] and the reactor water level is maintained at or above TAF, or fuel cladding temperature is maintained within normal operating temperature range. The NRC staff will conduct a detailed evaluation to confirm 10 CFR Part 50, Appendix A, GDC 35 is satisfied when an application for a BWRX-300 SMR is received.

4.1.13 10 CFR Part 50, Appendix A, General Design Criterion 37

In 10 CFR Part 50, Appendix A, GDC 37 requires that an ECCS be designed to permit appropriate periodic pressure and functional testing to ensure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

In Section 4.1.13, “10 CFR 50 Appendix A, GDC 37,” NEDC-33910 states that the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i), which has a calculated cooling performance following postulated LOCAs in compliance with the BWRX-300 acceptance criteria in response to a LOCA that bounds the acceptance criteria in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS in compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling, which is assured for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS [[ ]]. Specific requirements for periodic pressure and functional testing of the [[ ]] to ensure (1) the structural and leaktight integrity of these components, (2) the operability and performance of these active components, and (3) the operability of the systems as a whole containing the [[ ]] will be provided during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 37.

Section 4.1.13 in NEDC-33910 describes compliance with 10 CFR Part 50, Appendix A, GDC 37, with respect to testing of the ECCS for the BWRX-300 SMR. The NRC staff finds this approach acceptable to ensure GEH can demonstrate that that GDC 37 will be met. The NRC staff will conduct a detailed evaluation to confirm 10 CFR Part 50, Appendix A, GDC 37 is satisfied when an application for a BWRX-300 SMR is received.

4.2 Regulatory Guides

4.2.1 Regulatory Guide 1.26

RG 1.26, Revision 5, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” issued February 2017, describes methods for use in implementing the regulatory requirements of 10 CFR Part 50, Appendix A, GDC 1, with regard to a quality classification system related to specified national standards that may be used to determine quality standards acceptable to the NRC staff for components containing water, steam, or radioactive material in light-water-cooled nuclear power plants.

In Section 4.2.1, “Regulatory Guide 1.26,” NEDC-33910 states that the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR Part 50, Appendix A, GDC 1. The RPV isolation valves and the components of the ICS are classified in conformance with the guidance in RG 1.26. Therefore, GEH states the BWRX-300 design conforms to the guidance for the RPV isolation valves and the ICS, including the regulatory positions in RG 1.26.

In response to the NRC staff’s questions, GEH revised NEDC-33910 to clarify that no alternative approaches or exceptions were planned with respect to RG 1.26. The NRC staff finds that the BWRX-300 design is consistent with the guidance for the RPV isolation valves and

ICS and is, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm that the BWRX-300 design conforms to the guidance in RG 1.26 when an application for a BWRX-300 SMR is received.

4.2.2 Regulatory Guide 1.29

RG 1.29, Revision 5, "Seismic Design Classification," issued July 2016, describes methods acceptable to the NRC staff for use in implementing the regulatory requirements of 10 CFR 50.55a(h); 10 CFR Part 50, Appendix A, GDC 2; and 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants." These methods are useful for identifying and classifying those features of LWR nuclear power plants that must be designed to withstand the effects of the safe-shutdown earthquake.

In Section 4.2.2, "Regulatory Guide 1.29," NEDC-33910 states that the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a(h); 10 CFR Part 50, Appendix A, GDC 2; and 10 CFR Part 50, Appendix S. The components of the ICS and RPV isolation valves are classified as seismic Class I in conformance with the guidance in RG 1.29. GEH states that the BWRX-300 design therefore conforms to the guidance, including regulatory positions in RG 1.29.

In response to the NRC staff's questions, GEH revised NEDC-33910 to clarify that no alternative approaches or exceptions were planned with respect to RG 1.29. The NRC staff finds that the BWRX-300 design is consistent with the guidance in RG 1.29 for the seismic design classification and is, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm whether the BWRX-300 design conforms to the guidance in RG 1.29 when an application for a BWRX-300 SMR is received.

4.2.3 Regulatory Guide 1.45

RG 1.45, Revision 1, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," issued May 2008, describes methods acceptable to the NRC staff for use in implementing the regulatory requirements of 10 CFR Part 50, Appendix A, GDC 14 and 30, with regard to selecting reactor coolant leakage detection systems, monitoring for leakage, and responding to leakage for LWR nuclear power plants. This guidance also cites 10 CFR 50.55a, which requires the performance of ISI and inservice testing (IST) activities for nuclear power plant components so that the concept of defense in depth is applied to provide assurance that the structural integrity of the RCPB is maintained.

In Section 4.2.3, "Regulatory Guide 1.45," NEDC-33910 specifies that the LTR describes how the design of the [[ ]] complies with the requirements of 10 CFR Part 50, Appendix A, GDC 1 and GDC 14. Additionally, the means for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage in compliance with the requirements of 10 CFR Part 50, Appendix A, GDC 30, and the requirements for ISI and testing of the [[ ]] in compliance with the requirements of 10 CFR 50.55a, are to be demonstrated during future licensing activities. GEH states that the BWRX-300 design therefore conforms to the guidance, including regulatory positions of RG 1.45.

In response to the NRC staff's questions, GEH revised NEDC-33910 to clarify that it planned no alternative approaches or exceptions with respect to RG 1.45. The NRC staff finds that the BWRX-300 design is consistent with the guidance in RG 1.45 with respect to RCS leakage and is, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm whether the BWRX-300 design conforms to the guidance in RG 1.45 when an application for a BWRX-300 SMR is received.

4.2.4 Regulatory Guide 1.84

RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," lists the ASME BPV Code, Section III, Code Cases that the NRC has approved for use as voluntary alternatives to the mandatory ASME BPV Code provisions that are incorporated by reference in 10 CFR Part 50. This applies to reactor licensees subject to 10 CFR 50.55a. These ASME BPV Code, Section III, Code Cases are acceptable to the NRC staff for use in implementing the regulatory requirements of 10 CFR Part 50, Appendix A, GDC 1 and GDC 30, and 10 CFR 50.55a(c), which requires, in part, that components of the RCPB be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of the ASME BPV Code, Section III, or equivalent quality standards.

In Section 4.2.4, "Regulatory Guide 1.84," NEDC-33910 specifies that the LTR describes how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR Part 50, Appendix A, GDC 1 and GDC 30, respectively. Section 4.1.3 describes how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a. Compliance with 10 CFR 50.55a, including the use of ASME BPV Code, Section III, Code Cases endorsed in RG 1.84 where necessary, is to be demonstrated during future licensing activities. GEH states that the BWRX-300 design therefore conforms to the guidance, including the regulatory positions of RG 1.84.

In response to the NRC staff's questions, GEH revised NEDC-33910 to clarify that it planned no alternative approaches or exceptions with respect to RG 1.84. The NRC staff finds that the BWRX-300 design is consistent with the guidance in RG 1.84 with respect to the ASME BPV Code, Section III, Code Cases and is, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm whether the BWRX-300 design conforms to the guidance in RG 1.84 when an application for a BWRX-300 SMR is received.

4.2.5 Regulatory Guide 1.147

RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," lists the ASME BPV Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Code Cases that the NRC has approved for use as voluntary alternatives to the mandatory ASME BPV Code provisions that are incorporated by reference in 10 CFR Part 50.

In Section 4.2.5, "Regulatory Guide 1.147," NEDC-33910 discusses the approval of the ASME BPV Code, Section XI, Code Cases for voluntary alternatives to the mandatory ASME BPV Code provisions, as incorporated by reference in 10 CFR 50.55a, for ISI in RG 1.147. Section 4.2.5 indicates that the performance of ISI activities does not apply during the design phase of the BWRX-300 SMR. GEH states that Section 4.2.5 therefore specifies that the guidance in RG 1.147 does not apply to the BWRX-300 design phase in meeting 10 CFR 50.55a.

The NRC staff finds the plans to not apply the Code Cases identified in RG 1.147 to be acceptable for the design phase of the BWRX-300 SMR because the Code Cases are voluntary alternatives to the ASME BPV Code, Section XI. The NRC staff will conduct a detailed evaluation to determine if any of the Code Cases identified in RG 1.147 are applied during the design phase of the BWRX-300 SMR when an application for a BWRX-300 SMR is received.

4.2.6 Regulatory Guide 1.192

RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," lists Code Cases associated with the ASME *Operation and Maintenance of Nuclear Power Plants* (OM Code), Division 1, Section IST, that the NRC has approved for use as voluntary alternatives to the mandatory ASME OM Code provisions that are incorporated by reference in 10 CFR Part 50.

In Section 4.2.6, "Regulatory Guide 1.1.92," NEDC-33910 discusses the approval of the ASME OM Code Cases as voluntary alternatives to the mandatory ASME OM Code provisions, as incorporated by reference in 10 CFR 50.55a, for IST in RG 1.192. Section 4.2.5 indicates that the performance of IST activities does not apply during the design phase of the BWRX-300 SMR. Therefore, Section 4.2.6 specifies that the guidance in RG 1.192 does not apply to the BWRX-300 design phase in meeting 10 CFR 50.55a.

The NRC staff finds the plans to not apply the Code Cases identified in RG 1.192 to be acceptable for the design phase of the BWRX-300 SMR because the Code Cases are voluntary alternatives to the ASME OM Code. The NRC staff will conduct a detailed evaluation to determine if any of the Code Cases identified in RG 1.192 are applied during the design phase of the BWRX-300 SMR when an application for a BWRX-300 SMR is received.

4.3 NUREG-0800 Standard Review Plan Guidance

Based on GEH's proposed timeline for submission of the BWRX 300 design for review and requested review schedule, the NRC does not, at this time, plan to develop design specific review standards for the BWRX-300. However, GEH employs novel design features and strategies to ensure safety at the facility and identified departures from staff review guidance in this LTR. The NRC staff appreciates this effort to inform a future safety review of the BWRX-300 design upon receipt of the application.

4.3.1 Standard Review Plan 3.9.6

Section 4.3.1, "Standard Review Plan 3.9.6," in NEDC-33910, discusses SRP Section 3.9.6, Revision 4, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," issued March 2017 (ADAMS Accession No. ML16134A116). Section 4.3.1 states that the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 14, 15, and 37, with specific requirements for the [[

]] to be provided during future licensing activities. In addition, the BWRX-300 RPV isolation valves and overpressure protection design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within 6 months of any license application under 10 CFR Part 52. The requirements are to be implemented during detailed design of the safety-related activities and safety-related components of the [[ ]]. Therefore, GEH considers the existing SRP to provide adequate guidance for use during future NRC review of the BWRX-300 design for a 10 CFR Part 52 design certification, if pursued, or for future 10 CFR Part 50 license applications.

Based on its review, the NRC staff finds that the BWRX-300 design is consistent with the guidance in SRP Section 3.9.6 as it relates to the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints during future licensing activities and is therefore acceptable The NRC staff will conduct a detailed evaluation to confirm the BWRX-300 design satisfies the guidance in SRP Section 3.9.6 as it relates to the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints when an application for a BWRX-300 SMR is received.

4.4 Generic Issues

Section 4.4, “Generic Issues,” in NEDC-33910 provides a sample of generic issues based on their relevance to the scope of the topical report. In response to the NRC staff’s questions, GEH revised NEDC-33910 to specify that an up-to-date evaluation of generic issues will be provided during future licensing activities either by GEH in support of a 10 CFR Part 52 design certification application, or by a license applicant requesting a construction permit or operating license under 10 CFR Part 50 or a combined license (COL) under 10 CFR Part 52.

The NRC staff finds the plan to provide an up-to-date evaluation of generic issues applicable to the BWRX-300 design during future licensing activities to be acceptable.

4.5 Operational Experience and Generic Communications

Section 4.5, “Operational Experience and Generic Communications,” in NEDC-33910 provides a sample of operational experience and generic communications based on their relevance to the scope of the topical report. In response to the NRC staff questions, GEH revised NEDC-33910 to specify that an up-to-date evaluation of operating experience and generic communications will be provided during future licensing activities either by GEH in support of a 10 CFR Part 52 design certification application, or by a license applicant requesting a construction permit or operating license under 10 CFR Part 50 or a COL under 10 CFR Part 52. For example, GEH revised NEDC-33910 to include a discussion of NRC Generic Letter 95-07, “Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves,” dated August 17, 1995, which will be evaluated for applicability during future licensing activities.

The NRC staff finds the plan to provide an up-to-date evaluation of operating experience and generic communications applicable to the BWRX-300 design during future licensing activities to be acceptable.

5.0 Conclusion

Based on the above discussion, the NRC staff concludes that the design requirements, acceptance criteria, and regulatory bases for the design functions of the RPV isolation and overpressure protection for the BWRX-300 SMR, as des cribbed in GEH NEDC-33910, is acceptable. In particular, NEDC-33910 describes (1) design requirements for the RPV isolation valves and automatic actuation of the ICS to remove decay heat from large, medium, and small pipe breaks to meet the acceptance criteria in 10 CFR 50.46(b) and (2) design requirements for the RPS and ICS for overpressure protection of the BWRX-300 design. If an applicant for a construction permit under 10 CFR Part 50, or a design certification or COL 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 RPV isolation and overpressure protection 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 safety evaluation, 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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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 9730, Question 03.09.06-1, revised Section 1.1 and added new Sections 4.1.5 and 4.1.13 to address compliance with 10 CFR 50, Appendix A, GDC 2 and GDC 37, and to address addition of Generic Letter 95-07 as new Section 4.5.2 • NRC eRAI 9730, Question 03.09.06-1, supplemental response revised new Section 4.1.5 to address [[ ]]. • NRC eRAI 9730, Question 03.09.06-3, replaced the use of terms such as "consideration" and "considered" with appropriate terms in Sections 2.4, 2.4.1, 2.5, 2.6, 2.7, renumbered 4.1.6, and renumbered 4.3.2. • NRC eRAI 9730, Question 03.09.06-4, added the design requirements for the use of positive mechanical means in the design of the valve actuators to maintain these valves in their required post-accident valve positions in Sections 2.5 and 3.1.2. • NRC eRAI 9730, Question 03.09.06-5, revised Section 4.1.1 to describe requirements for 10 CFR 50.34(f)(2)(x) as not technically relevant rather than not required, but to also indicate that the [[ ]] will be qualified by testing to perform their safety-related design function for fluid conditions expected under operating conditions, transients and accidents including ATWS events. • NRC eRAI 9730, Questions 03.09.06-6, 03.09.06-7, and 03.09.06-14, clarified statements that no alternative approach, exception, or exemption from certain regulatory requirements is required providing a commitment to meeting the applicable regulatory requirements during detailed design in Sections 2.7, 4.1.2, 4.1.3, 4.1.4, renumbered 4.1.6, renumbered 4.1.7, renumbered 4.1.8, renumbered 4.1.9, renumbered 4.1.10, renumbered 4.1.12, 4.2.1, 4.2.2, 4.2.3, and 4.2.4.

Revision Number	Description of Change
Supplement 1 (continued)	Revised to incorporate the following responses to NRC Requests for Additional Information (eRAIs): • NRC eRAI 9730, Question 03.09.06-6 supplemental response, revised Section 4.1.2 to reflect [[ ]] and to include compliance with the requirements of the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(5) using a long-term cooling timeframe of [[ ]]. • NRC eRAI 9730, Question 03.09.06-8, revised Section renumbered 4.1.11 to address compliance with 10 CFR 50, Appendix A, GDC 33. • NRC eRAI 9730, Question 03.09.06-9, added new Section 4.1.5 to address compliance with 10 CFR 50, Appendix A, GDC 2. • NRC eRAI 9730, Question 03.09.06-11, added new Sections 4.2.5 and 4.2.6 to address conformance with the regulatory guidance of RG 1.147 and RG 1.192. • NRC eRAI 9730, Question 03.09.06-12, added new Section 4.3.1 to address conformance with the regulatory guidance of SRP 3.9.6. • NRC eRAI 9730, Question 03.09.06-13, revised Sections 4.4 and 4.5 identifying the limited scope of the evaluation of generic issues, and operational experience and generic communications, respectively, and committing to the up-to-date evaluation of these issues to be provided during future licensing activities. • NRC eRAI 9731, Question 03.06.02-3, and supplemental response, revised Section 2.4.1 to address material of construction for the [[ ]] which will be determined during detailed design and addressed in future licensing activities.

Revision Number	Description of Change
Supplement 1 (continued)	Revised to incorporate the following responses to NRC Requests for Additional Information (eRAIs): • NRC eRAI 9731, Question 03.06.02-1 supplemental response, revised Section 2.4.1 to describe the [[ ]] consistent with the discussions in NEDC-33911P, BWRX-300 Containment Performance. • NRC eRAI 9732, Question NONE-1, revised Section renumbered 4.1.11 to address compliance with 10 CFR 50, Appendix A, GDC 33. • NRC eRAI 9732, Question NONE-2, and supplemental response, revised Sections 1.1., 2.1, 2.7, 2.8, 4.1.1, 4.1.2, renumbered 4.1.11, renumbered 4.1.12, new 4.1.13, renumbered 4.3.4, and renumbered 4.3.5 to include compliance with the requirements of 10 CFR 50.46(b) by use of BWRX-300 acceptance criteria in response to a LOCA including maintaining reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range, which bound the acceptance criteria in 10 CFR 50.46(b). • NRC eRAI 9732, Question NONE-3, and supplemental response, revised Sections 2.1, 2.2, 2.4, 2.7, 4.1.2, and renumbered 4.1.12, and Table 2-1, to reflect [[ ]] and to include compliance with the requirements of 10 CFR 50.46(b)(5) using a long-term cooling timeframe of [[ ]].
Supplement 2	Section 2.5 is revised to clarify that the ICS RPV isolation valves automatic isolation function uses logic and functionality similar to, rather than the same as, the ESBWR ICS containment isolation valves, the [[ ]] is removed from the design, and the [[ ]] is clarified as an example of the ICS RPV isolation valves automatic isolation function logic and functionality.
1	Created "-A" version by adding GEH's responses to the NRC's Requests for Additional Information (RAIs) (References 5.8 and 5.9) and the NRC's Final Safety Evaluation (Reference 5.10). Added References 5.8 through 5.10.

2	Corrected SER Section 5.0 Conclusion to match SER transmitted to ADAMS under accession number ML20176A449 and corrected in Reference 5.10. A different version had been transmitted to GEH and was inserted into NEDC-33901P-A, Rev. 1.

Acronyms and Abbreviations

Term	Definition
ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ADS	Automatic Depressurization System
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
B&PV	Boiler & Pressure Vessel
BTP	Branch Technical Position
BWR	Boiling Water Reactor
COL	Combined Operating License
CP	Construction Permit
CRD	Control Rod Drive
DCA	Design Certification Application
DCD	Design Control Document
DPV	Depressurization Valve
ECCS	Emergency Core Cooling System
ESBWR	Economically Simplified Boiling Water Reactor
EQ	Environmental Qualification
FMCRD	Fine Motion Control Rod Drive
GDC	General Design Criteria
GEH	GE Hitachi Nuclear Energy
HGNE	Hitachi-GE Nuclear Energy Ltd.
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
I&C	Instrumentation and Control
IC	Isolation Condenser
ICS	Isolation Condenser System
ISI	Inservice Inspection
IST	Inservice Testing

Term	Definition
LOCA	Loss-of-Coolant Accident
LTR	Licensing Topical Report
LWR	Light-Water-Reactor
NRC	Nuclear Regulatory Commission
OL	Operating License
PCCS	Passive Containment Cooling System
PCV	Primary Containment Vessel
PORV	Power Operated Relief Valve
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SMR	Small Modular Reactor
SRP	Standard Review Plan
SSC	Structure, System, and Component
SSE	Safe-Shutdown Earthquake
TAF	Top of Active Fuel
TMI	Three Mile Island

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 Reactor Pressure Vessel (RPV) isolation and overpressure protection design functions, specifically for the following areas:

- Design requirements are specified for the RPV isolation valves and configuration with the function to close to limit the loss of coolant from large and medium pipe breaks, and design requirements are specified for automatic actuation of the Isolation Condenser System (ICS) to remove decay heat from large, medium, and small pipe breaks, to meet the BWRX-300 acceptance criteria in response to a Loss-Of-Coolant Accident (LOCA) which bound the acceptance criteria in 10 CFR 50.46(b). [[ ]] The design of the RPV isolation valves and ICS meet the requirements of 10 CFR 50.46(b) and 10 CFR 50 Appendix A, General Design Criteria, GDC 1, GDC 2, GDC 4, GDC 14, GDC 30, GDC 31, GDC 33, GDC 35, and GDC 37.
- Design requirements are specified for the Reactor Protection System (RPS) and ICS for overpressure protection. [[ ]] The design of the RPS and ICS meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 15, GDC 30, and GDC 31.

1.2 Scope

The scope of this report includes the following:

- A technical evaluation of the BWRX-300 RPV isolation and overpressure protection design features and design functions, 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 RPV isolation and overpressure protection design features and design functions to describe compliance with regulatory requirements and to describe the bases for any exemptions to regulatory requirements or approaches to regulatory guidance that may be referenced during 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 EVALUATION OF RPV ISOLATION

2.1 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 basic BWRX-300 safety design philosophy for the mitigation of LOCAs is built on utilization of inherent margins (e.g., larger water inventory) to eliminate system challenges, reduce number and size of RPV nozzles as compared to predecessor designs[[ ]]. The relatively large RPV volume of the BWRX-300, along with the relatively tall chimney region, provides a substantial reservoir of water above the core. This ensures the reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range following transients involving feedwater flow interruptions or LOCAs. [[ ]] These design features preserve reactor coolant inventory to ensure that adequate core cooling is maintained.

The large RPV volume also reduces the rate at which reactor pressurization occurs if the reactor is suddenly isolated from its normal heat sink. If isolation should occur, RPS is initiated to shut down the reactor and ICS is initiated to remove heat from the reactor. Heat from the reactor is rejected to the Isolation Condenser (IC) heat exchangers located within separate, large pools of water (the IC pools) positioned immediately above (and outside) the containment. [[ ]]

2.1.1 Reactor Pressure Vessel

The BWRX-300 RPV assembly consists of the pressure vessel, removable head, and its appurtenances, supports and insulation, and the reactor internals. The RPV instrumentation to monitor the conditions within the RPV is designed to cover the full range of reactor power operation. The RPV, together with its internals, provides guidance and support for the Fine Motion Control Rod Drives (FMCRDs).

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.

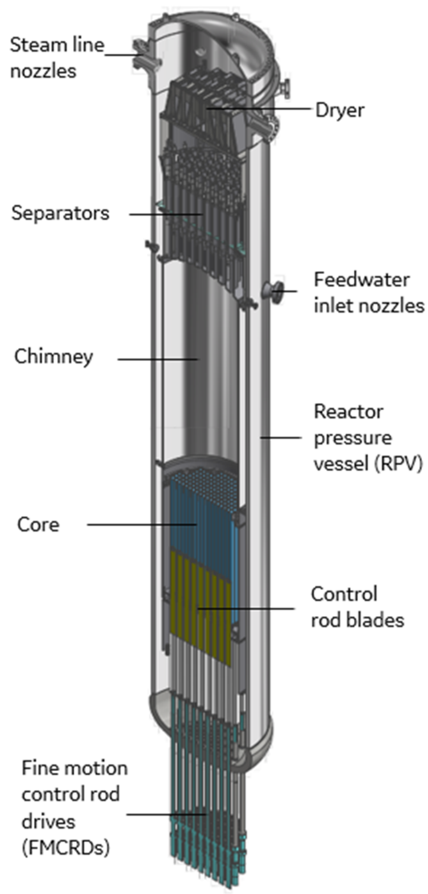

Figure 2-1: BWRX-300 Reactor Pressure Vessel and Internals

An increased internal flow path length, relative to forced circulation BWRs, is provided by a "chimney" in the space that extends from the top of the core to the entrance to the steam separator assembly. The chimney and steam separator assembly are supported by a shroud assembly that extends to the top of the core.

The major reactor internal components include:

- core (fuel, channels, control rods and instrumentation)
- core support and alignment structures (shroud, shroud support, top guide, core plate, control rod guide tube, Control Rod Drive (CRD) housings, and orificed fuel support)
- chimney
- chimney head and steam separator assembly
- steam dryer assembly
- feedwater spargers
- in-core guide tubes

Except for the Zircaloy in the reactor core, these reactor internals are stress corrosion resistant stainless steel or other high alloy steels.

The fuel assemblies (including fuel rods and channels), control rods, chimney head and steam separator assembly, steam dryers and in-core instrumentation assemblies are removable when the RPV is opened for refueling or maintenance.

2.1.2 Isolation Condenser System

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/evaporating water in the IC pool, which is vented to the atmosphere. The arrangement of one IC heat exchanger situated in an IC pool is shown in Figure 2-2.

[[

]]

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

The ICS is initiated automatically on high RPV pressure indicating an overpressure event or on signals indicating a LOCA. To start an IC train, the IC condensate return valve is opened whereupon the standing condensate drains into the reactor and the steam water interface in the IC tube bundle moves downward below the lower headers. [[

]] The IC pools are interconnected and 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.

2.2 General Overview of the Reactor Pressure Vessel Isolation Concept

[[

]]

[[

]]

Figure 2-3: RPV Isolation Valve Assembly (Example)

One of the design objectives of the BWRX-300 Reactor Coolant Pressure Boundary (RCPB) is to minimize the risks associated with LOCAs relative to the ESBWR design. Risk is minimized by the following:

- Reducing the number of nozzles,
- Reducing pipe lengths and nominal pipe diameters,
- Maximizing the elevation of the nozzles, and
- [[

  ]]

For the BWRX-300 the RPV nozzles are placed as high on the RPV as possible to limit the effect of a potential pipe break. [[

]] The main steam line nozzles are placed as high as possible on the RPV. The total number of nozzles are reduced from the ESBWR to the BWRX-300. [[

]]

Figure 2-4 shows a representation of the preliminary RPV assembly arrangement for the BWRX-300 and summarizes the relative locations of nozzles on the RPV assembly.

[[

]]

Figure 2-4: Preliminary BWRX-300 RPV Assembly Nozzles

Figure 2-5 shows a relative comparison of the RPV assembly for ABWR, ESBWR, and BWRX-300.

[[

]]

Figure 2-5: RPV Assembly for ABWR, ESBWR and the BWRX-300

2.3 Reactor Pressure Vessel Design Requirements

The BWRX-300 RPV is designed using the same codes and standards as the ESBWR RPV and with similar selection of design code accepted material specifications. This is described in the ESBWR Design Control Document (DCD), Tier 2, Section 5.3 [Reference 5.2]. Changes as a result of newer editions or published revisions of codes and standards used in the BWRX-300 design are subject to appropriate regulatory review and approval. The full details of the material specifications, and codes and standards, for the BWRX-300 are to be provided during future licensing activities.

2.4 Reactor Pressure Vessel Nozzle Design Requirements

The RPV nozzle design requirements for the BWRX-300 use the same design codes and standards, except for issue date, as the ESBWR, which are documented in the ESBWR DCD, Tier 2, Subsection 5.3.3.2.2, Reactor Vessel Design Data, for Reactor Vessel Nozzles [Reference 5.2].

There are some differences in the RPV nozzle designs between the BWRX-300 and the ESBWR. [[

]]

Design Requirements:

- All piping and valves connected to the nozzles shall be designed not to exceed the allowable loads on any nozzle.
- The feedwater inlet nozzles and IC condensate return nozzles shall be designed to account for stresses caused by cooler injection water.
- All nozzles shall be low alloy steel forgings; except the water level instrumentation nozzles.
- The design of the nozzles shall be in accordance with ASME Section III, Subsection NB and meet the applicable requirements of the vessel design documents.
- [[

]]

2.4.1 Connection of Reactor Pressure Vessel Isolation Valves to Reactor Vessel

[[

]] The BWRX-300 design requirements for identifying postulated pipe rupture locations and configurations inside containment conform to the guidance in Branch Technical Position (BTP) 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment." However, [[

]], most of the BTP 3-4, Part B, Item 1(ii) criteria do not apply. However, BTP 3-4, Part B. Item 1(ii) criteria generically involving design stress and fatigue limits and inservice inspection (ISI) guidelines are applicable.

[[

]] These [[

]] will be established during detailed design of the [[ ]] and provided during future licensing activities.

BTP 3-4, Part B, Item 1(ii)(1) specifies more conservative stress and fatigue limits for ASME Class 1 piping in containment penetration areas than those required for piping by ASME Code Section III, Paragraph NB-3653. The bases for these more conservative limits include limiting the stresses resulting from service loads (excluding those due to peak stresses) to within the material yield strength (i.e., elastic strains), and to ensure that the cumulative usage factor calculation account for the possibility of a faulty design or improperly controlled fabrication, installation errors, and unexpected modes of operation, vibration, and other structural degradation mechanisms.

[[

]] the design criteria given in ASME Code Section III, Paragraph NB-3230, provides greater margin against yielding due to service loads than do the rules of Paragraph NB-3653 for typical piping system materials, even when using the more restrictive limits of BTP 3-4, Part B, Item 1(ii)(1). Therefore, the imposition of more conservative stress limits of BTP 3-4, Part B, Item 1(ii)(1) is not justified.

[[

]]

[[

]] extending to the containment wall, the BWRX-300 design requirements include identifying postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, Part B, Item 1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2).

Where break locations are selected without the benefit of stress calculations, breaks are postulated at the piping welds to each fitting, valve, or welded attachment. Additionally, for a structure credited with separating a high-energy line from essential structures, systems, and components (SSCs), the separating structure is designed in accordance with BTP 3-4, Part B, Item 1(iii)(4) to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect on the structure. This is true even though the criteria described in BTP 3-4, Part B, Items 1(iii)(1) through (3) might not require the postulation of a break at that location.

2.5 Reactor Pressure Vessel Isolation Valve Design Requirements

GEH applies the following key factors to the selection of valves for the BWRX-300 RPV isolation valves design:

- The valve closure is a safety-related function.
- Compact valve and actuator assemblies are selected.
- Required Primary Containment Vessel (PCV) space allocation is minimized in proximity of the RPV.
- Electrical and digital controls are minimized inside the PCV.

Design Requirement:

- All BWRX-300 RPV isolation valves shall have a proven low leakage potential.

Design and administrative leakage limits are applied to valve selection during the BWRX-300 preliminary design and are based on plant design and event evaluations using offsite dose consequences compared to regulatory limits as well as containment design limits. The leakage criteria are analyzed as part of the plant safety analysis.

For BWRX-300 design, the application of motor-operated valves is constrained because there is no safety-related power supply other than limited Class 1E battery-stored power. Thus, motor-operated valves other than fail-closed magnetic-motor valves (i.e., solenoid operated valves) are not used for any RPV isolation valve applications.

Design Requirements:

- The RPV isolation valves for main steam line, feedwater, shutdown cooling, and reactor water cleanup shall fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means.
- [[
  ]] with valve actuators designed to maintain the valves in their as-is position by positive mechanical means.

A critical aspect of the valve and actuator selection to evaluate is the failure mode. The failure mode of the RPV isolation valves are determined based on the safety function of the connected system. [[

]]

The actuation signal for the RPV isolation valve closure is different for fail-close and fail-as-is isolation valves.

Design Requirements:

- The fail-close RPV isolation valves shall automatically close on high containment pressure indicating a LOCA.

- [[

]]

The ICS RPV isolation valves automatic isolation function uses logic and functionality similar to the ESBWR ICS containment isolation valves, which is described in the last two paragraphs of ESBWR DCD Tier 2 Subsection 5.4.6.2.2 [Reference 5.2]. However, the BWRX-300 Instrumentation and Control (I&C) system has three divisions of safety-related I&C. [[

]]

2.6 Reactor Pressure Vessel Isolation Valve Actuator Design Requirements

Design Requirement:

- The RPV isolation valves and actuators shall be operable during events when the containment pressure and temperature are elevated.

A key design requirement is control of the temperature at the valve-actuator interface in order to limit thermal effects on the actuator assembly. The RPV isolation valves are heated by process water or steam, which also elevates valve actuator temperatures above the local ambient. Valve and valve actuator designs are qualified in accordance with ASME QME-1 [Reference 5.5] to include evaluation of the local environmental conditions, including evaluation of the effects of heat transfer from the process water or steam and Design Basis Events. [[
]] The stem connection and actuator mounting method are studied to determine if thermal isolation needs to be implemented. High-temperature seals or lubricants are used for the actuators.

Design Requirement:

- Control devices (e.g., pilots) that rely on electric power may be located outside the PCV when practical.

Locating the control devices for the RPV isolation valve actuators that rely on electric power outside the PCV eliminates harsh Environmental Qualification (EQ) requirements.

2.7 Categories of Pipe Breaks

Steam and liquid line breaks are evaluated. The pipe breaks evaluated in the safety analysis are divided into two size categories:

- [[

]]

The largest steam line break is a main steam line break. The largest liquid line break is the feedwater line break. [[

]]

Design Requirements:

- [[

]]

For a postulated pipe break, the RPS performs the control of reactivity function by shutting down the core. [[

]]

The emergency core cooling system (ECCS) evaluation model for the BWRX-300 is to be developed using previously approved methodologies used in the ECCS performance analyses for the ESBWR as modified using BWRX-300 specific design requirements and parameters. Final ECCS performance analyses are to be completed during future licensing activities using an NRC-approved BWRX-300 ECCS evaluation model. Methodology for containment response is described in LTR NEDC-33911P, BWRX-300 Containment Performance [Reference 5.6].

Table 2-1 summarizes the pipe break categories and the key assumptions for each case.

Table 2-1: Pipe Break Categories

Break Type	[[ ]] Small Breaks	[[ ]] Large Breaks
Steam	[[ ]]	[[ ]]
Liquid	[[ ]]	[[ ]]

Small leaks in the RCPB that are [[ ]] (e.g., leakage from flanges or cracks in piping or other components) do not exceed the capability of the nonsafety-related high-pressure CRD system used as normal reactor coolant makeup during power operations. The maximum allowed leakage rate for continuing power operation is stipulated in the plant Technical Specifications. For leakage greater than the maximum allowed leakage rate, automatic reactor scram and automatic actuation of the [[ ]] is not anticipated for most of these small leaks because the normal means of makeup from the high-pressure CRD system and feedwater maintains the level in the normal operating range. However, these small leaks which do not exceed the capability of the nonsafety-related high-pressure CRD system are evaluated using specified acceptable fuel design limits rather than the BWRX-300 acceptance criteria in response to a LOCA. See Subsection 4.1.11 for further discussion of these small leaks.

2.8 LOCA Acceptance Criteria

The BWRX-300 acceptance criteria in response to a LOCA are that the reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, which bound the acceptance criteria of 10 CFR 50.46(b). Maintaining reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range ensures that:

- No significant fuel cladding heatup occurs.
- No significant fuel cladding oxidization occurs.
- No significant fuel cladding hydrogen generation occurs.
- No significant changes in core geometry occurs.
- Long term cooling to remove decay heat and maintain the core temperature to acceptably low values occurs.

3.0 TECHNICAL EVALUATION OF OVERPRESSURE PROTECTION

3.1 General Overview of the Overpressure Protection Concept

The BWRX-300 integrated overpressure protection during operation at power is ensured by application of the RPS to shut down the reactor [[
]]

As with other BWRs, the BWRX-300 does not operate in water-solid conditions and therefore is not subject to low-temperature operation requiring special overpressure protection. Additionally, for periodic leak testing while shutdown, the system is not subject to pressurization from the reactor, and special test conditions are established to allow for pressure control.

3.1.1 Reactor Protection System Design Requirements

The BWRX-300 RPS is based on the ESBWR RPS design [Reference 5.7]. The safety-related RPS performs the control of reactivity function for overpressure protection by initiating an automatic reactor shutdown by rapid insertion of control rods (scram) if monitored system variables exceed pre-established limits. This action prevents fuel damage and limits system pressure, thus aiding in the containment of radioactive materials function for overpressure protection.

The RPS implements the reactor trip functions. The RPS is the overall collection of instrument channels, trip logics, trip actuators, manual controls, and scram logic circuitry that initiates rapid insertion of control rods to shut down the reactor to help ensure established safety criteria are met.

The RPS is based on a fail-safe design philosophy. The RPS design provides reliable, single failure proof capability to automatically or manually initiate a reactor scram while maintaining protection against unnecessary scrams resulting from single failures. This is accomplished through the combination of fail-safe and fault-tolerant equipment design, and a two-out-of-three voting logic algorithm.

Design Requirements:

- RPS shall shutdown the reactor to ensure overpressure protection design requirements are met.
- RPS scram signals shall be established to ensure overpressure protection design requirements are met.
- RPS trip function performance shall be established to ensure overpressure protection design requirements are met.
- RPS functions to ensure overpressure protection design requirements are met shall be single failure proof.

3.1.2 Isolation Condenser System Design Requirements

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 and with a minimal loss of reactor coolant following reactor shutdown and isolation. [[

]] These functions aid in the containment of radioactive materials function for overpressure protection.

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. [[
]] with valve actuators designed to maintain the valves in their open position by positive mechanical means.

Design Requirements:

- [[

]]

3.2 ASME Requirements for Overpressure Protection

Overpressure protection for the RCPB is in compliance with ASME B&PV Code, Section III, Article NB-7000 [Reference 5.1]. Paragraph NB-7120 requires that overpressure protection of the components shall be provided by any of the following as an integrated overpressure protection:

a. The use of pressure relief devices and associated pressure sensing elements

b. The use of the reactor shutdown system

c. A design without pressure relief devices such that for each component in the protected system

    1. The overpressure does not exceed 1.1 times design pressure for the expected system pressure transient condition

    2. The calculated stress intensity and other design limitations for Service Level C are not exceeded for the unexpected system excess pressure transient condition

Overpressure protection for the BWRX-300 is provided in accordance with ASME B&PV Code, Section III, subparagraphs [[ ]].

4.0 REGULATORY EVALUATION

4.1 10 CFR 50 Regulations

4.1.1 10 CFR 50.34(f)

10 CFR 50.34(f), Additional Three Mile Island (TMI) related requirements, requires that each applicant for a design certification, design approval, combined license, or manufacturing license under Part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section, except for paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Although it is not yet determined whether a 10 CFR 52 license application may be submitted for a BWRX-300, these requirements are evaluated herein. 10 CFR 50.34(f)(1) states that to satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of these studies are factored into the final design of the facility, and that the studies must be submitted as part of the final safety analysis report. 10 CFR 50.34(f)(2) states that to satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. The following requirements are evaluated as they are related to [[

]], as being required following the worst-case LOCA to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b):

- Regulatory Requirement: 10 CFR 50.34(f)(1)(v) requires that an evaluation be performed of the safety effectiveness of providing for separation of High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with High Pressure Core Spray (HPCS) systems in lieu of high pressure coolant injection systems, substitute the words, “high pressure core spray” for “high pressure coolant injection” and “HPCS” for “HPCI”) (Applicable to BWR's only). (II.K.3.13)

  Statement of Compliance: The BWRX-300 does not include safety-related high-pressure injection systems, including RCIC, HPCI, and HPCS systems, because [[

  ]] Therefore, this requirement is not technically relevant to the BWRX-300.

- Regulatory Requirement: 10 CFR 50.34(f)(1)(vi) requires that a study be performed to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWR’s only). (II.K.3.16)

Statement of Compliance: The design features of the BWRX-300 RCPB include the use of the RPS and [[ ]] for overpressure protection, [[ ]]. The intent of this requirement is to minimize the potential for loss of reactor coolant through inadvertent operation of relief and safety valves, [[ ]] Therefore, this requirement is not technically relevant to the BWRX-300.

- Regulatory Requirement: 10 CFR 50.34(f)(1)(vii) requires that a feasibility and risk assessment study be performed to determine the optimum ADS design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWR's only). (II.K.3.18)

  Statement of Compliance: Automatic actuation of [[ ]] performs the function to mitigate the effects of a LOCA. [[ ]] is a one-time action to open or close appropriate valves during accident response which may also be initiated manually as a one-time action if necessary. However, manual actuation is not assumed in the accident analysis. Therefore, this requirement is not technically relevant to the BWRX-300.

- Regulatory Requirement: 10 CFR 50.34(f)(1)(viii) requires that a study be performed of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (Applicable to BWR's only). (II.K.3.21)

  Statement of Compliance: The BWRX-300 does not include core spray and low pressure coolant injection systems, because automatic actuation of [[ ]] are sufficient to mitigate the effects of a LOCA. [[ ]] is a one-time action to open or close appropriate valves during accident response and does not require the use of active pumps. Therefore, this this requirement is not technically relevant to the BWRX-300.

- Regulatory Requirement: 10 CFR 50.34(f)(1)(ix) requires that a study be performed to determine the need for additional space cooling to ensure reliable long-term operation of the RCIC and HPCI systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (Applicable to BWR's only). (II.K.3.24)

  Statement of Compliance: The BWRX-300 does not include RCIC and HPCI systems, because automatic actuation of [[ ]] are sufficient to mitigate the effects of a LOCA. [[ ]] is a one-time action to open or close appropriate valves located inside the PCV during accident response which are environmentally qualified to operate under post-accident conditions, and does not require the use of active pumps or other equipment requiring space cooling. Therefore, this requirement is not technically relevant to the BWRX-300.

- Regulatory Requirement: 10 CFR 50.34(f)(1)(x) requires that a study be performed to ensure that the ADS, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for nonsafety-related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWR's only). (II.K.3.28)

  Statement of Compliance: The BWRX-300 does not include [[ ]], because automatic actuation of [[ ]] are sufficient [[ ]] to mitigate the effects of a LOCA. [[ ]] is a one-time action to open or close appropriate valves located inside the PCV during accident response taking no credit for nonsafety-related equipment or instrumentation. The valves and actuators are environmentally qualified to operate under post-accident conditions. Therefore, this requirement is not technically relevant to the BWRX-300.

- Regulatory Requirement: 10 CFR 50.34(f)(2)(x) requires that a test program and associated model development be provided and tests conducted to qualify RCS relief and safety valves and, for PWRs, Power-Operated Relief Valve (PORV) block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of ATWS conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. (II.D.1)

  Statement of Compliance: The design features of the BWRX-300 RCPB include the use of the RPS [[ ]] for overpressure protection, [[ ]]. Because the [[ ]], this requirement is not technically relevant to the BWRX-300. However, the [[ ]] perform a similar function during an ATWS event in providing overpressure protection and will be qualified by testing to perform their safety-related design function for fluid conditions expected under operating conditions, transients and accidents including ATWS events.

- Regulatory Requirement: 10 CFR 50.34(f)(2)(xi) requires that direct indication of relief and safety valve position (open or closed) be provided in the control room. (II.D.3)

  Statement of Compliance: The design features of the BWRX-300 RCPB include the use of the RPS and [[ ]] for overpressure protection, [[ ]]. The intent of this requirement is to provide indication to the operator [[ ]]. Therefore, direct indication of relief and safety valve position (open or closed) provided in the control room is not technically relevant to the BWRX-300. However, direct position indication of the [[ ]] are provided in the BWRX-300 design.

Based on the above discussions, these requirements are not technically relevant to the BWRX-300. These statements of compliance may be used as the bases for this conclusion during 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.

4.1.2 10 CFR 50.46

10 CFR 50.46, Acceptance criteria for Emergency Core Cooling Systems (ECCS) for light-water nuclear power reactors, includes the following requirements:

- Regulatory Requirement: 10 CFR 50.46(a)(1)(i) requires that each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy or ZIRLO cladding must be provided with an emergency core cooling system that must be designed so that its calculated cooling performance following postulated LOCAs conforms to the criteria set forth in paragraph (b) of this section. As further defined in 10 CFR 50.46(c)(1), LOCAs are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

  Statement of Compliance: The design features of the BWRX-300 used to comply with this requirement include [[ ]] for all postulated LOCA break sizes, in conjunction with the [[ ]] for postulated LOCA break sizes [[ ]].

  - The required ECCS design functions of the [[ ]] to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(4) as further described below. Another design function of the [[ ]] is for overpressure protection which is further required to maintain a coolable geometry to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance 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 BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria of 10 CFR 50.46(b)(5) and only requires operator action to [[ ]] after approximately seven days.

  - [[ ]] for any breaks

that would result in a loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system, which for the BWRX-300 includes LOCA break sizes [[ ]]. The worst-case single failure of [[ ]] or the worst-case single failure affecting the [[ ]] does not prevent fulfillment of the required ECCS design functions. The [[ ]] are to be determined during the final ECCS performance analyses to be completed during future licensing activities.

Based on the above evaluation, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS consistent with the definition of a LOCA in 10 CFR 50.46(c)(1).

Although this is a non-traditional approach for the design of the ECCS for past LWRs, no [[ ]] is needed following the worst-case LOCA to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b). The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46.

- Regulatory Requirement: 10 CFR 50.46(a)(1)(i) requires that ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model.

  As further required in 10 CFR 50.46(a)(1)(ii), alternatively an ECCS evaluation model may be developed in conformance with the required and acceptable features of Appendix K ECCS Evaluation Models. 10 CFR 50.46(a)(2) defines an evaluation model as the calculational framework for evaluating the behavior of the reactor system during a postulated LOCA. It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

Statement of Compliance: The ECCS evaluation model for the BWRX-300 is to be developed using previously approved methodologies used in the ECCS performance analyses for the ESBWR as modified using BWRX-300 specific design requirements and parameters. The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities using an NRC-approved BWRX-300 ECCS evaluation model which includes reasonably conservative methods. Because of the BWRX-300 acceptance criteria being applied to bound the 10 CFR 50.46(b) acceptance criteria, uncertainties will be addressed in the BWRX-300 ECCS evaluation model to verify that there is a high level of probability that the BWRX-300 acceptance criteria would not be exceeded rather than the 10 CFR 50.46(b) acceptance criteria. The BWRX-300 evaluation model will not use the alternatives provided in 10 CFR 50 Appendix K.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(a)(1) and 10 CFR 50.46(a)(2).

- Regulatory Requirement: 10 CFR 50.46(b)(1), Peak cladding temperature, requires that the calculated maximum fuel element cladding temperature shall not exceed 2200°F.

  Statement of Compliance: The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant fuel cladding heatup occurs, such that the calculated maximum fuel element cladding temperature does not exceed the acceptance criterion of 2200°F.

  The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(1).

- Regulatory Requirement: 10 CFR 50.46(b)(2), Maximum cladding oxidation, requires that the calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

Statement of Compliance: The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant fuel cladding oxidization occurs, such that the calculated total oxidation of the cladding nowhere exceeds the acceptance criterion of 0.17 times the total cladding thickness before oxidation.

The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(2).

- Regulatory Requirement: 10 CFR 50.46(b)(3), Maximum hydrogen generation, requires that the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

  Statement of Compliance: The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant fuel cladding hydrogen generation occurs, such that the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed the acceptance criterion of 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

  The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(3).

- Regulatory Requirement: 10 CFR 50.46(b)(4), Coolable geometry, requires that the changes in core geometry shall be such that the core remains amenable to cooling.

  Statement of Compliance: The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant changes in core geometry occur, such that the acceptance criterion of the core remaining amenable to cooling is met.

  The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(4).

- Regulatory Requirement: 10 CFR 50.46(b)(5), Long-term cooling, requires that after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Statement of Compliance: The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that long-term cooling removes decay heat and maintains the core temperature to acceptably low values, such that after any calculated successful initial operation of the ECCS, the acceptance criteria of the calculated core temperature being maintained at an acceptably low value and decay heat being removed for the extended period of time required by the long-lived radioactivity remaining in the core are met.

For the BWRX-300, [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed power without any further need of onsite or offsite electric power system operation.

The selected long-term cooling timeframe of [[ ]] is sufficient to maintain reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range for both [[

]] Following [[

]], the [[ ]] continues to provide long-term cooling to meet the BWRX-300 acceptance criteria in response to a LOCA and only requires operator action to [[ ]] after approximately seven days.

The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(5).

- Regulatory Requirement: 10 CFR 50.46(d) states that the requirements of this section are in addition to any other requirements applicable to ECCS set forth in this part. The criteria set forth in paragraph (b), with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of Appendix A.

  Statement of Compliance: Compliance with these additional requirements are addressed in the discussions below.

4.1.3 10 CFR 50.55a

10 CFR 50.55a, Codes and standards, in 10 CFR 50.55a(a), Documents approved for incorporation by reference, lists the standards that have been approved for incorporation by reference by the Director of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51.

- Regulatory Requirement: 10 CFR 50.55a(a) includes standards that are required for evaluation of the RPV isolation valves. This rule establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components of BWR nuclear power plants by requiring conformance with appropriate editions of specified published industry codes and standards.

Statement of Compliance: The BWRX-300 RPV isolation valves and overpressure protection design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, including any application for a construction permit under 10 CFR 50 or design certification application under 10 CFR 52. These requirements are to be implemented during detailed design of the safety-related components of the [[ ]].

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.55a.

4.1.4 10 CFR 50 Appendix A, GDC 1

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 1, Quality standards and records, requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

  Statement of Compliance: The BWRX-300 RPV isolation and overpressure protection design features, [[ ]] are to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed in accordance with generally recognized codes and standards, and under an approved quality assurance program with approved control of records.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 1.

4.1.5 10 CFR 50 Appendix A, GDC 2

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 2, Design bases for protection against natural phenomena, requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Statement of Compliance: The BWRX-300 RPV isolation and overpressure protection design features, [[ ]] are to be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. Specific design requirements for the [[ ]] used to verify the capability to perform their safety functions, and the natural phenomena and effects evaluated, will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 2.

4.1.6 10 CFR 50 Appendix A, GDC 4

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 4, Environmental and dynamic effects design bases, requires that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

  Statement of Compliance: As stated in this LTR, a design requirement of the BWRX-300 is that the SSCs required to mitigate a LOCA shall be operable in the environmental conditions (PCV pressure, temperature, radiation, etc.) following a LOCA. In addition, the dynamic effects of postulated pipe breaks are to be evaluated in the BWRX-300 design. As described in this LTR, [[ ]] the BWRX-300 design requirements include evaluation of the acceptable criteria to identify postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment,” as discussed in Subsection 2.4.1 of this LTR.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 4.

4.1.7 10 CFR 50 Appendix A, GDC 14

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 14, Reactor coolant pressure boundary, requires that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Statement of Compliance: [[

]] This results in an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture, in compliance with this criterion. For piping that have [[

]]. Further design details are to be described during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 14.

4.1.8 10 CFR 50 Appendix A, GDC 15

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 15, RCS design, requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

  Statement of Compliance: Overpressure protection for the BWRX-300 is provided in accordance with ASME B&PV Code, Section III, Paragraph NB-7120 Subparagraphs [[ ]]. The combination of RPS and [[ ]] design features ensure that the acceptance criteria for each component in the protected system are met including 1) the overpressure does not exceed 1.1 times design pressure for the expected system pressure transient condition, and 2) the calculated stress intensity and other design limitations for Service Level C are not exceeded for the unexpected system excess pressure transient condition, with further design details to be described during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 15.

4.1.9 10 CFR 50 Appendix A, GDC 30

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 30, Quality of reactor coolant pressure boundary, requires that components which are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

  Statement of Compliance: The components of the RCPB, including the ICS and RPV isolation valves, and the overpressure protection equipment [[ ]] are to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed in accordance with generally recognized codes and standards, and under an approved quality assurance program with approved control of records, as required by 10 CFR 50.55a and 10 CFR 50 Appendix A, GDC 1. In addition, means are to be provided to detect and identify the location of the source of reactor coolant leakage, including the components of the ICS and

RPV isolation valves, for components of the RCPB, with further design details to be described during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 30.

4.1.10 10 CFR 50 Appendix A, GDC 31

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 31, Fracture prevention of reactor coolant pressure boundary, requires that the RCPB shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

  Statement of Compliance: The components of the RCPB, including the ICS and RPV isolation valves, are to be designed with sufficient margin to assure that these requirements are met, with further design details to be described during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 31.

4.1.11 10 CFR 50 Appendix A, GDC 33

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 33, Reactor coolant makeup, requires that a system to supply reactor coolant makeup for protection against small breaks in the RCPB shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.
- Statement of Compliance: GDC 33 applies to small leaks in the RCPB that are [[ ]] (e.g., leakage from flanges or cracks in piping or other components), and which do not exceed the capability of the nonsafety-related high-pressure CRD system used as normal reactor coolant makeup during power operations. The maximum allowed leakage rate for continuing power operation is stipulated in the plant Technical Specifications. For leakage greater than the maximum allowed leakage rate, automatic reactor scram and automatic actuation of the [[ ]] is not anticipated for most of these small leaks because the normal means of makeup from the high-pressure CRD system and feedwater maintains the level in the normal operating range. However, these small leaks which do not exceed the capability of the nonsafety-related high-pressure CRD system are evaluated using specified acceptable fuel design limits rather than the BWRX-300 acceptance criteria in response to a LOCA.

The high-pressure CRD system pumps can operate using either onsite electric power system operation (assuming offsite power is not available) backed up by nonsafety-related standby diesel generators and offsite electric power system operation (assuming onsite power is not available).

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 33.

4.1.12 10 CFR 50 Appendix A, GDC 35

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 35, Emergency core cooling, requires a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Statement of Compliance: As previously described, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS in compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling, which is assured for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS through the use of [[

]].

The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, such that the performance of the [[ ]] is sufficient to ensure that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

[[

]]

The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA (i.e., reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range) will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 35.

4.1.13 10 CFR 50 Appendix A, GDC 37

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 37, Testing of emergency core cooling system, requires that the emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

  Statement of Compliance: As previously described, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS in compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling, which is assured for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS through the use of [[ ]].

Specific requirements for periodic pressure and functional testing of the [[ ]] to assure (1) the structural and leaktight integrity of these components, (2) the operability and performance of these active components, and (3) the operability of the systems as a whole containing the [[ ]] will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 37.

4.2 Regulatory Guides

4.2.1 Regulatory Guide 1.26

Regulatory Guide (RG) 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, Rev. 5, describes methods acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50 Appendix A, GDC 1, Quality Standards and Records, with regard to a quality classification system related to specified national standards that may be used to determine quality standards acceptable to the NRC Staff for components containing water, steam, or radioactive material in light-water-cooled nuclear power plants. The design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50 Appendix A, GDC 1. The RPV isolation valves and the components of the ICS are classified in conformance with the guidance provided in RG 1.26.

Therefore, the BWRX-300 design conforms to the guidance for the RPV isolation valves and the ICS, including regulatory positions of RG 1.26.

4.2.2 Regulatory Guide 1.29

RG 1.29, Seismic Design Classification, Rev. 5, describes methods acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50.55a(h), 10 CFR 50 Appendix A, GDC 2, Design Bases for Protection Against Natural Phenomena, and 10 CFR 50 Appendix S, Earthquake Engineering Criteria for Nuclear Power Plants, for use in identifying and classifying those features of LWR nuclear power plants that must be designed to withstand the effects of the Safe-Shutdown Earthquake (SSE).

The design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a(h), 10 CFR 50 Appendix A, GDC 2, and 10 CFR 50 Appendix S. The components of the ICS and RPV isolation valves are classified as Seismic Class I in conformance with the guidance provided in RG 1.29.

Therefore, the BWRX-300 design conforms to the guidance, including regulatory positions of RG 1.29.

4.2.3 Regulatory Guide 1.45

RG 1.45, Guidance on Monitoring and Responding to RCS Leakage, Rev. 1, describes methods acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50 Appendix A, GDC 14, RCPB, and 10 CFR 50 Appendix A, GDC 30, Quality of RCPB, with regard to selecting reactor coolant leakage detection systems, monitoring for leakage, and responding to leakage for light-water-cooled reactors. This guidance additionally cites 10 CFR 50.55a, Codes and Standards, which requires the performance of ISI and testing of nuclear power plant components so that the concept of defense in depth is applied to provide assurance that the structural integrity of the RCPB is maintained.

Subsections 4.1.4 and 4.1.7 describe how the design of the [[ ]] complies with the requirements of 10 CFR 50 Appendix A, GDC 1 and GDC 14. Additionally, the means for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage in compliance with the requirements of 10 CFR 50 Appendix A, GDC 30, and the requirements for ISI and inservice testing (IST) of the [[ ]] in compliance with the requirements of 10 CFR 50.55a, are to be demonstrated during future licensing activities.

Therefore, the BWRX-300 design conforms to the guidance, including regulatory positions of RG 1.45.

4.2.4 Regulatory Guide 1.84

RG 1.84, Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, Rev. 37, lists the ASME B&PV Section III Code Cases that the NRC has approved for use as voluntary alternatives to the mandatory ASME B&PV Code provisions that are incorporated by reference into 10 CFR 50. This applies to reactor licensees subject to 10 CFR 50.55a, Codes and Standards. These ASME B&PV Section III Code Cases are acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50 Appendix A, GDC 1, Quality standards and records, and 10 CFR 50 Appendix A, GDC 30, Quality of reactor coolant pressure boundary, and the requirements of 10 CFR 50.55a(c), Reactor Coolant Pressure Boundary, which requires, in part, that components of the RCPB must be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of the ASME B&PV Section III Code or equivalent quality standards.

Subsections 4.1.4 and 4.1.9 describe how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50 Appendix A, GDC 1, and 10 CFR 50 Appendix A, GDC 30, respectively. Subsection 4.1.3 describes how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a. Compliance with the requirements of 10 CFR 50.55a, including the use of ASME B&PV Section III Code Cases endorsed in RG 1.84 where necessary, is to be demonstrated during future licensing activities.

Therefore, the BWRX-300 design will conform to the guidance, including regulatory positions of RG 1.84.

4.2.5 Regulatory Guide 1.147

RG 1.147, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, Rev. 19, lists the ASME B&PV Section XI Code Cases that the NRC has approved for use as voluntary alternatives to the mandatory ASME B&PV Code provisions that are incorporated by reference into 10 CFR 50. This applies to reactor licensees subject to 10 CFR 50.55a, Codes and Standards. These ASME B&PV Section XI Code Cases are acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50 Appendix A, GDC 1, Quality standards and records, and 10 CFR 50 Appendix A, GDC 30, Quality of reactor coolant pressure boundary, and the requirements of 10 CFR 52.79(a)(11) which requires the final safety analysis report to include “a description of the program(s), and their implementation, necessary to ensure that the systems and components meet the requirements of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants in accordance with 50.55a of this chapter.” In 10 CFR 50.55a(a)(1)(ii), the NRC references the latest editions and addenda of ASME B&PV Code Section XI that the agency has approved for use.

Section 4.1.3 describes how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a. However, the requirements of ASME B&PV Code Section XI, Division 1, specifically apply during construction and operation activities of nuclear power plants for performance of ISI activities, and do not apply during the design of the BWRX-300. Therefore, compliance with the requirements of 10 CFR 50.55a for the conduct of ISI activities, including the use of ASME B&PV Section XI Code Cases endorsed in RG 1.147 where necessary, is to be demonstrated during future licensing activities.

Based on this discussion, the guidance of RG 1.147 does not apply to the BWRX-300 design phase in meeting the requirements of 10 CFR 50.55a.

4.2.6 Regulatory Guide 1.192

RG 1.192, Operation and Maintenance Code Case Acceptability, ASME OM Code, Rev. 3, lists Code Cases associated with the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that the NRC has approved for use as voluntary alternatives to the mandatory ASME OM Code provisions that are incorporated by reference into 10 CFR 50. This applies to reactor licensees subject to 10 CFR 50.55a, Codes and Standards. These ASME OM Code Cases are acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50 Appendix A, GDC 1, Quality standards and records, and 10 CFR 50 Appendix A, GDC 30, Quality of reactor coolant pressure boundary, the requirements of 10 CFR 50.55a(f) which requires, in part, that Class 1, 2, and 3 components and their supports meet the requirements of the ASME OM Code or equivalent quality standards, and the requirements of 10 CFR 52.79(a)(11) which requires the final safety analysis report to include “a description of the program(s), and their implementation, necessary to ensure that the systems and components meet the requirements of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants in accordance with 50.55a of this chapter.” In 10 CFR 50.55a(a)(1)(iv), the NRC references the latest editions and addenda of ASME OM Code that the agency has approved for use.

Section 4.1.3 describes how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a. However, the requirements of ASME OM Code specifically apply during operation and maintenance activities of nuclear power plants for performance of IST activities, and do not apply during the design of the BWRX-300. Therefore, compliance with the requirements of 10 CFR 50.55a for the conduct of IST activities, including the use of ASME OM Code Cases endorsed in RG 1.192 where necessary, is to be demonstrated during future licensing activities.

Based on this discussion, the guidance of RG 1.192 does not apply to the BWRX-300 design phase in meeting the requirements of 10 CFR 50.55a.

4.3 NUREG-0800 Standard Review Plan Guidance

4.3.1 Standard Review Plan 3.9.6

Standard Review Plan (SRP) 3.9.6, Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints, Rev. 4, states that the areas of review include the functional design and qualification provisions and IST programs for safety-related pumps, valves, and dynamic restraints (snubbers) designated as Class 1, 2, or 3 under ASME B&PV Code Section III.

As described in Section 4.1, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 2, GDC 4, GDC 14, GDC 15, and GDC 37, and the requirements of 10 CFR 50.55a, during detailed design activities with specific requirements for the [[ ]] to be provided during future licensing activities. In addition, Section 4.1.3 describes that the BWRX-300 RPV isolation valves and overpressure protection design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, including any application for a construction permit under 10 CFR 50 or design certification application under 10 CFR 52. These requirements are to be implemented during detailed design of the safety-related components of the [[ ]]. Therefore, the existing SRP provides adequate guidance to use 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.

4.3.2 Standard Review Plan 5.2.2

SRP 5.2.2, Overpressure Protection, Rev. 3, states that the areas of review include the application of relief and safety valves and the RPS that ensures overpressure protection for the RCPB during operation at power. SRP 5.2.2 also discusses the application of pressure-relieving systems that function during low-temperature operation ensures overpressure protection for the RCPB during low-temperature operation of the plant (startup, shutdown).

The design features of the BWRX-300 RCPB include the use of the RPS and [[ ]] for overpressure protection, [[ ]]. As this is a non-traditional approach to meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 15, GDC 30, and GDC 31, alternate guidance applicable to SRP 5.2.2 for the BWRX-300 is recommended to provide guidance to use 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.

Specific discussions under Section I, Areas of Review, that are affected by the [[ ]] include the following:

- I.1.A. - For BWRs, the area of review for operation at power includes relief and safety valves on the main steamlines and piping from these valves to the suppression pool. The BWR design also may incorporate interfacing systems, such as an IC, to prevent challenges to the relief and safety valves during normal operations. The BWR description of the basic design concept; the systems, subsystems, and support systems providing overpressure protection to the RCPB; the components and instrumentation employed in these systems; and process and instrumentation diagrams should be reviewed for power operation.

It is recommended that this area of review should include [[ ]].

- Review Interfaces, item 8 – For BWRs, review of the IC for sufficient capacity to preclude actuation of the overpressure protection system (under SRP Section 5.4.13).

  This area of review should be revised to require review of the [[ ]].

- Review Interfaces, item 9 - For BWRs, review of the suppression pool capability to condense and cool the discharge from the safety valves (under SRP Section 6.2.1.1.C).

  This area of review [[ ]].

Specific discussions under Section II, Acceptance Criteria, for generic acceptance criteria that are affected by the [[ ]], include the following:

- Requirements, item 3 - 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi) require that RCS relief and safety valves meet TMI Action Plan Items II.D.1 and II.D.3 of NUREG-0737.

  These acceptance criteria should be revised, because the BWRX-300 [[ ]]. Refer to Subsection 4.1.1 of this LTR regarding the requirements of 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).

- Requirements, item 4 - 10 CFR 52.47(a)(8) provides the requirement for design certification reviews to comply with the technically relevant portions of the TMI requirements in 10 CFR 50.34(f).

  These acceptance criteria may be retained, although a reference to revising the review guidance for 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi) should be added. Refer to Subsection 4.1.1 of this LTR regarding the requirements of 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).

- Requirements, item 5 - 10 CFR 52.79(a)(17) provides the requirement for COL applications to comply with the technically relevant information in 10 CFR 50.34. This includes the TMI-related requirements specified by 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).

  These acceptance criteria may be retained, although the second sentence should be deleted and replaced with a reference to revising the review guidance for 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi). Refer to Subsection 4.1.1 of this LTR regarding the requirements of 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).

Specific SRP acceptance criteria that are affected by the [[ ]] include the following:

- 2.A. - For overpressure protection during power operation of the BWR reactor, the designs of the pilot-operated relief valves with auxiliary actuation devices, ICs, or other pressure dissipation systems should have sufficient capacity to preclude actuation of safety valves during normal operational transients when assuming the following conditions at the plant:

  i. The reactor is operating at the licensed core thermal power level.

  ii. All system and core parameters have values within normal operating range that produce the highest anticipated pressure.

  iii. All components, instrumentation, and controls function normally.

  These acceptance criteria should be revised to require review of the [[

  ]].

- 2.B. - The design of safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe AOO with reactor scram, as specified by ASME Code Article NB-7000. Sufficient available margin should account for uncertainties in the design and operation of the plant, assuming the following:

  i. The reactor is operating at a power level that will produce the most severe overpressurization transient.

  ii. All system and core parameters have values within normal operating range, including uncertainties and technical specification limits that produce the highest anticipated pressure.

  iii. The second safety-grade signal from the RPS initiates the reactor scram.

  iv. The discharge flow is based on the rated capacities specified in ASME III for each type of valve.

  v. The design of the safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe infrequent event, as specified by ASME Code Article NB-7000.

  These acceptance criteria should be revised to require review of the [[

  ]].

- 2.C. - A single malfunction or failure of an active component should not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions, including loss of offsite power.

  Full credit is allowed for spring-loaded safety valves designed in accordance with the requirements of ASME Code Article NB-7511.1.

  These acceptance criteria should be revised to [[

  ]].

- 7. - TMI Action Plan Requirements. Section II.D.1 of the TMI Action Plan requires an applicant submit a plant specific report regarding relief and safety valve testing. Section II.D.3 of the TMI Action Plan requires that relief and safety valves be provided with direct valve position indication. Generic Letters No. 82-16 and 83-02 requires Sections II.D.1 and II.D.3 be covered by technical specifications while NUREG-0737 Section II.K.3.3 specifies reporting for Section II.D.1 and II.D.3.

  These acceptance criteria should be revised to require review of the [[

  ]]. Refer to Subsection 4.1.1 of this LTR regarding the requirements of 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).

Specific discussions under Section III, Review Procedures, that are affected by the [[ ]], include the following:

- 1.A.i. - The reviewer examines the piping and instrumentation diagrams to determine the number, type, and location of the relief and safety valves on the BWR RCS main steamlines and on the primary side of any auxiliary or emergency system that interfaces with the RCS. The reviewer also analyzes the functions of other pressure dissipation systems, such as ICs.

  These review procedures should be revised to require examination of the [[

  ]].

- 1.A.ii. - The reviewer identifies all other functions of the components, instruments, or controls used for overpressure protection and the interfaces with all other systems. This includes any blowdown or heat dissipation systems connected to the discharge side of any pressure-relieving devices such as the suppression pool. The reviewer determines the effects of these other functions or systems on the operation of the overpressure protection system.

  These review procedures should be revised to [[

  ]].

- 1.A.iii. - The reviewer identifies the capacities, setpoints, and setpoint tolerances for all relief and safety valves or other overpressure protection system devices. The reviewer verifies that these constraints are adequate to provide overpressure protection to the RCPB at critical values of pressure and temperature based on RCPB material parameters. The reviewer identifies allowable power levels with one or more inoperable relief and safety valves to ensure that they are suitably conservative, as specified in RS-001, and confirms that the plant technical specifications limit power operation as appropriate.

  These review procedures should be revised to require review of the [[

]].

- 2.A. - Tests for relief and safety valves operability are scheduled to be conducted as specified in Section III of the ASME Code Article NB-7000.

  These review procedures should be revised to verify appropriate tests for [[

  ]].

- 3.D. - Verify compliance with TMI Action Plan Item II.K.3.3 of NUREG-0737 regarding reporting of relief and safety valves challenges and failures. Generic Letters No. 82-16 and 83-02 provide descriptions of this NUREG-0737 item, include guidance regarding appropriate technical specifications to address the reporting requirements of II.K.3.3 of Section 5.6.4 of Standard Technical Specifications NUREG-1430 through NUREG-1434 regarding monthly operating reports, and offer related guidance on an appropriate technical specification to address this issue for those applicants implementing improved technical specifications.

  These review procedures should be revised, because the BWRX-300 [[
  ]]. Refer to Subsection 4.1.1 of this LTR regarding the requirements of 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).

Specific discussions under Section IV, Evaluation Findings, that are affected by the [[
]]
include the following:

- 1. BWRs. The overpressure protection system prevents overpressurization of the RCPB under the most severe transients and limits the reactor pressure during normal operational transients. The ICS provides overpressure protection.

  The relief and safety valves located on the ____ main steamlines between the reactor vessel and the first isolation valve inside the drywell provide defense-in-depth. The relief and safety valves are distributed among the ____ main steamlines such that a single accident cannot disable the automatic overpressure protection function. The valves discharge through piping to the suppression pool. The valves have setpoints that range from ______ to ______ kilopascal gauge (kPag) (______ to ______ pounds per square inch gauge (psig)). The total capacity at their setpoints is ______ percent of rated steam flow.

  To determine the ability of the overpressure protection system to prevent overpressurization, the applicant has analyzed the most severe anticipated overpressure transients. The analysis assumed that (1) the plant is in operation at design conditions of ____ percent of rated steam flow and a reactor vessel dome pressure of ____ kPag (__ psig) and (2) the reactor is shut down by ______. The calculated peak pressure at the bottom of the vessel is ______ kPag (______ psig), a value within the code allowable of ______ kPag (______ psig) (110 percent of vessel design pressure).

  These evaluation findings should be revised to [[

]].

The following Section VI, References, that are affected by the [[ ]] should be deleted:

- 6. 10 CFR 50.34(f), "Additional TMI-related Requirements."
- 14. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7511.1, "Spring-Loaded Valves."
- 18. NUREG-0737, "Clarification of TMI Action Plan Requirements."
- 26. NRC Letter to All Boiling Water Reactor Licensees, "NUREG-0737 Technical Specifications," Generic Letter 83-02, January 10, 1983.

In addition, any specific discussions in the above SRP sections that are only applicable to PWRs should be deleted.

4.3.3 Standard Review Plan 5.4.13

SRP 5.4.13, Isolation Condenser System (BWR), Rev. 0, states that the areas of review include the system design bases, design criteria, components, support systems, and instrumentation and controls employed in the system. The areas of review, review interfaces, acceptance criteria, review procedures, evaluation findings, and references are acceptable for use for the BWRX-300 based on the design description and design requirements discussed in Subsection 2.1.2 of this LTR. Therefore, the existing SRP provides adequate guidance to use 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.

4.3.4 Standard Review Plan 6.3

SRP 6.3, ECCS, Rev. 3, states that the areas of review include the following (Note: these are the areas of review applicable to the design stage of the nuclear power plant only):

1. The design bases for the ECCS are reviewed to assure that they satisfy applicable regulations, including the general design criteria and the requirements of 10 CFR 50.46 regarding ECCS acceptance criteria.
2. The design bases for the ADS are also reviewed for compliance with TMI Action Plan Items and associated guidance. This applies to BWRs and the advanced passive reactors (both Pressurized Water Reactors (PWRs) and BWRs).
3. For advanced passive reactors which rely on gravitational head to provide ECCS injection to the RCS, the RCS must be designed with an ADS such that the available gravitational head is sufficient to provide adequate core cooling when depressurized.
4. For advanced reactors which rely on passive safety-related systems and equipment to automatically establish and maintain safe-shutdown conditions for the plant, these passive safety systems must be designed with sufficient capability to maintain safe-shutdown conditions for 72 hours, without operator actions and without nonsafety-related onsite or offsite power.

5. The design of the ECCS is reviewed to determine that it is capable of performing all of the functions required by the design bases.

The design functions of the [[

]] to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(5). The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range. These BWRX-300 acceptance criteria ensure the following:

1. No significant fuel cladding heatup occurs in the short-term.
2. No significant fuel cladding oxidization occurs.
3. No significant fuel cladding hydrogen generation occurs.
4. No significant changes in core geometry occur.
5. No significant fuel cladding heatup occurs in the long-term.

Although this is a non-traditional approach for the design of the ECCS for past LWRs, no active or passive injection of additional water inventory is required following the worst-case LOCA to meet these BWRX-300 acceptance criteria and to meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 14, GDC 30, GDC 31, and GDC 35. Therefore, the second and third areas of review regarding ADS and passive ECCS injection are not applicable. Other than these areas of review, the review interfaces, acceptance criteria, review procedures, evaluation findings, and references are acceptable for use for the BWRX-300 recognizing that those discussions related to the ADS and active or passive ECCS injection are not applicable. Based on these factors, the existing SRP provides adequate guidance to use 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.

4.3.5 Standard Review Plan 15.6.5

SRP 15.6.5, LOCAs resulting from Spectrum of Postulated Piping Breaks Within the RCPB, Rev. 3, includes review for compliance with the requirements of 10 CFR 50 Appendix A, GDC 35, as well as 10 CFR 50.46 and 10 CFR 50 Appendix K, and the applicable general design requirements discussed in SRP Section 6.3.

The design functions of the [[

]] to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(5). [[

]] to meet the requirements of 10 CFR 50 Appendix A, GDC 35, 10 CFR 50.46, and 10 CFR 50 Appendix K, and the applicable general design requirements discussed in SRP Section 6.3. Based on these factors, the existing SRP provides adequate guidance to use 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.

4.4 Generic Issues

The following generic issues provided are based on their relevance to the scope of this LTR, and an up-to-date evaluation of generic issues is to be provided during future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant requesting a CP and OL under 10 CFR 50 or a COL under 10 CFR 52.

4.4.1 NUREG-0737

NUREG-0737, Clarification of TMI Action Plan Requirements, November 1980, contains requirements approved for implementation by the NRC Commissioners as a result of the accident at TMI Unit 2. The NRC Commission subsequently recommended that certain of these requirements be added to the 10 CFR 50 regulations, which were subsequently implemented in 10 CFR 50.34(f). Compliance with the items that are related to [[ ]] are discussed in Subsection 4.1.1 of this LTR.

4.5 Operational Experience and Generic Communications

The operational experience and generic communications provided are based on their relevance to the scope of this LTR, and an up-to-date evaluation of operational experience and generic communications is to be provided during future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant requesting a CP and OL under 10 CFR 50 or a COL under 10 CFR 52.

4.5.1 Generic Letter 83-02

Generic Letter 83-02, NUREG-0737 Technical Specifications, dated January 10, 1983, contains a request for information for the current BWR licensees regarding NUREG-0737 items for which technical specifications are required, including guidance on the scope of a specification which the staff would find acceptable and sample technical specifications. This includes NUREG-0737 item II.K.3.3 for reporting relief and safety valve failures. This requirement is not applicable because [[ ]]. In addition, this requirement was not subsequently implemented in 10 CFR 50.34(f). However, technical specifications for the [[ ]] are to be proposed during future licensing activities.

4.5.2 Generic Letter 95-07

Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, dated August 17, 1995, contains a request to ensure that safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding are evaluated to ensure they are capable of performing the safety functions described in the licensing basis. This guidance will be evaluated for applicability during future licensing activities.

5.0 REFERENCES

5.1 ASME Boiler and Pressure Vessel Code Section III Rules for Construction of Nuclear Facility Components, Division 1 – Subsection NB Class 1 Components

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 ASME B16.5-2017 "Pipe Flanges and Flanged Fittings NPS ½ Through NPS 24 Metric/Inch Standard," American Society of Mechanical Engineers, 2017

5.4 ASME B16.34-2017 "Valves-Flanged, Threaded, and Welding End," American Association of Mechanical Engineers, 2017

5.5 ASME QME-1-2007 "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," American Society of Mechanical Engineers, 2007

5.6 NEDC-33911P, "BWRX-300 Containment Performance"

5.7 26A66412AW, Rev 10, "ESBWR Design Control Document, Tier 2, Chapter 7 Instrumentation and Control Systems," GE Hitachi Nuclear Energy, April 2014

5.8 GEH Letter M200064, "Response to Requests for Additional Information (eRAIs) 9730, 9731 and 9732 for Licensing Topical Report NEDC-33910P, Revision 0, BWRX-300 Reactor Pressure Vessel (RPV) Isolation and Overpressure Protection," dated April 20, 2020

5.9 GEH Letter M200071, "Response to Requests for Additional Information (eRAIs) 9730, 9731 and 9732, Supplement 1, for Licensing Topical Report NEDC-33910P, Revision 0, BWRX-300 Reactor Pressure Vessel (RPV) Isolation and Overpressure Protection," dated May 4, 2020

5.10 Letter from Rani Franovich (NRC) to Michelle Catts (GEH), Subject: Final Safety Evaluation for GE-Hitachi Licensing Topical Report NEDC-33910P, Revision 0, "BWRX-300 Reactor Pressure Vessel (RPV) Isolation and Overpressure Protection," ADAMS Accession Number ML20176A450, November 18, 2020

Appendix A
GEH Responses to NRC RAIs on NEDC-33910P Revision 0

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eRAI No.: 9730
Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-1

Section 2.1.2 in NEDC-33910 describes the Isolation Condenser System (ICS) for the BWRX-300 nuclear power plant. This section indicates that the ICS includes [[ ]]. To support the NRC staff review of NEDC-33910P [sic] and its conformance to 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 35 and 37 for the IC condensate return valves, the NRC staff requests that GEH describe the following:

(a) Any first of a kind (FOAK) features,

(b) Valve and actuator types,

(c) Valve size,

(d) Qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100,

(e) Plans for valve and actuator diversity,

(f) Incorporation of lessons learned from international operating experience where ICS valves failed to open as designed,

(g) Accessibility for inservice testing (IST) activities in accordance with 10 CFR 50.55a,

(h) Design features to avoid thermal binding or pressure locking of the valves, and

(i) OM Code leakage classification.

If any of this information is not available at this time, the staff requests that GEH indicate its plans to provide this information during future licensing activities for the BWRX-300 nuclear power plant.

GEH Response to NRC Question 03.09.06-1

Although detailed design of the IC condensate return valves has not yet been completed, the design functions and features of the IC condensate return valves are anticipated to be like what is described in Section 5.4.6 of the Economically Simplified Boiling Water Reactor (ESBWR) Design Control Document (DCD). Compliance with the requirements of 10 CFR 50, Appendix A, General Design Criteria (GDC) 1, GDC 2, GDC 4, and GDC 37 for the IC condensate return valves is anticipated to be the same as described in Section 5.4.6 of the ESBWR DCD. However, NEDC-33910P is not requesting NRC approval for the IC condensate return valves to meet these GDC. Instead, it is requested that the limited design requirements specified for the ICS including the IC condensate return valves be found acceptable for ensuring that the ICS can perform the limited functions that are a subject of NEDC-33910P in demonstrating compliance with 10 CFR 50, Appendix A, GDC 35. These include the following design functions:

- ICS is initiated automatically on high reactor pressure vessel (RPV) pressure indicating an overpressure event or on signals indicating a loss-of-coolant accident (LOCA).
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]

NEDC-33910P did not describe compliance with 10 CFR 50 Appendix A, GDC 2, and will be revised to include this information. The conclusion of this additional information to be provided is that the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 2. Subsection 5.1.6 of NEDC-33911P, BWRX-300 Containment Performance, also describes that the RPV isolation valves will comply with 10 CFR 50 Appendix A, GDC 2.

As described in NEDC-33910P, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS in compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling. [[ ]] 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 35.

NEDC-33910P did not describe compliance with 10 CFR 50 Appendix A, GDC 37, and will be revised to include this information. The conclusion of this additional information to be provided is that the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 37.

The details of the design functions and features beyond those requirements described above and already described in NEDC-33910P are to be addressed 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. The following discussions address the specific questions provided:

(a) No FOAK features will be specified for the IC condensate return valves.

(b) Valve and actuator types will be addressed in the detailed design of the valves.

(c) Valve size will be addressed in the detailed design of the valves and will be specified during future licensing activities.

(d) Qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100, will be addressed in the detailed design of the valves and will be specified during future licensing activities.

(e) As described in NEDC-33910P, the [[

]] These requirements provide for sufficient description of valve and actuator diversity in support of the NEDC-33910P conclusions that [[

]].

(f) Incorporation of lessons learned from international operating experience where ICS valves failed to open as designed will be addressed in the detailed design of the valves and will be specified during future licensing activities. However, NEDC-33910P will be revised to include discussion for Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, as this operating experience may be applicable to the detailed design of the valves.

(g) Accessibility for IST activities in accordance with 10 CFR 50.55a will be addressed in the detailed design of the valves and will consider IST requirements like those described in ESBWR DCD, Tier 2, 26A6642AK, Rev. 10, April 2014, Table 3.9-8. The IC condensate return valves design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, 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. The specific IST requirements for the BWRX-300 design will be specified during future licensing activities.

(h) Design features to avoid thermal binding or pressure locking of the valves are not necessary for the IC condensate return valves. During normal operation with the ICS in standby, there are [[ ]].

(i) IST requirements will be like those described in ESBWR DCD, Tier 2, 26A6642AK, Rev. 10, April 2014, Table 3.9-8, which consists of position indication verification (ASME OM Code Paragraph ISTC-3700), stroke open testing (ASME OM Code Paragraph ISTC-3521), and fail open testing (ASME OM Code Paragraph ISTC-3560). Like the ESBWR, the IC condensate return valves are classified as ASME OM Code Category B valves, and do not require leakage testing. However, the specific IST requirements for the BWRX-300 design will be specified during future licensing activities.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to add the following as new Subsections 4.1.5 and 4.1.13 to address compliance with 10 CFR 50, Appendix A, GDC 2 and GDC 37, and to address addition of Generic Letter 95-07 as new Subsection 4.5.2:

. . .

4.1.5 10 CFR 50 Appendix A, GDC 2

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 2, Design bases for protection against natural phenomena, requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

  Statement of Compliance: The BWRX-300 RPV isolation and overpressure protection design features, [[ ]] are to be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. Specific design requirements for the [[ ]] used to verify the capability to perform their safety functions, and the natural phenomena and effects evaluated, will be provided during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 2.

. . .

4.1.13 10 CFR 50 Appendix A, GDC 37

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 37, Testing of emergency core cooling system, requires that the emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Statement of Compliance: As previously described, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS in compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling, which is assured for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS through the use of [[ ]].

Specific requirements for periodic pressure and functional testing of the [[ ]] to assure (1) the structural and leaktight integrity of these components, (2) the operability and performance of these active components, and (3) the operability of the systems as a whole containing the [[ ]] will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 37.

. . .

4.5.2 Generic Letter 95-07

Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, dated August 17, 1995, contains a request to ensure that safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding are evaluated to ensure they are capable of performing the safety functions described in the licensing basis. This guidance will be evaluated for applicability during future licensing activities.

eRAI No.: 9730
Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-2

Section 2.2 in NEDC-33910 provides a general overview of reactor pressure vessel (RPV) isolation concept, and Section 2.5 in NEDC-33910 specifies the RPV isolation valve design requirements for the BWRX-300 nuclear power plant. These sections indicate that there will be [[

]] To support the NRC staff review of NEC-33910 and its conformance to 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 54, 55, and 56 for the RPV isolation valves, the NRC staff requests that GEH describe the following:

(a) Any FOAK features,

(b) Valve types and sizes,

(c) Qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100,

(d) Plans for valve diversity,

(e) Accessibility for IST activities in accordance with 10 CFR 50.55a,

(f) Design to avoid thermal binding or pressure locking of the valves, and

(g) ASME OM Code leakage classification.

If any of this information is not available at this time, the staff requests that GEH indicate its plans to provide this information during future licensing activities for the BWRX-300 nuclear power plant.

GEH Response to NRC Question 03.09.06-2

The detailed design of the RPV isolation valve assemblies has not yet been completed. Compliance with the requirements of 10 CFR 50, Appendix A, General Design Criteria (GDC) 1, GDC 2 and GDC 4 for the RPV isolation valve assemblies is anticipated to be the same as for other ASME Class 1 valves described in the ESBWR DCD. Therefore, the BWRX-300 will comply with the requirements of GDC 1, GDC 2 and GDC 4 for the RPV isolation valve assemblies. NEDC-33910P is requesting that the limited design requirements specified for the RPV isolation valve assemblies be found acceptable for ensuring that the limited functions that are a subject of NEDC-33910P are met in demonstrating compliance with 10 CFR 50, Appendix A, GDC 33 and GDC 35.

Compliance with the requirements of 10 CFR 50, Appendix A, GDC 54, GDC 55, and GDC 56 as related to the design of the [[ ]] are described in Subsections 5.1.21, 5.1.22 and 5.1.26 of NEDC-33911P, Revision 0, BWRX-300 Containment Performance.

The details of the design functions and features beyond those requirements described above and already described in NEDC-33910P are to be addressed 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. The following discussions address the specific questions provided:

(a) Any FOAK features will be addressed in the detailed design of the valves and will be specified during future licensing activities.

(b) Valve types and sizes will be addressed in the detailed design of the valves and will be specified during future licensing activities.

(c) Qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100, will be addressed in the detailed design of the valves and will be specified during future licensing activities.

(d) As described in NEDC-33910P, the [[ ]] These requirements provide for sufficient description of diverse actuation in support of the NEDC-33910P conclusions that the [[ ]] to meet the criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(5). The [[ ]] for any breaks that would result in a loss of reactor coolant at a rate in excess of the capability of the nonsafety-related normal reactor coolant makeup systems, which for the BWRX-300 includes LOCA break sizes [[ ]]. The worst-case single failure affecting the [[ ]] does not prevent fulfillment of the required ECCS design functions. Therefore, there is no need for the [[ ]].

(e) Accessibility for IST activities in accordance with 10 CFR 50.55a will be addressed in the detailed design of the valves and will consider IST requirements like those described in ESBWR DCD, Tier 2, 26A6642AK, Rev. 10, April 2014, Table 3.9-8, for other active ASME Class 1 valves. The specific IST requirements for the BWRX-300 design will be specified during future licensing activities.

(f) Design features to avoid thermal binding or pressure locking of the valves are not necessary for the RPV isolation valves. The safety function of the RPV isolation valves is to close as opposed to having an opening safety function. Automatic actuation of [[ ]] performs the function to mitigate the effects of a LOCA. [[ ]] is a one-time action to open or close appropriate valves during accident response which may also be initiated manually as a one-time action if necessary.

(g) As described in NEDC-33910P all BWRX-300 RPV isolation valves shall have a proven low leakage potential. Design and administrative leakage limits are applied to valve selection during the BWRX-300 preliminary design and are based on plant design and event evaluations using offsite dose consequences compared to regulatory limits as well as containment design limits. The leakage criteria are analyzed as part of the plant safety analysis. Therefore, the RPV isolation valves are classified as ASME OM Code Category A valves requiring a seat leakage rate test (ASME OM Code Paragraph ISTC-3600). IST requirements will be like what is described in ESBWR DCD, Tier 2, 26A6642AK, Rev. 10, April 2014, Table 3.9-8, for ASME Class 1 valves. The specific IST requirements for the BWRX-300 design will be specified during future licensing activities.

Proposed Changes to NEDC-33910P, Revision 0

None

eRAI No.: 9730
Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-3

Section 2.6 in NEDC-33910 specifies the RPV isolation valve actuator design requirements for the BWRX-300 nuclear power plant. This section specifies that the valve and actuator designs will be qualified with ASME QME-1. This section refers to several aspects for consideration or to be considered as part of the RPV isolation valve actuator design requirements. To support the NRC staff review of NEDC-33910 [sic] and its conformance to 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 54, 55, and 56 for the RPV isolation valve actuator design requirements, the NRC staff requests that GEH describe the following:

(a) Any FOAK features,

(b) Actuator types,

(c) Specific QME-1 edition for qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100,

(d) Plans for actuator diversity,

(e) Accessibility for IST activities in accordance with 10 CFR 50.55a, and

(f) Intent of terms such as "consideration" and "considered" in this section and elsewhere in NEDC-33910.

If any of this information is not available at this time, the staff requests that GEH indicate its plans to provide this information during future licensing activities for the BWRX-300 nuclear power plant.

GEH Response to NRC Question 03.09.06-3

The detailed design of the RPV isolation valve actuators has not yet been completed. Compliance with the requirements of 10 CFR 50, Appendix A, General Design Criteria (GDC) 1, GDC 2 and GDC 4 for the RPV isolation valve actuators is anticipated to be the same as actuators for other ASME Class 1 valves described in the ESBWR DCD. Therefore, the BWRX-300 will comply with the requirements of GDC 1, GDC 2 and GDC 4 for the RPV isolation valve actuators. NEDC-33910P is requesting that the limited design requirements specified RPV isolation valve actuators be found acceptable for ensuring that the limited functions that are a subject of NEDC-33910P are met in demonstrating compliance with 10 CFR 50, Appendix A, GDC 33 and GDC 35.

Compliance with the requirements of 10 CFR 50, Appendix A, GDC 54, GDC 55, and GDC 56 as related to the design of the [[ ]] are described in Subsections 5.1.21, 5.1.22 and 5.1.26 of NEDC-33911P, Revision 0, BWRX-300 Containment Performance.

The details of the design functions and features beyond those requirements described above and already described in NEDC-33910P are to be addressed 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. The following discussions address the specific questions provided:

(a) Any FOAK features will be addressed in the detailed design of the valve actuators and will be specified during future licensing activities.

(b) Valve actuator types will be addressed in the detailed design of the valve actuators.

(c) Qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100, will be addressed in the detailed design of the valve actuators and will be specified during future licensing activities.

(d) As described in NEDC-33910P, the [[ ]] The method to establish this diversity requirement will be addressed in the detailed design of the valve actuators and will be specified during future licensing activities.

(e) IST activities in accordance with 10 CFR 50.55a are specified for the RPV isolation valves as discussed in the response to NRC Question 03.09.06-2. IST requirements will be like those described in ESBWR DCD, Tier 2, 26A6642AK, Rev. 10, April 2014, Table 3.9-8, for ASME Class 1 valves. The specific IST requirements for the BWRX-300 design will be specified during future licensing activities.

(f) GEH understands that the use of terms such as “consideration” and “considered” is not appropriate where used. Therefore, the discussion containing those terms are proposed to be revised in NEDC-33910P as described below.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to replace the use of terms such as “consideration” and “considered” with appropriate terms:

. . .

2.4 Reactor Pressure Vessel Nozzle Design Requirements

. . .

[[

]]

2.4.1 Connection of Reactor Pressure Vessel Isolation Valves to the Reactor Vessel

The BWRX-300 design requirements for identifying postulated pipe rupture locations and configurations inside containment ~~consider~~conform to the guidance in Branch Technical Position (BTP) 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment.”

. . .

[[ ]] the design criteria given in ASME Code Section III, Paragraph NB-3230, provides greater margin against yielding due to service loads than do the rules of Paragraph NB-3653 for typical piping system materials, even when ~~considering~~using the more restrictive limits of BTP 3-4, Part B, Item 1(ii)(1).

. . .

2.5 Reactor Pressure Vessel Isolation Valve Design Requirements

. . .

Thus, motor operated valves other than fail closed magnetic motor valves (i.e., solenoid operated valves) are not ~~considered~~used for any RPV isolation valve applications.

. . .

A critical aspect of the valve and actuator selection to ~~consider~~evaluate is the failure mode. The failure mode of the RPV isolation valves are determined based on the safety function of the connected system.

. . .

2.6 Reactor Pressure Vessel Isolation Valve Actuator Design Requirements

. . .

A key design ~~consideration~~requirement is control of the temperature at the valve actuator interface in order to limit thermal effects on the actuator assembly. The RPV isolation valves are heated by process water or steam, which also elevates valve actuator temperatures above the local ambient. Valve and valve actuator designs are qualified in accordance with ASME QME-1 [Reference 5.5] to include ~~consideration~~evaluation of the local environmental conditions, including ~~consideration~~evaluation of the effects of heat transfer from the process water or steam and Design Basis Events. [[ ]] The stem connection and actuator mounting method are studied to determine if thermal isolation needs to be implemented. High temperature seals or lubricants are ~~considered~~used for the actuators.

. . .

2.7 Categories of Pipe Breaks

. . .

Steam and liquid line breaks are ~~considered~~evaluated. The pipe breaks ~~considered~~evaluated in the safety analysis are divided into two size categories:

. . .

~~4.1.5~~4.1.6 10 CFR 50 Appendix A, GDC 4

. . .

In addition, the dynamic effects of postulated pipe breaks are to be ~~considered~~evaluated in the BWRX-300 design.

. . .

[[ ]] the BWRX-300 design requirements include ~~consideration~~evaluation of the acceptable criteria to identify postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," as discussed in Subsection 2.4.1 of this LTR.

. . .

~~4.3.1~~4.3.2 Standard Review Plan 5.2.2

. . .

It is recommended that this area of review should ~~consider~~include [[ ]].

eRAI No.: 9730
Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-4

Section 4.1.1 in NEDC-33910 discusses compliance with specific requirements in 10 CFR 50.34(f) related to the TMI-2 accident lessons learned. In describing compliance with 10 CFR 50.34(f)(1)(x) with respect to nonsafety-related equipment and accounting for leakage, NEDC-33910 indicates that the [[ ]] are one-time actuation systems. The NRC staff requests that GEH clarify this section, or describe the [[

]].

GEH Response to NRC Question 03.09.06-4

As stated in NEDC-33910P, the RPV isolation valves for main steam line, feedwater, shutdown cooling, and reactor water cleanup shall fail in the closed position on loss of signal or control power. [[

]] In addition to their failure positions upon a loss of signal or control power, these valves are also placed in their required post-accident valve positions using a single automatic actuation signal.

Although the detailed design of the valves and valve actuators has not been completed, positive mechanical means shall be required in the design of the valve actuators to ensure that upon automatic actuation or a loss of signal or control power to both place and then maintain the valves in the required post-accident valve positions are to be used. These requirements are to be implemented during detailed design of the valves and actuators. Therefore, additional information requiring the use of positive mechanical means in the design of the valve actuators to maintain these valves in their required post-accident valve positions is proposed to be added to NEDC-33910P.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to add the design requirements for the use of positive mechanical means in the design of the valve actuators to maintain these valves in their required post-accident valve positions:

. . .

2.5 Reactor Pressure Vessel Isolation Valve Design Requirements

. . .

Design Requirements:

- The RPV isolation valves for main steam line, feedwater, shutdown cooling, and reactor water cleanup shall fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means.
- [[ ]] with valve actuators designed to maintain the valves in their as-is position by positive mechanical means.

. . .

3.1.2 Isolation Condenser System Design Requirements

. . .

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. [[ ]] with valve actuators designed to maintain the valves in their open position by positive mechanical means.

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eRAI No.: 9730
Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-5

Section 4.1.1 in NEDC-33910 discusses compliance with specific requirements in 10 CFR 50.34(f). In describing compliance with 10 CFR 50.34(f)(2)(x) with respect to anticipated transient without scram (ATWS), NEDC-33910 indicates that based on [[ ]] qualification testing, the testing associated with 10 CFR 50.34(f)(2)(x) is not required. The NRC staff requests that GEH clarify this section, or explain why [[ ]] qualification testing is not subject to 10 CFR 50.34(f)(2)(x).

GEH Response to NRC Question 03.09.06-5

10 CFR 50.34(f)(2)(x) requires that a test program and associated model development be provided, and tests conducted, to qualify RCS relief and safety valves for BWRs. As stated in NEDC-33910P, [[ ]]. Therefore, this requirement is not technically relevant to the BWRX-300 design. However, the [[ ]] perform a similar function during an ATWS event in providing overpressure protection and will be qualified by testing to perform their safety-related design function for fluid conditions expected under operating conditions, transients and accidents including ATWS events.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to describe this requirement as not technically relevant rather than not required, but to also indicate that the [[ ]] will be qualified by testing to perform their safety-related design function for fluid conditions expected under operating conditions, transients and accidents including ATWS events:

. . .

4.1.1 10 CFR 50.34(f)

. . .

- Regulatory Requirement: 10 CFR 50.34(f)(2)(x) requires that a test program and associated model development be provided and tests conducted to qualify RCS relief and safety valves and, for PWRs, Power-Operated Relief Valve (PORV) block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of ATWS conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. (II.D.1)

  Statement of Compliance: The design features of the BWRX-300 RCPB include the use of the RPS [[ ]] for overpressure protection, [[

]]. ~~Qualification tests of the [[ ]] are performed to verify proper operation under all fluid conditions expected under operating conditions, transients and accidents, and operation of the [[ ]] under ATWS conditions is evaluated. Based on the required [[ ]] qualification tests and evaluation, and because~~Because the [[ ]], this ~~test program and associated model development, including conduct of test, is not required.~~requirement is not technically relevant to the BWRX-300. However, the [[ ]] perform a similar function during an ATWS event in providing overpressure protection and will be qualified by testing to perform their safety-related design function for fluid conditions expected under operating conditions, transients and accidents including ATWS events.

eRAI No.: 9730
Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-6

Section 4.1.3 in NEDC-33910 indicates that the requirements of 10 CFR 50.55a will be satisfied. This section specifically references the RPV isolation valves. This section also states that no alternative approach, exception, or exemption from these requirements is required. The NRC staff requests that GEH describe the compliance with the requirements in 10 CFR 50.55a for the [[ ]]. The staff also requests that GEH clarify the intent of the statement in this section and elsewhere in NEDC-33910 that no alternative approach, exception, or exemption from these requirements is required.

GEH Response to NRC Question 03.09.06-6

GEH understands that the use of statements like "Full compliance with these requirements is to be demonstrated in the preliminary and final design of the BWRX-300 to be completed during future licensing activities. Therefore, no alternative approach, exception, or exemption from these requirements is required" are not appropriate where used. Therefore, the discussion containing those statements, including in Section 4.1.3, are proposed to be revised in NEDC-33910P as described below for clarification of the intent, which is to provide a commitment to meeting the applicable regulatory requirements during detailed design.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to replace the use of these statements with appropriate statements of compliance to provide a commitment to meeting the applicable regulatory requirements during detailed design:

. . .

2.7 Categories of Pipe Breaks

. . .

The emergency core cooling system (ECCS) evaluation model for the BWRX-300 is to be developed using previously approved methodologies used in the ECCS performance analyses for the ESBWR as modified using BWRX-300 specific design requirements and parameters. ~~Full compliance with these requirements is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities using an NRC-approved BWRX-300 ECCS evaluation model. Methodology for containment response is described in ~~a separate~~ LTR NEDC-33911P, BWRX-300 Containment Performance [Reference 5.6].

. . .

4.1.2 10 CFR 50.46

. . .

Based on the above evaluation, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in ~~full~~ compliance with the criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS ~~in full compliance~~consistent with the definition of a LOCA in 10 CFR 50.46(c)(1).

Although this is a~~n alternative and~~ non-traditional approach for the design of the ECCS for past LWRs, no [[ ]] is ~~required~~needed following the worst case LOCA to meet the acceptance criteria in 10 CFR 50.46(b). Final ECCS performance analyses are to be completed during future licensing activities. Therefore, ~~no exception or exemption is required for these regulatory requirements. Full compliance with these requirements is to be demonstrated in the final ECCS performance analyses to be completed during future licensing activities.~~the BWRX-300 design will meet the requirements of 10 CFR 50.46.

. . .

Statement of Compliance: The ECCS evaluation model for the BWRX-300 is to be developed using previously approved methodologies used in the ECCS performance analyses for the ESBWR as modified using BWRX-300 specific design requirements and parameters. ~~Full compliance with these requirements is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities using an NRC-approved BWRX-300 ECCS evaluation model. Therefore, ~~no alternative approach, exception, or exemption from these requirements is required~~the BWRX-300 design will meet the requirements of 10 CFR 50.46(a)(1).

. . .

Statement of Compliance: The design goal is that the core does not uncover during LOCAs, and that no significant fuel cladding heatup occurs, such that the calculated maximum fuel element cladding temperature does not exceed the acceptance criterion of 2200°F. ~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(1).

. . .

Statement of Compliance: The design goal is that the core does not uncover during LOCAs, and that no significant fuel cladding oxidization occurs, such that the calculated total oxidation of the cladding does not nowhere exceed the acceptance criterion of 0.17 times the total cladding thickness before oxidation. ~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(2).

. . .

Statement of Compliance: The design goal is that the core does not uncover during LOCAs, and that no significant fuel cladding hydrogen generation occurs, such that the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed the acceptance criterion of 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. ~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(3).

. . .

Statement of Compliance: The design goal is that ~~the core does not uncover during LOCAs, and that~~ no significant changes in core geometry occur, such that the acceptance criterion of the core remaining amenable to cooling is met. ~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(4).

. . .

Statement of Compliance: The design goal is that the core does not uncover during LOCAs, and that ~~no significant fuel cladding heatup~~long term cooling to remove decay heat and maintain the core temperature to acceptably low values occurs, such that after any calculated successful initial operation of the ECCS, the acceptance criteria of the calculated core temperature being maintained at an acceptably low value and decay heat being removed for the extended period of time required by the long lived radioactivity remaining in the core are met.

For the BWRX-300, [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed power without any further need of onsite or offsite electric power system operation.

The selected long-term cooling timeframe of [[ ]] is sufficient to maintain reactor water level at TAF for both [[

]] Following the worst-case postulated LOCA (i.e., [[
]]), 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.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(5).

. . .

4.1.3 10 CFR 50.55a

. . .

These requirements are to be implemented during detailed design of the safety-related components of the [[ ]].~~Full compliance with these requirements is to be demonstrated in the preliminary and final design of the BWRX-300 to be completed during future licensing activities.~~

Therefore, ~~no alternative approach, exception, or exemption from these requirements is required~~the BWRX-300 design will meet the requirements of 10 CFR 50.55a.

. . .

4.1.4 10 CFR 50 Appendix A, GDC 1

. . .

~~Full compliance with these requirements is to be demonstrated in the preliminary and final design of the BWRX-300 to be completed during future licensing activities.~~

Therefore, ~~no alternative approach, exception, or exemption from these requirements is required~~the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 1.

. . .

~~4.1.5~~4.1.6 10 CFR 50 Appendix A, GDC 4

. . .

Therefore, ~~no alternative approach, exception, or exemption from these requirements is required~~the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 4.

. . .

~~4.1.6~~4.1.7 10 CFR 50 Appendix A, GDC 14

. . .

This results in an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture, in ~~full~~ compliance with this criterion. For piping that have [[

]]. Further design details are to be described during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 14.

. . .

~~4.1.7~~4.1.8 10 CFR 50 Appendix A, GDC 15

. . .

The combination of RPS and [[ ]] design features ensure that the acceptance criteria for each component in the protected system are met including 1) the overpressure does not exceed 1.1 times design pressure for the expected system pressure transient condition, and 2) the calculated stress intensity and other design limitations for Service Level C are not exceeded for the unexpected system excess pressure transient condition, with further design details to be described during future licensing activities. ~~Full compliance with these requirements is to be demonstrated in the preliminary and final design of the BWRX-300 to be completed during future licensing activities.~~

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 15.

. . .

~~4.1.8~~4.1.9 10 CFR 50 Appendix A, GDC 30

. . .

In addition, means are to be provided to detect and identify the location of the source of reactor coolant leakage, including the components of the ICS and RPV isolation valves, for components of the RCPB, with further design details to be described during future licensing activities. ~~Full compliance with these requirements is to be demonstrated in the preliminary and final design of the BWRX-300 to be completed during future licensing activities.~~

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 30.

. . .

~~4.1.9~~4.1.10 10 CFR 50 Appendix A, GDC 31

. . .

The components of the RCPB, including the ICS and RPV isolation valves, are to be designed with sufficient margin to assure that these requirements are met, with further design details to be described during future licensing activities. ~~Full compliance with these requirements is to be demonstrated in the preliminary and final design of the BWRX-300 to be completed during future licensing activities.~~

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 31.

. . .

~~4.1.11~~4.1.12 10 CFR 50 Appendix A, GDC 35

. . .

As previously described, the combined design features of the [[ ]] meet the as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in ~~full~~ compliance with the criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS in ~~full~~ compliance with the definition of a LOCA in 10 CFR 50.46(c)(1).

. . .

~~Full compliance with these requirements is to be demonstrated in the final ECCS performance~~The analyses to ~~be completed~~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 35.

. . .

4.2.1 Regulatory Guide 1.26

. . .

Therefore, the BWRX-300 design conforms to the guidance for the RPV isolation valves and the ICS, including regulatory positions of RG 1.26~~, without requiring an alternative approach or exception~~.

. . .

4.2.2 Regulatory Guide 1.29

. . .

Therefore, the BWRX-300 design conforms to the guidance, including regulatory positions of RG 1.29~~, without requiring an alternative approach or exception~~.

. . .

4.2.3 Regulatory Guide 1.45

. . .

Additionally, the means for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage in compliance with the requirements of 10 CFR 50 Appendix A, GDC 30, and the requirements for in-service inspection and testing of the [[ ]] in compliance with the requirements of 10 CFR 50.55a, are to be demonstrated during future licensing activities~~, and no alternative approach, exception, or exemption from these requirements is required~~.

Therefore, the BWRX-300 design conforms to the guidance, including regulatory positions of RG 1.45~~, without requiring an alternative approach or exception~~.

. . .

4.2.4 Regulatory Guide 1.84

. . .

Compliance with the requirements of 10 CFR 50.55a, including the use of ASME B&PV Section III Code Cases endorsed in RG 1.84 where necessary, is to be demonstrated during future licensing activities~~, and no alternative approach, exception, or exemption from these requirements is required~~.

Therefore, the BWRX-300 design conforms to the guidance, including regulatory positions of RG 1.84~~, without requiring an alternative approach or exception~~.

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Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-7

Section 4.1.6 in NEDC-33910 describes compliance with 10 CFR Part 50, Appendix A, GDC 14 regarding the reactor coolant pressure boundary. This section does not indicate if full compliance with GDC 14 is planned. The NRC staff requests that GEH clarify its compliance with GDC 14.

GEH Response to NRC Question 03.09.06-7

GEH understands that the use of statements such as "full compliance will be demonstrated during future licensing activities" is not appropriate where used. Therefore, the discussion containing those statements are proposed to be revised in NEDC-33910P as described in the response to NRC Question 03.09.06-6. These changes include stating that further design details are to be described in future licensing activities to demonstrate that the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 14.

Proposed Changes to NEDC-33910P, Revision 0

See the changes proposed in response to NRC Question 03.09.06-6.

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Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-8

Section 4.1.10 in NEDC-33910 describes compliance with the intent of GDC 33 regarding reactor coolant makeup. The NRC staff requests that GEH clarify its statement that the intent of this criterion is met.

GEH Response to NRC Question 03.09.06-8

As stated in NEDC-33910P, 10 CFR 50, Appendix A, GDC 33, Reactor coolant makeup, requires that a system to supply reactor coolant makeup for protection against small breaks in the RCPB shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

For the BWRX-300, [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed power without any further need of onsite or offsite electric power system operation. Further discussion regarding the selected long-term cooling period for the BWRX-300 is included in the GEH response to NRC Question NONE-2 in Enclosures 3 (Proprietary) and 6 (Non-Proprietary).

Because specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary [[ ]] 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 33. 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, a new Principal Design Criterion (PDC) 33 is proposed as described in the proposed changes to NEDC-33910P provided below.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to address proposal of a PDC 33 and provide justification for an exemption to these specific requirements of 10 CFR 50 Appendix A, GDC 33, which may be used as the bases for the necessary exemption during 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:

. . .

~~4.1.10~~4.1.11 10 CFR 50 Appendix A, GDC 33

. . .

Statement of Compliance: The safety analysis assumes that the small pipe breaks [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed ~~DC~~ power without any further need of onsite or offsite electric power system operation.

~~Although the BWRX-300 [[ ]] nonsafety-related injection systems may be used for manual addition of reactor coolant inventory by the operator using high pressure CRD injection or by reestablishing feedwater injection. These nonsafety-related injection systems can be used at any time following a LOCA. The timing of such operator actions is to be determined during the final ECCS performance analyses to be completed during future licensing activities.~~ Therefore, specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary [[

]]. 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 33. 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 33 is proposed:

> PDC 33, Maintaining ~~R~~reactor coolant ~~makeup~~inventory, a safety-related system to ~~supply~~maintain reactor coolant ~~makeup~~inventory and provide decay heat removal for protection against small breaks in the ~~RCPB~~reactor coolant pressure boundary ~~is not needed~~shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the ~~RCPB~~reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. ~~In addition, the operation of the systems to mitigate the consequences of small breaks in the RCPB do not require offsite electric power system operation, and only require one-time automatic actuation using onsite Class 1E battery-backed power without any further need of onsite or offsite electric power system operation.~~The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished.

These statements of compliance and proposed PDC 33 may be used as the bases for the necessary exemption during 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.

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Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-9

Section 4.1 in NEDC-33910, discussing 10 CFR Part 50 regulations and GDCs, does not mention GDC 2, as it relates to pumps, valves, and dynamic restraints important to safety to withstand the effects of natural phenomena combined with the effects of normal and accident conditions. The NRC staff requests that GEH clarify how GDC 2 will be met.

GEH Response to NRC Question 03.09.06-9

NEDC-33910P did not describe compliance with 10 CFR 50 Appendix A, GDC 2, and will be revised to include this information. The conclusion of this additional information to be provided is that the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 2.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to add the following as new Subsection 4.1.5 to address compliance with 10 CFR 50, Appendix A, GDC 2:

. . .

4.1.5 10 CFR 50 Appendix A, GDC 2

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 2, Design bases for protection against natural phenomena, requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

  Statement of Compliance: The BWRX-300 RPV isolation and overpressure protection design features, [[ ]] are to be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. Specific design requirements for the [[ ]] used to verify the capability to perform their safety functions, and the natural phenomena and effects evaluated, will be provided during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 2.

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NRC Question 03.09.06-10

Section 4.1 in NEDC-33910, discussing 10 CFR Part 50 regulations and GDCs, does not mention GDC 54, as it relates to designing piping systems penetration containment with the capability to test periodically the operability of the isolation valves and determine valve leakage acceptability. The NRC staff requests that GEH clarify how GDC 54 will be met.

GEH Response to NRC Question 03.09.06-10

Compliance with the requirements of 10 CFR 50, Appendix A, GDC 54, GDC 55, and GDC 56 as related to the design of the [[ ]] are described in Subsections 5.1.21, 5.1.22 and 5.1.26 of NEDC-33911P, Revision 0, BWRX-300 Containment Performance.

Proposed Changes to NEDC-33910P, Revision 0

None

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Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-11

Section 4.2 in NEDC-33910, discussing NRC Regulatory Guides (RGs), does not discuss RG 1.147 or RG 1.192, as they relate to the acceptability of Code Cases for the ASME Boiler and Pressure Vessel Code (BPV Code) and ASME Operation and Maintenance of Nuclear Power Plants (OM Code) for inservice inspection and IST activities in satisfying 10 CFR 50.55a, respectively. The NRC staff requests GEH to clarify the intent of the topical report regarding these RGs.

GEH Response to NRC Question 03.09.06-11

NEDC-33910P did not describe conformance to the regulatory guidance of RG 1.147 and RG 1.192 and will be revised to include this information. However, the requirements of ASME B&PV Code Section XI, Division 1, specifically apply during construction and operations activities of nuclear power plants for performance of inservice inspection activities, and the requirements of ASME OM Code specifically apply during operations and maintenance activities of nuclear power plants for performance of inservice testing activities, and do not apply during the design of the BWRX-300. Therefore, the conclusion of this additional information to be provided is that the guidance of RG 1.147 and RG 1.192 do not apply to the BWRX-300 design phase in meeting the requirements of 10 CFR 50.55a.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to add the following as new Subsections 4.2.5 and 4.2.6 to address conformance with the regulatory guidance of RG 1.147 and RG 1.192, respectively:

. . .

4.2.5 Regulatory Guide 1.147

RG 1.147, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, Rev. 19, lists the ASME B&PV Section XI Code Cases that the NRC has approved for use as voluntary alternatives to the mandatory ASME B&PV Code provisions that are incorporated by reference into 10 CFR 50. This applies to reactor licensees subject to 10 CFR 50.55a, Codes and Standards. These ASME B&PV Section XI Code Cases are acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50 Appendix A, GDC 1, Quality standards and records, and 10 CFR 50 Appendix A, GDC 30, Quality of reactor coolant pressure boundary, and the requirements of 10 CFR 52.79(a)(11) which requires the final safety analysis report to include "a description of the program(s), and their implementation, necessary to ensure that the systems and components meet the requirements of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants in accordance with

50.55a of this chapter." In 10 CFR 50.55a(a)(1)(ii), the NRC references the latest editions and addenda of ASME B&PV Code Section XI that the agency has approved for use.

Section 4.1.3 describes how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a. However, the requirements of ASME B&PV Code Section XI, Division 1, specifically apply during construction and operation activities of nuclear power plants for performance of inservice inspection activities, and do not apply during the design of the BWRX-300. Therefore, compliance with the requirements of 10 CFR 50.55a for the conduct of ISI activities, including the use of ASME B&PV Section XI Code Cases endorsed in RG 1.147 where necessary, is to be demonstrated during future licensing activities.

Based on this discussion, the guidance of RG 1.147 does not apply to the BWRX-300 design phase in meeting the requirements of 10 CFR 50.55a.

4.2.6 Regulatory Guide 1.192

RG 1.192, Operation and Maintenance Code Case Acceptability, ASME OM Code, Rev. 3, lists Code Cases associated with the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that the NRC has approved for use as voluntary alternatives to the mandatory ASME OM Code provisions that are incorporated by reference into 10 CFR 50. This applies to reactor licensees subject to 10 CFR 50.55a, Codes and Standards. These ASME OM Code Cases are acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50 Appendix A, GDC 1, Quality standards and records, and 10 CFR 50 Appendix A, GDC 30, Quality of reactor coolant pressure boundary, the requirements of 10 CFR 50.55a(f) which requires, in part, that Class 1, 2, and 3 components and their supports meet the requirements of the ASME OM Code or equivalent quality standards, and the requirements of 10 CFR 52.79(a)(11) which requires the final safety analysis report to include "a description of the program(s), and their implementation, necessary to ensure that the systems and components meet the requirements of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants in accordance with 50.55a of this chapter." In 10 CFR 50.55a(a)(1)(iv), the NRC references the latest editions and addenda of ASME OM Code that the agency has approved for use.

Section 4.1.3 describes how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a. However, the requirements of ASME OM Code specifically apply during operation and maintenance activities of nuclear power plants for performance of IST activities, and do not apply during the design of the BWRX-300. Therefore, compliance with the requirements of 10 CFR 50.55a for the conduct of IST activities, including the use of ASME OM Code Cases endorsed in RG 1.192 where necessary, is to be demonstrated during future licensing activities.

Based on this discussion, the guidance of RG 1.192 does not apply to the BWRX-300 design phase in meeting the requirements of 10 CFR 50.55a.

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NRC Question 03.09.06-12

Section 4.3 in NEDC-33910, discussing NUREG-0800 Standard Review Plan (SRP), does not mention SRP Section 3.9.6, as it relates to the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints. The NRC staff requests GEH to clarify the intent of the topical report regarding SRP Section 3.9.6.

GEH Response to NRC Question 03.09.06-12

NEDC-33910P did not describe conformance to the regulatory guidance of SRP 3.9.6 and will be revised to include this information. The BWRX-300 RPV isolation valves and overpressure protection design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, including any application for a construction permit under 10 CFR 50 or design certification application under 10 CFR 52. These requirements are to be implemented during detailed design of the safety-related components of the [[ ]]. Therefore, the conclusion of this additional information to be provided is that the existing SRP provides adequate guidance to use 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.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to add the following as new Subsection 4.3.1 to address conformance with the regulatory guidance of SRP 3.9.6:

. . .

4.3.1 Standard Review Plan 3.9.6

Standard Review Plan (SRP) 3.9.6, Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints, Rev. 4, states that the areas of review include the functional design and qualification provisions and IST programs for safety-related pumps, valves, and dynamic restraints (snubbers) designated as Class 1, 2, or 3 under ASME B&PV Code Section III.

As described in Section 4.1, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 2, GDC 4, GDC 14, GDC 15, and GDC 37, with specific requirements for the [[ ]] to be provided during future licensing activities. In addition, Section 4.1.3 describes that the BWRX-300 RPV isolation valves and overpressure protection design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, including any application for a construction permit under 10 CFR 50 or design certification application under 10 CFR 52.the requirements of 10 CFR 50.55a, during detailed design activities. These requirements are to be implemented during detailed design of the safety-related components of the [[

]]. Therefore, the existing SRP provides adequate guidance to use 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.

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NRC Question 03.09.06-13

Sections 4.4 and 4.5 in NEDC-33910 refer to generic issues, and operational experience and generic communications, respectively, applicable to the BWRX-300 nuclear power plant. This section only discusses two items with respect to these topics. The NRC staff requests that GEH indicate whether it will provide an up-to-date evaluation of generic issues, and operational experience and generic communications, during future licensing activities under 10 CFR Part 50 or Part 52.

GEH Response to NRC Question 03.09.06-13

Sections 4.4 and 4.5 of NEDC-33910P 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 and are provided based on their relevance to the scope of this LTR. Therefore, NEDC-33910P will be revised to include a discussion in each section identifying the limited scope of this evaluation, and committing to the up-to-date evaluation of these issues to be provided during future licensing activities in support of a BWRX-300 10 CFR 52 design certification application or for future 10 CFR 50 license applications.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to include the following changes to Sections 4.4 and 4.5 identifying the limited scope of this evaluation, and committing to the up-to-date evaluation of these issues to be provided during future licensing activities:

. . .

4.4 Generic Issues

<u>The following generic issues provided are based on their relevance to the scope of this LTR, and an up-to-date evaluation of generic issues is to be provided during 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.</u>

. . .

4.5 Operational Experience and Generic Communications

<u>The operational experience and generic communications provided are based on their relevance to the scope of this LTR, and an up-to-date evaluation of operational experience and generic communications is to be provided during 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.</u>

eRAI No.: 9730
Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-14

In referring to specific NRC regulations, NEDC-33910 indicates in several instances that full compliance will be demonstrated during future licensing activities. The NRC staff requests that GEH clarify the intent of this statement and indicate its plans to fully comply with specific NRC regulations.

GEH Response to NRC Question 03.09.06-14

GEH understands that the use of statements such as "full compliance will be demonstrated during future licensing activities" is not appropriate where used. Therefore, the discussion containing those statements are proposed to be revised in NEDC-33910P as described in the response to NRC Question 03.09.06-6.

Proposed Changes to NEDC-33910P, Revision 0

See the changes proposed in response to NRC Question 03.09.06-6.

eRAI No.: 9731
Date of eRAI Issue: 04/09/20

NRC Question 03.06.02-1

Section 2.2 in LTR NEDC-33910 states that for piping that have [[

]]. The NRC staff requests that GEH describe the design of the [[
]].

GEH Response to NRC Question 03.06.02-1

The design of the [[
]] Further details regarding the [[
]] are provided in Subsections 2.2.2, 2.2.7.1 and 5.1.22 of NEDC-33911P, BWRX-300 Containment Performance. As stated in NEDC-33911P, [[

]] extending to the containment wall, the BWRX-300 design requirements include identifying postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, Part B, Item 1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2).

Proposed Changes to NEDC-33910P, Revision 0

None

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eRAI No.: 9731
Date of eRAI Issue: 04/09/20

NRC Question 03.06.02-2

Section 4.1.5 of the LTR states that for piping connected to [[ ]], the BWRX-300 design requirements include consideration of the acceptance criteria to identify postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment." To support the NRC staff review of NEDC-33910P [sic] and its conformance to 10 CFR Part 50, Appendix A, GDC 4 related to protection against postulated breaks, the NRC staff requests [sic] the following:

(a) That GEH clarify whether a terminal end break would be postulated at the piping connection to the [[ ]].

(b) If a piping terminal break is postulated, that GEH describe how the BWRX-300 design requirements will provide assurance that the functionality of the [[ ]] will not be affected by the dynamic effects resulting from the postulated terminal end break.

(c) If breaks are not postulated at those piping connection terminal ends, that GEH describe how the BWRX-300 design provides reasonable assurance that the probability of postulated failures at those connection terminal ends is extremely low.

GEH Response to NRC Question 03.06.02-2

As stated in Subsections 2.2.2, 2.2.7.1 and 5.1.22 of NEDC-33911P, BWRX-300 Containment Performance, [[

]] extending to the containment wall, the BWRX-300 design requirements include identifying postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, Part B, Item 1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2). Therefore, the following responses to these specific questions are provided:

(a) A terminal end break would be postulated at the piping connection to the [[ ]] as specified in BTP 3-4, Part B, Item 1(iii)(2).

(b) As described in Subsection 5.1.7 of NEDC-33911P, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 4. This will be addressed during detailed design of the RPV isolation valves by designing the valves to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents, and the valves will be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

(c) As described in Subsection 2.2.2 of NEDC-33911P, a terminal end break would be postulated at the piping connection to the [[ ]] as specified in BTP 3-4, Part B, Item 1(iii)(2).

Proposed Changes to NEDC-33910P, Revision 0

None

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eRAI No.: 9731
Date of eRAI Issue: 04/09/20

NRC Question 03.06.02-3

Section 2.4.1 of the LTR states that the bolted flange connections of the [[ ]]. GEH also states that the design provisions, the leakage detection systems design, and the inservice inspection requirements demonstrate that the probability of gross rupture at these [[ ]] is extremely low. To support the NRC staff review of NEDC-33910P [sic] and its conformance to 10 CFR Part 50, Appendix A, GDC 4 related to protection against postulated breaks, the NRC staff requests that GEH indicate the material of construction for these [[ ]] and discuss why the potential failure mechanisms, such as vibration, water hammer, corrosion, and unexpected modes of operation, are not applicable for these [[ ]].

GEH Response to NRC Question 03.06.02-3

The material of construction for the [[ ]] will be determined during detailed design and addressed in future licensing activities. Therefore, the discussions addressing these [[ ]] are proposed to be revised in NEDC-33910P as described below.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to address these [[ ]]:

. . .

2.4.1 Connection of Reactor Pressure Vessel Isolation Valves to the Reactor Vessel

. . .

[[ ]] These [[ ]] will be established during detailed design of the [[ ]] and provided during future licensing activities.

eRAI No.: 9732
Date of eRAI Issue: 04/09/20

NRC Question NONE-1

10 CFR 50, Appendix A, General Design Criteria (GDC) 33 requires that a system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. Specifically, it states, "[t]he system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation."

Licensing Topical Report (LTR) NEDC-33910P, Revision 0, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," Section 4.1.10, states that [[
]] and concludes [[
]] In addition, it states "nonsafety-related injection systems may be used for manual addition of reactor coolant inventory by the operator using high-pressure CRD [control rod drive] injection or by reestablishing feedwater injection. These nonsafety-related injection systems can be used at any time following a LOCA. The timing of such operator actions is to be determined during the final ECCS performance analyses to be completed during future licensing activities."

i) The NRC staff understands that in some cases GEH believes that regulations are not applicable or can be satisfied by meeting the intent of the regulation. The requirement of GDC 33 appears to be applicable to the BWRX-300 design; however, to the extent GEH can show that the requirement is not necessary to meet the underlying purpose of the regulation, that showing would appear to address the "special circumstances" required to justify an exemption from the regulation under 10 CFR 50.12. The NRC staff notes that GDC 33 requires makeup provisions for both emergency and normal operating conditions. Therefore, the NRC staff requests GEH to clarify the method by which it plans to satisfy GDC 33 for the BWRX-300 design. In addition, the NRC staff requests GEH to indicate if it plans to develop a principle design criterion to address the design-specific nature by which the intent of GDC 33 will be satisfied.

ii) As noted above, LTR NEDC-33910 states that nonsafety-related systems may be used by operators, and the timing of such actions will be determined later. The NRC staff infers that GEH plans to credit nonsafety-related systems to mitigate the effects of small breaks in the reactor coolant system (RCS) pressure boundary, which is non-conforming to regulatory requirements. The staff requests GEH clarify its statements regarding the use and crediting of nonsafety-related systems following a LOCA or other small breaks in the RCS.

GEH Response to NRC Question NONE-1

As stated in NEDC-33910P, 10 CFR 50, Appendix A, GDC 33, Reactor coolant makeup, requires that a system to supply reactor coolant makeup for protection against small breaks in the RCPB shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

i) For the BWRX-300, [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed power and then remains in service without any further need of onsite or offsite electric power system operation. Further discussion regarding the selected long-term cooling period for the BWRX-300 is included in the GEH response to NRC Question NONE-2.

Because specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary [[ ]] 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 33. 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, a new Principal Design Criterion (PDC) 33 is proposed as described in the proposed changes to NEDC-33910P provided below.

ii) Although NEDC-33910P stated that “nonsafety-related systems may be used by operators, and the timing of such actions will be determined later,” those statements were not intended to mean that those actions would be credited to meet the regulatory requirements. Therefore, NEDC-33910P will be revised to delete those statements.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to include a proposed PDC 33 and provide justification for an exemption to these specific requirements of 10 CFR 50 Appendix A, GDC 33, which may be used as the bases for the necessary exemption during 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:

. . .

~~4.1.10~~4.1.11 10 CFR 50 Appendix A, GDC 33

. . .

Statement of Compliance: The safety analysis assumes that the small pipe breaks [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed ~~DC~~ power without any further need of onsite or offsite electric power system operation.

~~Although the BWRX-300 [[ ]] nonsafety-related injection systems may be used for manual addition of reactor coolant inventory by the operator using high pressure CRD injection or by reestablishing feedwater injection. These nonsafety-related injection systems can be used at any time following a LOCA. The timing of such operator actions is to be determined during the final ECCS performance analyses to be completed during future licensing activities.~~ Therefore, specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary [[

]]. 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 33. 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 33 is proposed:

> PDC 33, Reactor coolant makeup, a safety-related system to supply reactor coolant makeup for protection against small breaks in the RCPB is not needed to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary. In addition, the operation of the systems to mitigate the consequences of small breaks in the RCPB do not require offsite electric power system operation, and only require one-time automatic actuation using onsite Class 1E battery-backed power without any further need of onsite or offsite electric power system operation.

These statements of compliance and proposed PDC 33 may be used as the bases for the necessary exemption during 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.

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eRAI No.: 9732
Date of eRAI Issue: 04/09/20

NRC Question NONE-2

10 CFR 50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, requires that "Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section." The criteria in paragraph (b) include peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling with associated limits included in the regulation. Moreover, each criterion is quantitatively defined.

Licensing Topical Report (LTR) NEDC-33910P, Revision 0, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," Section 2.8, states that:

> The primary design goal of the BWRX-300 in response to a LOCA is that the core does not uncover so that the following acceptance criteria are met:
>
> - No significant fuel cladding heat up shall occur.
> - No significant fuel cladding oxidization shall occur.
> - No significant fuel cladding hydrogen generation shall occur.
> - Long-term cooling shall remove decay heat and maintain core temperature to acceptably low values.

These criteria lack definition and, as such, are subject to interpretation. The staff requests that GEH clarify whether it intends is to use and more explicitly define the criteria currently outlined in Section 2.8 as acceptance criteria in future licensing of the BWRX-300 design, or if GEH intends to use the criteria contained and explicitly defined in 10 CFR 50.46(b).

GEH Response to NRC Question NONE-2

GEH understands that the design goals provided in NEDC-33910P are subject to interpretation. These design goals were not intended to imply that they were to be used instead of the specific criteria defined in 10 CFR 50.46(b). Instead, it was intended to specify that the acceptance criteria in 10 CFR 50.46(b) can be demonstrated by maintaining reactor water level above the top of active fuel (TAF) such that the core does not uncover following a LOCA.

As described in NEDC-33910P, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS in compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling. [[

]] The analyses to demonstrate compliance will be provided during future licensing activities.

Therefore, NEDC-33910P will be revised to include a more explicit commitment to meeting the specific acceptance criteria defined in 10 CFR 50.46(b).

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to include compliance with the requirements of 10 CFR 50.46(b):

. . .

2.8 LOCA Acceptance Criteria

The primary design goal of the BWRX-300 in response to a LOCA is that the core does not uncover so that the ~~following~~ acceptance criteria of 10 CFR 50.46(b) are met. Maintaining reactor water level above TAF ensures that:

- No significant fuel cladding heat up ~~shall~~ occurs.
- No significant fuel cladding oxidization ~~shall~~ occurs.
- No significant fuel cladding hydrogen generation ~~shall~~ occurs.
- No significant changes in core geometry ~~shall~~ occurs.
- Long term cooling ~~shall~~to remove decay heat and maintain the core temperature to acceptably low values occurs.

~~These acceptance criteria can be demonstrated by maintaining reactor water level at TAF.~~

. . .

4.1.2 10 CFR 50.46

. . .

Statement of Compliance: The design goal is that the core does not uncover during LOCAs, and that no significant fuel cladding heatup occurs, such that the calculated maximum fuel element cladding temperature does not exceed the acceptance criterion of 2200°F. ~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(1).

. . .

Statement of Compliance: The design goal is that the core does not uncover during LOCAs, and that no significant fuel cladding oxidization occurs, such that the calculated total oxidation of the cladding does not nowhere exceed the acceptance criterion of 0.17 times the total cladding thickness before oxidation. ~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(2).

. . .

Statement of Compliance: The design goal is that the core does not uncover during LOCAs, and that no significant fuel cladding hydrogen generation occurs, such that the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed the acceptance criterion of 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react. ~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(3).

. . .

Statement of Compliance: The design goal is that the core does not uncover during LOCAs, and that no significant changes in core geometry occur, such that the acceptance criterion of the core remaining amenable to cooling is met. ~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(4).

. . .

Statement of Compliance: The design goal is that the core does not uncover during LOCAs, and that ~~no significant fuel cladding heatup~~long term cooling to remove decay heat and maintain the core temperature to acceptably low values occurs, such that after any calculated successful initial operation of the ECCS, the acceptance criteria of the calculated core temperature being maintained at an acceptably low value and decay heat being removed for the extended period of time required by the long lived radioactivity remaining in the core are met.

. . .

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(5).

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eRAI No.: 9732
Date of eRAI Issue: 04/09/20

NRC Question NONE-3

10 CFR 50.46(b)(5), requires the calculated core temperature to be maintained at an acceptably low value and decay heat to be removed for the extended period of time required by the long-lived radioactivity remaining in the core. 10 CFR 50, Appendix A, GDC 35 requires a system to provide abundant emergency core cooling. In addition, the 2008 Advanced Reactor Policy Statement (73 FRN 60612) describes the Commission's expectation for new reactors' safety systems to provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety functions (e.g. longer time constants and reduced operator actions).

One of the principle design requirements of the Electric Power Research Institute's (EPRI) Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) for passive designs is that the core must be cooled and containment integrity maintained with only safety-related SSCs and without reliance on ac power and operator actions for a minimum of 72 hours. The staff concludes in its corresponding safety evaluation (NUREG-1242, Vol. 3, Pt.1) that these requirements relative to the 72-hour capability are acceptable and are consistent with Commission policies related to passive designs.

Licensing Topical Report NEDC-33910P, specifies that adequate core cooling is maintained [[ ]], and the [[ ]] will be used to ensure full compliance with 10 CFR 50.46(b)(5), and GDC 35. However, it is unclear to the staff what long-term cooling timeframe will be established for the BWRX-300, and when water injection from nonsafety-related sources will be used by the operators after the design-basis long-term cooling period. Therefore, the staff requests GEH to define the design-basis long-term cooling timeframe for the BWRX-300 (e.g. 72 hours, 7 days, etc.) or ensure Licensing Topical Report NEDC-33910P clearly describes where it will be addressed in future submittals.

GEH Response to NRC Question NONE-3

GEH understands that the design-basis long-term cooling timeframe for the BWRX-300 was not clearly provided in NEDC-33910P. Instead, although not explicitly defined, GEH intended for the final ECCS performance analyses to provide additional insight for when water injection from nonsafety-related sources will be used by the operators after the design-basis long-term cooling period. However, GEH has completed some preliminary evaluations to determine a conservative long-term cooling period for the BWRX-300.

For the BWRX-300, [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed power and then remains in service without any further need of onsite or offsite electric power system operation.

The selected long-term cooling timeframe of [[ ]] is sufficient to maintain reactor water level above TAF for both [[

]]

Following the worst-case postulated LOCA (i.e., [[ ]]), 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.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to include compliance with the requirements of 10 CFR 50.46(b)(5):

. . .

4.1.2 10 CFR 50.46

. . .

- Regulatory Requirement: 10 CFR 50.46(b)(5), Long-term cooling, requires that after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

  Statement of Compliance: The design goal is that the core does not uncover during LOCAs, and that ~~no significant fuel cladding heatup~~long term cooling to remove decay heat and maintain the core temperature to acceptably low values occurs, such that after any calculated successful initial operation of the ECCS, the acceptance criteria of the calculated core temperature being maintained at an acceptably low value and decay heat being removed for the extended period of time required by the long lived radioactivity remaining in the core are met.

  For the BWRX-300, [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed DC power and then remains in service for at least [[ ]] without any further need of onsite or offsite electric power system operation.

The selected long-term cooling timeframe of [[ ]] is sufficient to maintain reactor water level above TAF for both [[

]] Following the worst-case postulated LOCA (i.e., [[ ]]), 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.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities. Therefore, the BWRX 300 design will meet the requirements of 10 CFR 50.46(b)(5).

Appendix B
GEH Responses to NRC RAIs on NEDC-33910P Revision 0 Supplement 1

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eRAI No.: 9730
Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-1

Section 2.1.2 in NEDC-33910 describes the Isolation Condenser System (ICS) for the BWRX-300 nuclear power plant. This section indicates that the ICS includes [[ ]]. To support the NRC staff review of NEDC-33910P [sic] and its conformance to 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 35 and 37 for the IC condensate return valves, the NRC staff requests that GEH describe the following:

(a) Any first of a kind (FOAK) features,

(b) Valve and actuator types,

(c) Valve size,

(d) Qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100,

(e) Plans for valve and actuator diversity,

(f) Incorporation of lessons learned from international operating experience where ICS valves failed to open as designed,

(g) Accessibility for inservice testing (IST) activities in accordance with 10 CFR 50.55a,

(h) Design features to avoid thermal binding or pressure locking of the valves, and

(i) OM Code leakage classification.

If any of this information is not available at this time, the staff requests that GEH indicate its plans to provide this information during future licensing activities for the BWRX-300 nuclear power plant.

GEH Response to NRC Question 03.09.06-1

Although detailed design of the IC condensate return valves has not yet been completed, the design functions and features of the IC condensate return valves are anticipated to be like what is described in Section 5.4.6 of the Economically Simplified Boiling Water Reactor (ESBWR) Design Control Document (DCD). Compliance with the requirements of 10 CFR 50, Appendix A, General Design Criteria (GDC) 1, GDC 2, GDC 4, and GDC 37 for the IC condensate return valves is anticipated to be the same as described in Section 5.4.6 of the ESBWR DCD. However, NEDC-33910P is not requesting NRC approval for the IC condensate return valves to meet these GDC. Instead, it is requested that the limited design requirements specified for the ICS including the IC condensate return valves be found acceptable for ensuring that the ICS can perform the limited functions that are a subject of NEDC-33910P in demonstrating compliance with 10 CFR 50, Appendix A, GDC 35. These include the following design functions:

- ICS is initiated automatically on high reactor pressure vessel (RPV) pressure indicating an overpressure event or on signals indicating a loss-of-coolant accident (LOCA).
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]
- [[ ]]

NEDC-33910P did not describe compliance with 10 CFR 50 Appendix A, GDC 2, and will be revised to include this information. The conclusion of this additional information to be provided is that the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 2. Subsection 5.1.6 of NEDC-33911P, BWRX-300 Containment Performance, also describes that the RPV isolation valves will comply with 10 CFR 50 Appendix A, GDC 2.

As described in NEDC-33910P, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS in compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling. [[

]] 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 35.

NEDC-33910P did not describe compliance with 10 CFR 50 Appendix A, GDC 37, and will be revised to include this information. The conclusion of this additional information to be provided is that the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 37.

The details of the design functions and features beyond those requirements described above and already described in NEDC-33910P are to be addressed 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. The following discussions address the specific questions provided:

(a) No FOAK features will be specified for the IC condensate return valves.

(b) Valve and actuator types will be addressed in the detailed design of the valves.

(c) Valve size will be addressed in the detailed design of the valves and will be specified during future licensing activities.

(d) Qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100, will be addressed in the detailed design of the valves and will be specified during future licensing activities.

(e) As described in NEDC-33910P, the [[

]] These requirements provide for sufficient description of valve and actuator diversity in support of the NEDC-33910P conclusions that [[
]].

(f) Incorporation of lessons learned from international operating experience where ICS valves failed to open as designed will be addressed in the detailed design of the valves and will be specified during future licensing activities. However, NEDC-33910P will be revised to include discussion for Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, as this operating experience may be applicable to the detailed design of the valves.

(g) Accessibility for IST activities in accordance with 10 CFR 50.55a will be addressed in the detailed design of the valves and will consider IST requirements like those described in ESBWR DCD, Tier 2, 26A6642AK, Rev. 10, April 2014, Table 3.9-8. The IC condensate return valves design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, 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. The specific IST requirements for the BWRX-300 design will be specified during future licensing activities.

(h) Design features to avoid thermal binding or pressure locking of the valves are not necessary for the IC condensate return valves. During normal operation with the ICS in standby, there are [[

]].

(i) IST requirements will be like those described in ESBWR DCD, Tier 2, 26A6642AK, Rev. 10, April 2014, Table 3.9-8, which consists of position indication verification (ASME OM Code Paragraph ISTC-3700), stroke open testing (ASME OM Code Paragraph ISTC-3521), and fail open testing (ASME OM Code Paragraph ISTC-3560). Like the ESBWR, the IC condensate return valves are classified as ASME OM Code Category B valves, and do not require leakage testing. However, the specific IST requirements for the BWRX-300 design will be specified during future licensing activities.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to reflect [[

]] and to add the following as new Subsections 4.1.5 and 4.1.13 to address compliance with 10 CFR 50, Appendix A, GDC 2 and GDC 37, and to address addition of Generic Letter 95-07 as new Subsection 4.5.2:

. . .

1.1 Purpose

. . .

The design of the RPV isolation valves and ICS meet the requirements of 10 CFR 50.46(b) and 10 CFR 50 Appendix A, General Design Criteria, GDC 1, GDC 2, GDC 4, GDC 14, GDC 30, GDC 31, GDC 33, ~~and~~ GDC 35, and GDC 37.

. . .

4.1.5 10 CFR 50 Appendix A, GDC 2

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 2, Design bases for protection against natural phenomena, requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

  Statement of Compliance: The BWRX-300 RPV isolation and overpressure protection design features, [[ ]] are to be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. Specific design requirements for the [[

]] used to verify the capability to perform their safety functions, and the natural phenomena and effects evaluated, will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 2.

. . .

4.1.13 10 CFR 50 Appendix A, GDC 37

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 37, Testing of emergency core cooling system, requires that the emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

  Statement of Compliance: As previously described, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS in compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling, which is assured for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS through the use of [[ ]].

  Specific requirements for periodic pressure and functional testing of the [[ ]] to assure (1) the structural and leaktight integrity of these components, (2) the operability and performance of these active components, and (3) the operability of the systems as a whole containing the [[ ]] will be provided during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 37.

. . .

4.5.2 Generic Letter 95-07

Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, dated August 17, 1995, contains a request to ensure that safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding are evaluated to ensure they are capable of performing the safety functions described in the licensing basis. This guidance will be evaluated for applicability during future licensing activities.

eRAI No.: 9730
Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-2

Section 2.2 in NEDC-33910 provides a general overview of reactor pressure vessel (RPV) isolation concept, and Section 2.5 in NEDC-33910 specifies the RPV isolation valve design requirements for the BWRX-300 nuclear power plant. These sections indicate that there will be [[

]] To support the NRC staff review of NEC-33910 and its conformance to 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 54, 55, and 56 for the RPV isolation valves, the NRC staff requests that GEH describe the following:

(a) Any FOAK features,

(b) Valve types and sizes,

(c) Qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100,

(d) Plans for valve diversity,

(e) Accessibility for IST activities in accordance with 10 CFR 50.55a,

(f) Design to avoid thermal binding or pressure locking of the valves, and

(g) ASME OM Code leakage classification.

If any of this information is not available at this time, the staff requests that GEH indicate its plans to provide this information during future licensing activities for the BWRX-300 nuclear power plant.

GEH Response to NRC Question 03.09.06-2

The detailed design of the RPV isolation valve assemblies has not yet been completed. Compliance with the requirements of 10 CFR 50, Appendix A, General Design Criteria (GDC) 1, GDC 2 and GDC 4 for the RPV isolation valve assemblies is anticipated to be the same as for other ASME Class 1 valves described in the ESBWR DCD. Therefore, the BWRX-300 will comply with the requirements of GDC 1, GDC 2 and GDC 4 for the RPV isolation valve assemblies. NEDC-33910P is requesting that the limited design requirements specified for the RPV isolation valve assemblies be found acceptable for ensuring that the limited functions that are a subject of NEDC-33910P are met in demonstrating compliance with 10 CFR 50, Appendix A, GDC 33 and GDC 35.

Compliance with the requirements of 10 CFR 50, Appendix A, GDC 54, GDC 55, and GDC 56 as related to the design of the [[ ]] are described in Subsections 5.1.21, 5.1.22 and 5.1.26 of NEDC-33911P, Revision 0, BWRX-300 Containment Performance.

The details of the design functions and features beyond those requirements described above and already described in NEDC-33910P are to be addressed 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. The following discussions address the specific questions provided:

(a) Any FOAK features will be addressed in the detailed design of the valves and will be specified during future licensing activities.

(b) Valve types and sizes will be addressed in the detailed design of the valves and will be specified during future licensing activities.

(c) Qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100, will be addressed in the detailed design of the valves and will be specified during future licensing activities.

(d) As described in NEDC-33910P, the [[ ]] These requirements provide for sufficient description of diverse actuation in support of the NEDC-33910P conclusions that the [[ ]] to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(5). The [[ ]] for any breaks that would result in a loss of reactor coolant at a rate in excess of the capability of the nonsafety-related normal reactor coolant makeup systems, which for the BWRX-300 includes LOCA break sizes [[ ]]. The worst-case single failure affecting the [[ ]] does not prevent fulfillment of the required ECCS design functions. Therefore, there is no need for the [[ ]].

(e) Accessibility for IST activities in accordance with 10 CFR 50.55a will be addressed in the detailed design of the valves and will consider IST requirements like those described in ESBWR DCD, Tier 2, 26A6642AK, Rev. 10, April 2014, Table 3.9-8, for other active ASME Class 1 valves. The specific IST requirements for the BWRX-300 design will be specified during future licensing activities.

(f) Design features to avoid thermal binding or pressure locking of the valves are not necessary for the RPV isolation valves. The safety function of the RPV isolation valves is to close as opposed to having an opening safety function. Automatic actuation of [[ ]] performs the function to mitigate the effects of a LOCA. [[ ]] is a one-time action to open or close appropriate valves during accident response which may also be initiated manually as a one-time action if necessary.

(g) As described in NEDC-33910P all BWRX-300 RPV isolation valves shall have a proven low leakage potential. Design and administrative leakage limits are applied to valve selection during the BWRX-300 preliminary design and are based on plant design and event evaluations using offsite dose consequences compared to regulatory limits as well as containment design limits. The leakage criteria are analyzed as part of the plant safety analysis. Therefore, the RPV isolation valves are classified as ASME OM Code Category A valves requiring a seat leakage rate test (ASME OM Code Paragraph ISTC-3600). IST requirements will be like what is described in ESBWR DCD, Tier 2, 26A6642AK, Rev. 10, April 2014, Table 3.9-8, for ASME Class 1 valves. The specific IST requirements for the BWRX-300 design will be specified during future licensing activities.

Proposed Changes to NEDC-33910P, Revision 0

None

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eRAI No.: 9730
Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-6

Section 4.1.3 in NEDC-33910 indicates that the requirements of 10 CFR 50.55a will be satisfied. This section specifically references the RPV isolation valves. This section also states that no alternative approach, exception, or exemption from these requirements is required. The NRC staff requests that GEH describe the compliance with the requirements in 10 CFR 50.55a for the [[ ]]. The staff also requests that GEH clarify the intent of the statement in this section and elsewhere in NEDC-33910 that no alternative approach, exception, or exemption from these requirements is required.

GEH Response to NRC Question 03.09.06-6

GEH understands that the use of statements like "Full compliance with these requirements is to be demonstrated in the preliminary and final design of the BWRX-300 to be completed during future licensing activities. Therefore, no alternative approach, exception, or exemption from these requirements is required" are not appropriate where used. Therefore, the discussion containing those statements, including in Section 4.1.3, are proposed to be revised in NEDC-33910P as described below for clarification of the intent, which is to provide a commitment to meeting the applicable regulatory requirements during detailed design.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to reflect [[ ]] and to include compliance with the requirements of the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(5) using a long-term cooling timeframe of [[ ]], and to replace the use of these statements with appropriate statements of compliance to provide a commitment to meeting the applicable regulatory requirements during detailed design:

. . .

2.7 Categories of Pipe Breaks

. . .

The emergency core cooling system (ECCS) evaluation model for the BWRX-300 is to be developed using previously approved methodologies used in the ECCS performance analyses for the ESBWR as modified using BWRX-300 specific design requirements and parameters. ~~Full compliance with these requirements is to be demonstrated in the final~~Final ECCS performance analyses are to be completed during future licensing activities using an NRC-approved BWRX-300 ECCS evaluation model. Methodology for containment response is described in ~~a separate~~ LTR NEDC-33911P, BWRX-300 Containment Performance [Reference 5.6].

. . .

4.1.2 10 CFR 50.46

. . .

Based on the above evaluation, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in ~~full~~ compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS ~~in full compliance~~consistent with the definition of a LOCA in 10 CFR 50.46(c)(1).

Although this is a~~n alternative and~~ non-traditional approach for the design of the ECCS for past LWRs, no [[ ]] is ~~required~~needed following the worst case LOCA to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b). ~~Final ECCS performance analyses are to~~ The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, ~~no exception or exemption is required for these regulatory requirements. Full compliance with these requirements is to be demonstrated in the final ECCS performance analyses to be completed during future licensing activities.~~the BWRX-300 design will meet the requirements of 10 CFR 50.46.

. . .

Statement of Compliance: The ECCS evaluation model for the BWRX-300 is to be developed using previously approved methodologies used in the ECCS performance analyses for the ESBWR as modified using BWRX-300 specific design requirements and parameters. ~~Full compliance with these requirements is to be demonstrated in the final ECCS performance analyses to~~The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities using an NRC-approved BWRX-300 ECCS evaluation model which includes reasonably conservative methods. Because of the BWRX-300 acceptance criteria being applied to bound the 10 CFR 50.46(b) acceptance criteria, uncertainties will be addressed in the BWRX-300 ECCS evaluation model to verify that there is a high level of probability that the BWRX-300 acceptance criteria would not be exceeded rather than the 10 CFR 50.46(b) acceptance criteria. The BWRX-300 evaluation model will not use the alternatives provided in 10 CFR 50 Appendix K.

Therefore, ~~no alternative approach, exception, or exemption from these requirements is required~~the BWRX-300 design will meet the requirements of 10 CFR 50.46(a)(1).

. . .

Statement of Compliance: The ~~design goal~~BWRX-300 acceptance criteria in response to a LOCA is that ~~the core does not uncover during LOCAs~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant fuel cladding heatup occurs, such that the calculated maximum fuel element cladding temperature does not exceed the acceptance criterion of 2200°F.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the finalFinal ECCS performance analyses are to~~The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(1).

. . .

Statement of Compliance: The ~~design goal~~BWRX-300 acceptance criteria in response to a LOCA is that ~~the core does not uncover during LOCAs~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant fuel cladding oxidization occurs, such that the calculated total oxidation of the cladding does not nowhere exceed the acceptance criterion of 0.17 times the total cladding thickness before oxidation.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the finalFinal ECCS performance analyses are to~~The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(2).

. . .

Statement of Compliance: The ~~design goal~~BWRX-300 acceptance criteria in response to a LOCA is that ~~the core does not uncover during LOCAs~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant fuel cladding hydrogen generation occurs, such that the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed the acceptance criterion of 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the finalFinal ECCS performance analyses are to~~The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(3).

. . .

Statement of Compliance: The ~~design goal~~BWRX-300 acceptance criteria in response to a LOCA is that ~~the core does not uncover during LOCAs~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant changes in core geometry occur, such that the acceptance criterion of the core remaining amenable to cooling is met.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the finalFinal ECCS performance analyses are to~~The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will~~Final ECCS performance analyses are to~~ be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(4).

. . .

Statement of Compliance: The ~~design goal~~BWRX-300 acceptance criteria in response to a LOCA is that ~~the core does not uncover during LOCAs~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that ~~no significant fuel cladding heatup~~long-term cooling removes decay heat and maintains the core temperature to acceptably low values ~~occurs~~, such that after any calculated successful initial operation of the ECCS, the acceptance criteria of the calculated core temperature being maintained at an acceptably low value and decay heat being removed for the extended period of time required by the long lived radioactivity remaining in the core are met.

For the BWRX-300, [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed power without any further need of onsite or offsite electric power system operation. ~~For the BWRX-300, [[~~

~~]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed DC power and then remains in service for at least [[ ]] without any further need of onsite or offsite electric power system operation.~~

The selected long-term cooling timeframe of [[ ]] is sufficient to maintain reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range for both [[

]] Following [[

]], the [[ ]] continues to provide long-term cooling to meet the BWRX-300 acceptance criteria in response to a LOCA and only requires operator action to [[ ]] after approximately seven days.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~~~Final ECCS performance analyses are to~~The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will~~Final ECCS performance analyses are to~~ be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(5).

. . .

4.1.3 10 CFR 50.55a

. . .

These requirements are to be implemented during detailed design of the safety-related components of the [[ ]].~~Full compliance with these requirements is to be demonstrated in the preliminary and final design of the BWRX-300 to be completed during future licensing activities.~~

Therefore, ~~no alternative approach, exception, or exemption from these requirements is required~~the BWRX-300 design will meet the requirements of 10 CFR 50.55a.

. . .

4.1.4 10 CFR 50 Appendix A, GDC 1

. . .

~~Full compliance with these requirements is to be demonstrated in the preliminary and final design of the BWRX-300 to be completed during future licensing activities.~~

Therefore, ~~no alternative approach, exception, or exemption from these requirements is required~~the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 1.

. . .

~~4.1.5~~4.1.6 10 CFR 50 Appendix A, GDC 4

. . .

Therefore, ~~no alternative approach, exception, or exemption from these requirements is required~~the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 4.

. . .

~~4.1.6~~4.1.7 10 CFR 50 Appendix A, GDC 14

. . .

This results in an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture, in ~~full~~ compliance with this criterion. For piping that have [[ ]]. Further design details are to be described during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 14.

. . .

~~4.1.7~~4.1.8 10 CFR 50 Appendix A, GDC 15

. . .

The combination of RPS and [[ ]] design features ensure that the acceptance criteria for each component in the protected system are met including 1) the overpressure does not exceed 1.1 times design pressure for the expected system pressure transient condition, and 2) the calculated stress intensity and other design limitations for Service Level C are not exceeded for the unexpected system excess pressure transient condition, with further design details to be described during future licensing activities.~~ Full compliance with these requirements is to be demonstrated in the preliminary and final design of the BWRX-300 to be completed during future licensing activities.~~

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 15.

. . .

~~4.1.8~~4.1.9 10 CFR 50 Appendix A, GDC 30

. . .

In addition, means are to be provided to detect and identify the location of the source of reactor coolant leakage, including the components of the ICS and RPV isolation valves, for components of the RCPB, with further design details to be described during future licensing activities.~~ Full compliance with these requirements is to be demonstrated in the preliminary and final design of the BWRX-300 to be completed during future licensing activities.~~

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 30.

. . .

~~4.1.9~~4.1.10 10 CFR 50 Appendix A, GDC 31

. . .

The components of the RCPB, including the ICS and RPV isolation valves, are to be designed with sufficient margin to assure that these requirements are met, with further design details to be described during future licensing activities.~~ Full compliance with these requirements is to be demonstrated in the preliminary and final design of the BWRX-300 to be completed during future licensing activities.~~

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 31.

. . .

~~4.1.11~~4.1.12 10 CFR 50 Appendix A, GDC 35

. . .

As previously described, the combined design features of the [[ ]] meet the as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in ~~full~~ compliance with the criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS in ~~full~~ compliance with the definition of a LOCA in 10 CFR 50.46(c)(1).

. . .

~~Full compliance with these requirements is to be demonstrated in the final ECCS performance~~The analyses to ~~be completed~~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 35.

. . .

4.2.1 Regulatory Guide 1.26

. . .

Therefore, the BWRX-300 design conforms to the guidance for the RPV isolation valves and the ICS, including regulatory positions of RG 1.26~~, without requiring an alternative approach or exception~~.

. . .

4.2.2 Regulatory Guide 1.29

. . .

Therefore, the BWRX-300 design conforms to the guidance, including regulatory positions of RG 1.29~~, without requiring an alternative approach or exception~~.

. . .

4.2.3 Regulatory Guide 1.45

. . .

Additionally, the means for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage in compliance with the requirements of 10 CFR 50 Appendix A, GDC 30, and the requirements for in-service inspection and testing of the [[ ]] in compliance with the requirements of 10 CFR 50.55a, are to be demonstrated during future licensing activities~~, and no alternative approach, exception, or exemption from these requirements is required~~.

Therefore, the BWRX-300 design conforms to the guidance, including regulatory positions of RG 1.45~~, without requiring an alternative approach or exception~~.

. . .

4.2.4 Regulatory Guide 1.84

. . .

Compliance with the requirements of 10 CFR 50.55a, including the use of ASME B&PV Section III Code Cases endorsed in RG 1.84 where necessary, is to be demonstrated during future licensing activities~~, and no alternative approach, exception, or exemption from these requirements is required~~.

Therefore, the BWRX-300 design conforms to the guidance, including regulatory positions of RG 1.84~~, without requiring an alternative approach or exception~~.

eRAI No.: 9730
Date of eRAI Issue: 04/09/20

NRC Question 03.09.06-8

Section 4.1.10 in NEDC-33910 describes compliance with the intent of GDC 33 regarding reactor coolant makeup. The NRC staff requests that GEH clarify its statement that the intent of this criterion is met.

GEH Response to NRC Question 03.09.06-8

As stated in NEDC-33910P, 10 CFR 50, Appendix A, GDC 33, Reactor coolant makeup, requires that a system to supply reactor coolant makeup for protection against small breaks in the RCPB shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

~~For the BWRX-300, [[~~

~~]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed power without any further need of onsite or offsite electric power system operation. Further discussion regarding the selected long-term cooling period for the BWRX-300 is included in the GEH response to NRC Question NONE-2 in Enclosures 3 (Proprietary) and 6 (Non-Proprietary).~~

~~Because specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary [[ ]] 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 33. 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, a new Principal Design Criterion (PDC) 33 is proposed as described in the proposed changes to NEDC-33910P provided below.~~GDC 33 applies to small leaks in the RCPB that are [[

]] (e.g., leakage from flanges or cracks in piping or other components), and which do not exceed the capability of the nonsafety-related high-pressure CRD system used as normal reactor coolant makeup during power operations. The maximum allowed leakage rate for continuing power operation is stipulated in the plant Technical Specifications. For leakage greater than the maximum allowed leakage rate, automatic reactor scram and automatic actuation of the [[ ]] is not anticipated for most of these small leaks because the normal means of makeup from the high-pressure CRD system and feedwater maintains the level in the normal operating range. However, these small leaks which do not exceed the capability of the nonsafety-related high-pressure CRD system are evaluated using specified acceptable fuel design limits rather than the BWRX-300 acceptance criteria in response to a LOCA.

The high-pressure CRD system pumps can operate using either onsite electric power system operation (assuming offsite power is not available) backed up by nonsafety-related standby diesel generators and offsite electric power system operation (assuming onsite power is not available).

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to address ~~proposal of a PDC 33 and provide justification for an exemption to these specific~~compliance with the regulatory requirements of 10 CFR 50 Appendix A, GDC 33~~, which may be used as the bases for the necessary exemption during 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~~:

. . .

~~4.1.10~~4.1.11 10 CFR 50 Appendix A, GDC 33

. . .

Statement of Compliance: ~~The safety analysis assumes that the small pipe breaks [[~~

~~]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed DC power without any further need of onsite or offsite electric power system operation.~~

~~Although the BWRX-300 [[ ]] nonsafety-related injection systems may be used for manual addition of reactor coolant inventory by the operator using high pressure CRD injection or by reestablishing feedwater injection. These nonsafety-related injection systems can be used at any time following a LOCA. The timing of such operator actions is to be determined during the final ECCS performance analyses to be completed during future licensing activities. Therefore, specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary [[~~

~~]]. 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 33. 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 33 is proposed:~~

~~PDC 33, Reactor coolant makeup, a safety-related system to supply reactor coolant makeupfor protection against small breaks in the is not needed to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary. In addition, the operation of the systems to mitigate the consequences of small breaks in the RCPB do not require offsite electric power system operation, and only require one-time automatic actuation using onsite Class 1E battery-backed power without any further need of onsite or offsite electric power system operation.~~

~~These statements of compliance and proposed PDC 33 may be used as the bases for the necessary exemption during 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.~~

~~The safety analysis assumes that the small pipe breaks [[~~

~~]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time actuation using onsite Class 1E battery-backed DC power.~~

~~Although the BWRX-300 [[ ]] nonsafety-related injection systems may be used for manual addition of reactor coolant inventory by the operator using high pressure CRD injection or by reestablishing feedwater injection. These nonsafety-related injection systems can be used at any time following a LOCA. The timing of such operator actions is to be determined during the final ECCS performance analyses to be completed during future licensing activities. Therefore, specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary [[~~

~~]]~~GDC 33 applies to small leaks in the RCPB that are [[

]] (e.g., leakage from flanges or cracks in piping or other components), and which do not exceed the capability of the nonsafety-related high-pressure CRD system used as normal reactor coolant makeup during power operations. The maximum allowed leakage rate for continuing power operation is stipulated in the plant Technical Specifications. For leakage greater than the maximum allowed leakage rate, automatic reactor scram and automatic actuation of the [[ ]] is not anticipated for most of these small leaks because the normal means of makeup from the high-pressure CRD system and feedwater maintains the level in the normal operating range. However, these small leaks which do not exceed the capability of the nonsafety-related high-pressure CRD system are evaluated using specified acceptable fuel design limits rather than the BWRX-300 acceptance criteria in response to a LOCA.

The high-pressure CRD system pumps can operate using either onsite electric power system operation (assuming offsite power is not available) backed up by nonsafety-related standby diesel generators and offsite electric power system operation (assuming onsite power is not available).

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 33.

eRAI No.: 9731
Date of eRAI Issue: 04/09/20

NRC Question 03.06.02-1

Section 2.2 in LTR NEDC-33910 states that for piping that have [[ ]]. The NRC staff requests that GEH describe the design of the [[ ]].

GEH Response to NRC Question 03.06.02-1

The design of the [[ ]] Further details regarding the [[ ]] are provided in Subsections 2.2.2, 2.2.7.1 and 5.1.22 of NEDC-33911P, BWRX-300 Containment Performance. As stated in NEDC-33911P, [[ ]] extending to the containment wall, the BWRX-300 design requirements include identifying postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, Part B, Item 1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2).

Proposed Changes to NEDC-33910P, Revision 0

~~None~~NEDC-33910P, Revision 0, will be revised to describe the [[ ]] consistent with the discussions in NEDC-33911P, BWRX-300 Containment Performance:

. . .

2.4.1 Connection of Reactor Pressure Vessel Isolation Valves to ~~the~~ Reactor Vessel

. . .

[[ ]], extending to the containment wall, the BWRX-300 design requirements include identifying postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, Part B, Item 1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2).

Where break locations are selected without the benefit of stress calculations, breaks are postulated at the piping welds to each fitting, valve, or welded attachment.

eRAI No.: 9731
Date of eRAI Issue: 04/09/20

NRC Question 03.06.02-2

Section 4.1.5 of the LTR states that for piping connected to [[ ]], the BWRX-300 design requirements include consideration of the acceptance criteria to identify postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment." To support the NRC staff review of NEDC-33910P [sic] and its conformance to 10 CFR Part 50, Appendix A, GDC 4 related to protection against postulated breaks, the NRC staff requests [sic] the following:

(a) That GEH clarify whether a terminal end break would be postulated at the piping connection to the [[ ]].

(b) If a piping terminal break is postulated, that GEH describe how the BWRX-300 design requirements will provide assurance that the functionality of the [[ ]] will not be affected by the dynamic effects resulting from the postulated terminal end break.

(c) If breaks are not postulated at those piping connection terminal ends, that GEH describe how the BWRX-300 design provides reasonable assurance that the probability of postulated failures at those connection terminal ends is extremely low.

GEH Response to NRC Question 03.06.02-2

As stated in Subsections 2.2.2, 2.2.7.1 and 5.1. of NEDC-33911P, BWRX-300 Containment Performance, [[

]] extending to the containment wall, the BWRX-300 design requirements include identifying postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, Part B, Item 1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2). Therefore, the following responses to these specific questions are provided:

(a) A terminal end break would be postulated at the piping connection to the [[ ]] as specified in BTP 3-4, Part B, Item 1(iii)(2).

(b) As described in Subsection 5.1.7 of NEDC-33911P, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 4. This will be addressed during detailed design of the RPV isolation valves by designing the valves to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents, and the valves will be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. Qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100, will be addressed in the detailed design of the valves and will be specified during future licensing activities. This is further discussed in response to eRAI 9730 NRC Question 03.09.06-2.

(c) As described in Subsection 2.2.2 of NEDC-33911P, a terminal end break would be postulated at the piping connection to the [[ ]] as specified in BTP 3-4, Part B, Item 1(iii)(2).

Proposed Changes to NEDC-33910P, Revision 0

None

eRAI No.: 9731
Date of eRAI Issue: 04/09/20

NRC Question 03.06.02-3

Section 2.4.1 of the LTR states that the bolted flange connections of the [[ ]]. GEH also states that the design provisions, the leakage detection systems design, and the inservice inspection requirements demonstrate that the probability of gross rupture at these [[ ]] is extremely low. To support the NRC staff review of NEDC-33910P [sic] and its conformance to 10 CFR Part 50, Appendix A, GDC 4 related to protection against postulated breaks, the NRC staff requests that GEH indicate the material of construction for these [[ ]] and discuss why the potential failure mechanisms, such as vibration, water hammer, corrosion, and unexpected modes of operation, are not applicable for these [[ ]].

GEH Response to NRC Question 03.06.02-3

The material of construction for the [[ ]] will be determined during detailed design and addressed in future licensing activities. Therefore, the discussions addressing these [[ ]] are proposed to be revised in NEDC-33910P as described below.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to address these [[ ]]:

. . .

2.4.1 Connection of Reactor Pressure Vessel Isolation Valves to the Reactor Vessel

. . .

[[ ]] These [[ ]] will be established during detailed design of the [[ ]] and provided during future licensing activities.

eRAI No.: 9732
Date of eRAI Issue: 04/09/20

NRC Question NONE-1

10 CFR 50, Appendix A, General Design Criteria (GDC) 33 requires that a system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. Specifically, it states, "[t]he system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation."

Licensing Topical Report (LTR) NEDC-33910P, Revision 0, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," Section 4.1.10, states that [[ ]] and concludes [[ ]] In addition, it states "nonsafety-related injection systems may be used for manual addition of reactor coolant inventory by the operator using high-pressure CRD [control rod drive] injection or by reestablishing feedwater injection. These nonsafety-related injection systems can be used at any time following a LOCA. The timing of such operator actions is to be determined during the final ECCS performance analyses to be completed during future licensing activities."

i) The NRC staff understands that in some cases GEH believes that regulations are not applicable or can be satisfied by meeting the intent of the regulation. The requirement of GDC 33 appears to be applicable to the BWRX-300 design; however, to the extent GEH can show that the requirement is not necessary to meet the underlying purpose of the regulation, that showing would appear to address the "special circumstances" required to justify an exemption from the regulation under 10 CFR 50.12. The NRC staff notes that GDC 33 requires makeup provisions for both emergency and normal operating conditions. Therefore, the NRC staff requests GEH to clarify the method by which it plans to satisfy GDC 33 for the BWRX-300 design. In addition, the NRC staff requests GEH to indicate if it plans to develop a principle design criterion to address the design-specific nature by which the intent of GDC 33 will be satisfied.

ii) As noted above, LTR NEDC-33910 states that nonsafety-related systems may be used by operators, and the timing of such actions will be determined later. The NRC staff infers that GEH plans to credit nonsafety-related systems to mitigate the effects of small breaks in the reactor coolant system (RCS) pressure boundary, which is non-conforming to regulatory requirements. The staff requests GEH clarify its statements regarding the use and crediting of nonsafety-related systems following a LOCA or other small breaks in the RCS.

GEH Response to NRC Question NONE-1

As stated in NEDC-33910P, 10 CFR 50, Appendix A, GDC 33, Reactor coolant makeup, requires that a system to supply reactor coolant makeup for protection against small breaks in the RCPB shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

i) GDC 33 applies to small leaks in the RCPB that are [[ ]] (e.g., leakage from flanges or cracks in piping or other components), and which do not exceed the capability of the nonsafety-related high-pressure CRD system used as normal reactor coolant makeup during power operations. The maximum allowed leakage rate for continuing power operation is stipulated in the plant Technical Specifications. For leakage greater than the maximum allowed leakage rate, automatic reactor scram and automatic actuation of the [[ ]] is not anticipated for most of these small leaks because the normal means of makeup from the high-pressure CRD system and feedwater maintains the level in the normal operating range. However, these small leaks which do not exceed the capability of the nonsafety-related high-pressure CRD system are evaluated using specified acceptable fuel design limits rather than the BWRX-300 acceptance criteria in response to a LOCA.

~~iii)~~ The high-pressure CRD system pumps can operate using either onsite electric power system operation (assuming offsite power is not available) backed up by nonsafety-related standby diesel generators and offsite electric power system operation (assuming onsite power is not available).~~For the BWRX-300, [[~~

~~]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one time automatic actuation using onsite Class 1E battery-backed power and then remains in service without any further need of onsite or offsite electric power system operation. Further~~

~~discussion regarding the selected long-term cooling period for the BWRX-300 is included in the GEH response to NRC Question NONE-2.~~

~~Because specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary [[ ]] 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 33. 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, a new Principal Design Criterion (PDC) 33 is proposed as described in the proposed changes to NEDC-33910P provided below.~~

~~i)~~ii) Although NEDC-33910P stated that "nonsafety-related systems may be used by operators, and the timing of such actions will be determined later," those statements were not intended to mean that those actions would be credited to meet the regulatory requirements of 10 CFR 50.46(b). Therefore, NEDC-33910P will be revised to delete those statements.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to ~~include a proposed PDC 33 and provide justification for an exemption to these specific~~ address compliance with the regulatory requirements of 10 CFR 50 Appendix A, GDC 33, which may be used as the bases for the necessary exemption during 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:

. . .

~~4.1.10~~4.1.11 10 CFR 50 Appendix A, GDC 33

. . .

Statement of Compliance: ~~The safety analysis assumes that the small pipe breaks [[~~

~~]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time actuation using onsite Class 1E battery-backed DC power.~~

~~Although the BWRX-300 [[ ]] nonsafety-related injection systems may be used for manual addition of reactor coolant inventory by the operator using high pressure CRD injection or by reestablishing feedwater injection. These nonsafety-related injection systems can be used at any time following a LOCA. The timing of such operator actions is to be determined during the final ECCS performance analyses to be~~

~~completed during future licensing activities. Therefore, specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary [[~~

~~]] The safety analysis assumes that the small pipe breaks [[~~

~~]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed DC power without any further need of onsite or offsite electric power system operation.~~

~~Although the BWRX-300 [[ ]] nonsafety-related injection systems may be used for manual addition of reactor coolant inventory by the operator using high pressure CRD injection or by reestablishing feedwater injection. These nonsafety-related injection systems can be used at any time following a LOCA. The timing of such operator actions is to be determined during the final ECCS performance analyses to be completed during future licensing activities. Therefore, specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary [[~~

~~]]. 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 33. 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 33 is proposed:~~

~~PDC 33, Reactor coolant makeup, a safety-related system to supply reactor coolant makeup for protection against small breaks in the RCPB is not needed to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary. In addition, the operation of the systems to mitigate the consequences of small breaks in the RCPB do not require offsite electric power system operation, and only require one-time automatic actuation using onsite Class 1E battery-backed power without any further need of onsite or offsite electric power system operation.~~

~~These statements of compliance and proposed PDC 33 may be used as the bases for the necessary exemption during 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.~~GDC 33 applies to small leaks in the RCPB that are [[ ]] (e.g., leakage from flanges or cracks in piping or other components), and which do not exceed the capability of the nonsafety-related high-pressure CRD system used as normal reactor coolant makeup during power operations. The maximum allowed leakage rate for continuing power operation is stipulated in the plant Technical Specifications. For leakage greater than the maximum allowed leakage rate, automatic reactor scram and automatic actuation of the [[ ]] is not anticipated for most of these small leaks because the normal means of makeup from the high-pressure CRD system and feedwater maintains the level in the normal operating range. However, these small leaks which do not exceed the capability of the nonsafety-related high-pressure CRD system are evaluated using specified acceptable fuel design limits rather than the BWRX-300 acceptance criteria in response to a LOCA.

The high-pressure CRD system pumps can operate using either onsite electric power system operation (assuming offsite power is not available) backed up by nonsafety-related standby diesel generators and offsite electric power system operation (assuming onsite power is not available).

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 33.

eRAI No.: 9732
Date of eRAI Issue: 04/09/20

NRC Question NONE-2

10 CFR 50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, requires that "Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section." The criteria in paragraph (b) include peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling with associated limits included in the regulation. Moreover, each criterion is quantitatively defined.

Licensing Topical Report (LTR) NEDC-33910P, Revision 0, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," Section 2.8, states that:

> The primary design goal of the BWRX-300 in response to a LOCA is that the core does not uncover so that the following acceptance criteria are met:
>
> - No significant fuel cladding heat up shall occur.
> - No significant fuel cladding oxidization shall occur.
> - No significant fuel cladding hydrogen generation shall occur.
> - Long-term cooling shall remove decay heat and maintain core temperature to acceptably low values.

These criteria lack definition and, as such, are subject to interpretation. The staff requests that GEH clarify whether it intends is to use and more explicitly define the criteria currently outlined in Section 2.8 as acceptance criteria in future licensing of the BWRX-300 design, or if GEH intends to use the criteria contained and explicitly defined in 10 CFR 50.46(b).

GEH Response to NRC Question NONE-2

GEH understands that the design goals provided in NEDC-33910P are subject to interpretation. These design goals were not intended to imply that they were to be used instead of the specific criteria defined in 10 CFR 50.46(b). Instead, it was intended to specify that the acceptance criteria in 10 CFR 50.46(b) can be demonstrated by maintaining reactor water level above the top of active fuel (TAF) such that the core does not uncover following a LOCA. However, upon further evaluation, GEH has determined that the term "design goals" and "primary design goals" should be replaced with a more appropriate term (i.e., BWRX-300 acceptance criteria in response to a LOCA) which bound the acceptance criteria in 10 CFR 50.46(b). In addition, GEH has determined that these BWRX-300 acceptance criteria in response to a LOCA can include maintaining reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range.

As described in NEDC-33910P, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS in compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling. [[

]] The analyses to demonstrate compliance will be provided during future licensing activities.

Therefore, NEDC-33910P will be revised to include ~~a more explicit commitment to meeting~~BWRX-300 acceptance criteria in response to a LOCA including maintaining reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range which bound the specific acceptance criteria defined in 10 CFR 50.46(b).

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to include compliance with the requirements of 10 CFR 50.46(b) by use of BWRX-300 acceptance criteria in response to a LOCA including maintaining reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range which bound the acceptance criteria in 10 CFR 50.46(b):

. . .

1.1 Purpose

. . .

- Design requirements are specified for the RPV isolation valves and configuration with the function to close to limit the loss of coolant from large and medium pipe breaks, and design requirements are specified for automatic actuation of the Isolation Condenser System (ICS) to remove decay heat from large, medium, and small pipe breaks, to meet the BWRX-300 acceptance criteria in response to a Loss-Of-Coolant Accident (LOCA) which bound the acceptance criteria in 10 CFR 50.46(b)~~ following Loss-Of-Coolant Accidents (LOCAs)~~.

. . .

2.1 General Introduction

. . .

The relatively large RPV volume of the BWRX-300, along with the relatively tall chimney region, provides a substantial reservoir of water above the core. This ensures the ~~core remains~~

~~covered~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range following transients involving feedwater flow interruptions or LOCAs.

. . .

2.7 Categories of Pipe Breaks

. . .

[[

-
-
-
-
-
-
-
-

]]

. . .

Small leaks in the RCPB that are [[ ]] (e.g., leakage from flanges or cracks in piping or other components) do not exceed the capability of the nonsafety-related high-pressure CRD system used as normal reactor coolant makeup during power operations. The maximum allowed leakage rate for continuing power operation is stipulated in the plant Technical Specifications. For leakage greater than the maximum allowed leakage rate, automatic reactor scram and automatic actuation of the [[ ]] is not anticipated for most of these small leaks because the normal means of makeup from the high-pressure CRD system and feedwater maintains the level in the normal operating range. However, these small leaks which do not exceed the capability of the nonsafety-related high-pressure CRD system are evaluated using specified acceptable fuel design limits rather than the BWRX-300 acceptance criteria in response to a LOCA. See Section 4.1.11 for further discussion of these small leaks.

. . .

2.8 LOCA Acceptance Criteria

The ~~primary design goal of the~~ BWRX-300 acceptance criteria in response to a LOCA ~~is~~are that the ~~core does not uncover so that~~ reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range which bound the ~~following~~ acceptance criteria of 10 CFR 50.46(b)~~ are met~~. Maintaining reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range ensures that:

- No significant fuel cladding heat up ~~shall~~ occurs.
- No significant fuel cladding oxidization ~~shall~~ occurs.
- No significant fuel cladding hydrogen generation ~~shall~~ occurs.
- No significant changes in core geometry ~~shall~~ occurs.
- Long term cooling ~~shall~~to remove decay heat and maintain the core temperature to acceptably low values occurs.

~~These acceptance criteria can be demonstrated by maintaining reactor water level at TAF.~~

. . .

4.1.1 10 CFR 50.34(f)

. . .

The following requirements are evaluated as they are related to [[

]], as being required following the worst-case LOCA to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b):

. . .

4.1.2 10 CFR 50.46

. . .

- The required ECCS design functions of the [[

  ]] to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(4) as further described below. Another design function of the [[ ]] is for overpressure protection which is further required to maintain a coolable geometry to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance 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 BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria~~requirements~~ of 10 CFR 50.46(b)(5) and only requires operator action to [[ ]] after approximately seven days.

. . .

Based on the above evaluation, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in ~~full~~ compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b).

. . .

Although this is a~~n alternative and~~ non-traditional approach for the design of the ECCS for past LWRs, no [[ ]] is ~~required~~needed following the worst-case LOCA to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b). The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, ~~no exception or exemption is required for these regulatory requirements. Full compliance with these requirements is to be demonstrated in the final ECCS performance analyses to be completed during future licensing activities~~the BWRX-300 design will meet the requirements of 10 CFR 50.46.

. . .

Statement of Compliance: The ECCS evaluation model for the BWRX-300 is to be developed using previously approved methodologies used in the ECCS performance analyses for the ESBWR as modified using BWRX-300 specific design requirements and parameters. ~~Full compliance with these requirements is to be demonstrated in the final ECCS performance analyses to~~The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities using an NRC-approved BWRX-300 ECCS evaluation model which includes reasonably conservative methods. Because of the BWRX-300 acceptance criteria being applied to bound the 10 CFR 50.46(b) acceptance criteria, uncertainties will be addressed in the BWRX-300 ECCS evaluation model to verify that there is a high level of probability that the BWRX-300 acceptance criteria would not be exceeded rather than the 10 CFR 50.46(b) acceptance criteria. The BWRX-300 evaluation model will not use the alternatives provided in 10 CFR 50 Appendix K.

. . .

Statement of Compliance: The ~~design goal~~BWRX-300 acceptance criteria in response to a LOCA is that ~~the core does not uncover during LOCAs~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant fuel cladding heatup occurs, such that the calculated maximum fuel element cladding temperature does not exceed the acceptance criterion of 2200°F.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ~~ECCS performance analyses are to~~ The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(1).

. . .

Statement of Compliance: The ~~design goal~~BWRX-300 acceptance criteria in response to a LOCA is that ~~the core does not uncover during LOCAs~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant fuel cladding oxidization occurs, such that the calculated total oxidation of the cladding does not nowhere exceed the acceptance criterion of 0.17 times the total cladding thickness before oxidation.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ~~ECCS performance~~

~~analyses are to~~ The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(2).

. . .

Statement of Compliance: The ~~design goal~~ BWRX-300 acceptance criteria in response to a LOCA is that ~~the core does not uncover during LOCAs~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant fuel cladding hydrogen generation occurs, such that the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed the acceptance criterion of 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the finalFinal ECCS performance analyses are to~~The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(3).

. . .

Statement of Compliance: The ~~design goal~~ BWRX-300 acceptance criteria in response to a LOCA is that ~~the core does not uncover during LOCAs~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant changes in core geometry occur, such that the acceptance criterion of the core remaining amenable to cooling is met.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the finalFinal ECCS performance analyses are to~~The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(4).

. . .

Statement of Compliance: The ~~design goal~~BWRX-300 acceptance criteria in response to a LOCA is that ~~the core does not uncover during LOCAs~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that ~~no significant fuel cladding heatup~~long-term cooling removes decay heat and maintains the core temperature to acceptably low values ~~occurs~~, such that after any calculated successful initial operation of the ECCS, the acceptance criteria of the calculated core temperature being maintained at an acceptably low value and decay heat being removed for the extended period of time required by the long lived radioactivity remaining in the core are met.

. . .

The selected long-term cooling timeframe of [[ ]] is sufficient to maintain reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range for both [[

]] Following [[

]], the [[ ]] continues to provide long-term cooling to meet the BWRX-300 acceptance criteria in response to a LOCA and only requires operator action to [[ ]] after approximately seven days.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the final~~Final ~~ECCS performance analyses are to~~ The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(5).

. . .

~~4.1.10~~4.1.11 10 CFR 50 Appendix A, GDC 33

. . .

Statement of Compliance: ~~The safety analysis assumes that the small pipe breaks [[~~

~~]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time actuation using onsite Class 1E battery-backed DC power.~~

~~Although the BWRX-300 [[ ]] nonsafety-related injection systems may be used for manual addition of reactor coolant inventory by the operator using high pressure CRD injection or by reestablishing feedwater injection. These nonsafety-related injection systems can be used at any time following a LOCA. The timing of such operator actions is to be determined during the final ECCS performance analyses to be completed during future licensing activities. Therefore, specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary [[~~

~~]]~~GDC 33 applies to small leaks in the RCPB that are [[ ]] (e.g., leakage from flanges or cracks in piping or other components), and which do not exceed the capability of the nonsafety-related high-pressure CRD system used as normal reactor coolant makeup during power operations. The maximum allowed leakage rate for continuing power operation is stipulated in the plant Technical Specifications. For leakage greater than the maximum allowed leakage rate, automatic reactor scram and automatic actuation of the [[ ]] is not anticipated for most of these small leaks because the normal means of makeup from the high-pressure CRD system and feedwater maintains the level in the normal operating range. However, these small leaks which do not exceed the capability of the nonsafety-related high-pressure CRD system are evaluated using specified acceptable fuel design limits rather than the BWRX-300 acceptance criteria in response to a LOCA.

The high-pressure CRD system pumps can operate using either onsite electric power system operation (assuming offsite power is not available) backed up by nonsafety-related standby diesel generators and offsite electric power system operation (assuming onsite power is not available).

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 33.

. . .

~~4.1.11~~4.1.12 10 CFR 50 Appendix A, GDC 35

. . .

Statement of Compliance: As previously described, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in ~~full~~ compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS in ~~full~~ compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling, which is assured for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS through the use of [[ ]].

~~A design goal of the BWRX-300 is~~The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, such that the performance of the [[ ]] is sufficient to ensure that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

[[

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~~Full compliance with these requirements is to be demonstrated in the final ECCS performance~~The analyses to ~~be completed~~demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA (i.e., reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range) will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 35.

. . .

4.1.13 10 CFR 50 Appendix A, GDC 37

. . .

Statement of Compliance: As previously described, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b).

. . .

~~4.3.3~~4.3.4 Standard Review Plan 6.3

. . .

The design functions of the [[ ]] to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(5). The ~~primary design goal of the~~BWRX-300 acceptance criteria in response to a LOCA is that ~~the core does not uncover so that these acceptance criteria are met~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range. These BWRX-300 acceptance criteria ensure the following:

6. No significant fuel cladding heatup occurs in the short-term.
7. No significant fuel cladding oxidization occurs.
8. No significant fuel cladding hydrogen generation occurs.
9. No significant changes in core geometry occur.
10. No significant fuel cladding heatup occurs in the long-term.

Although this is a~~n alternative and~~ non-traditional approach for the design of the ECCS for past LWRs, no active or passive injection of additional water inventory is required following the worst-case LOCA to meet these BWRX-300 acceptance criteria and to meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 14, GDC 30, GDC 31, and GDC 35.

. . .

~~4.3.4~~4.3.5 Standard Review Plan 15.6.5

. . .

The design functions of the [[

]] to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(5).

eRAI No.: 9732
Date of eRAI Issue: 04/09/20

NRC Question NONE-3

10 CFR 50.46(b)(5), requires the calculated core temperature to be maintained at an acceptably low value and decay heat to be removed for the extended period of time required by the long-lived radioactivity remaining in the core. 10 CFR 50, Appendix A, GDC 35 requires a system to provide abundant emergency core cooling. In addition, the 2008 Advanced Reactor Policy Statement (73 FRN 60612) describes the Commission's expectation for new reactors' safety systems to provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety functions (e.g. longer time constants and reduced operator actions).

One of the principle design requirements of the Electric Power Research Institute's (EPRI) Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) for passive designs is that the core must be cooled and containment integrity maintained with only safety-related SSCs and without reliance on ac power and operator actions for a minimum of 72 hours. The staff concludes in its corresponding safety evaluation (NUREG-1242, Vol. 3, Pt.1) that these requirements relative to the 72-hour capability are acceptable and are consistent with Commission policies related to passive designs.

Licensing Topical Report NEDC-33910P, specifies that adequate core cooling is maintained [[ ]], and the [[ ]] will be used to ensure full compliance with 10 CFR 50.46(b)(5), and GDC 35. However, it is unclear to the staff what long-term cooling timeframe will be established for the BWRX-300, and when water injection from nonsafety-related sources will be used by the operators after the design-basis long-term cooling period. Therefore, the staff requests GEH to define the design-basis long-term cooling timeframe for the BWRX-300 (e.g. 72 hours, 7 days, etc.) or ensure Licensing Topical Report NEDC-33910P clearly describes where it will be addressed in future submittals.

GEH Response to NRC Question NONE-3

GEH understands that the design-basis long-term cooling timeframe for the BWRX-300 was not clearly provided in NEDC-33910P. Instead, although not explicitly defined, GEH intended for the final ECCS performance analyses to provide additional insight for when water injection from nonsafety-related sources will be used by the operators after the design-basis long-term cooling period. However, GEH has completed some preliminary evaluations to determine a conservative long-term cooling period for the BWRX-300.

For the BWRX-300, [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed power and then remains in service without any further need of onsite or offsite electric power system operation.

The selected long-term cooling timeframe of [[ ]] is sufficient to maintain reactor water level at or above TAF or fuel cladding temperature within normal operating temperature for both [[

]] Following ~~the worst-case postulated LOCA (i.e.,~~ [[

]]), 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.

Proposed Changes to NEDC-33910P, Revision 0

NEDC-33910P, Revision 0, will be revised to reflect [[

]] and to include compliance with the requirements of 10 CFR 50.46(b)(5) using a long-term cooling timeframe of [[ ]]:

. . .

2.1 General Introduction

. . .

[[

]] These design features preserve reactor coolant inventory to ensure that adequate core cooling is maintained.

. . .

2.2 General Overview of the Reactor Pressure Vessel Isolation Concept

. . .

[[

]] The main steam line nozzles are placed as high as possible on the RPV. The total number of nozzles are reduced from the ESBWR to the BWRX 300. [[

]]

. . .

2.4 Reactor Pressure Vessel Nozzle Design Requirements

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[[

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[[

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. . .

2.7 Categories of Pipe Breaks

Steam and liquid line breaks are ~~considered~~evaluated. The pipe breaks ~~considered~~evaluated in the safety analysis are divided into two size categories:

- [[

The largest steam line break is a main steam line break. The largest liquid line break is the feedwater line break. [[

]]

Design Requirements:

- [[

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  -
  -

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[[

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[[

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Table 2-1 summarizes the ~~LOCA cases~~pipe break categories and the key assumptions for each case.

Table 2-1: Pipe Break Categories

Break Type ~~(Steam or liquid)~~	[[ ]] Small Breaks	[[ ]] Large Breaks
Steam	[[ ]]	[[ ]]
Liquid	[[ ]]	[[ ]]

. . .

4.1.2 10 CFR 50.46

. . .

Statement of Compliance: The design features of the BWRX-300 used to comply with this requirement include [[ ]] for all postulated LOCA break sizes, in conjunction with the [[ ]] for postulated LOCA break sizes [[ ]].

. . .

o [[ ]] for any breaks that would result in a loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system, which for the BWRX-300 includes LOCA break sizes [[ ]].

. . .

- Regulatory Requirement: 10 CFR 50.46(b)(5), Long-term cooling, requires that after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

  Statement of Compliance: The ~~design goal~~BWRX-300 acceptance criteria in response to a LOCA is that ~~the core does not uncover during LOCAs~~reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range~~The design goal is that the core does not uncover during LOCAs~~, and that ~~no significant fuel cladding heatup~~long-term cooling ~~to~~ removes decay heat and maintains the core temperature to acceptably low values ~~occurs~~, such that after any calculated successful initial operation of the ECCS, the acceptance criteria of the calculated core temperature being maintained at an acceptably low value and decay heat being removed for the extended period of time required by the long lived radioactivity remaining in the core are met.

  For the BWRX-300, [[

  ]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed ~~DC~~ power and then remains in service for at least [[ ]] without any further need of onsite or offsite electric power system operation.

  The selected long-term cooling timeframe of [[ ]] is sufficient to maintain reactor water level at or above TAF or fuel cladding temperature within normal operating temperature for both [[

  ]] Following ~~the worst-case postulated LOCA (i.e., [[~~

  ]], the [[ ]] continues to provide long-term cooling to meet the ~~requirements of 10 CFR 50.46(b)(5)~~ BWRX-300 acceptance criteria in response to a LOCA and only requires operator action to [[ ]] after approximately seven days.

~~Therefore, no alternative approach, exception, or exemption from these requirements is required. Full compliance with this requirement is to be demonstrated in the~~ ~~finalFinal ECCS performance analyses are to~~The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX 300 design will meet the requirements of 10 CFR 50.46(b)(5).

. . .

~~4.1.11~~4.1.12 10 CFR 50 Appendix A, GDC 35

Statement of Compliance: As previously described, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in ~~full~~ compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS in ~~full~~ compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling, which is assured for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS through the use of the [[ ]].

Appendix C

Replaced Pages from NEDC-33910P 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 9730, Question 03.09.06-1, revised Section 1.1 and added new Sections 4.1.5 and 4.1.13 to address compliance with 10 CFR 50, Appendix A, GDC 2 and GDC 37, and to address addition of Generic Letter 95-07 as new Section 4.5.2 • NRC eRAI 9730, Question 03.09.06-1, supplemental response revised new Section 4.1.5 to address [[ ]]. • NRC eRAI 9730, Question 03.09.06-3, replaced the use of terms such as "consideration" and "considered" with appropriate terms in Sections 2.4, 2.4.1, 2.5, 2.6, 2.7, renumbered 4.1.6, and renumbered 4.3.2. • NRC eRAI 9730, Question 03.09.06-4, added the design requirements for the use of positive mechanical means in the design of the valve actuators to maintain these valves in their required post-accident valve positions in Sections 2.5 and 3.1.2. • NRC eRAI 9730, Question 03.09.06-5, revised Section 4.1.1 to describe requirements for 10 CFR 50.34(f)(2)(x) as not technically relevant rather than not required, but to also indicate that the [[ ]] will be qualified by testing to perform their safety-related design function for fluid conditions expected under operating conditions, transients and accidents including ATWS events. • NRC eRAI 9730, Questions 03.09.06-6, 03.09.06-7, and 03.09.06-14, clarified statements that no alternative approach, exception, or exemption from certain regulatory requirements is required providing a commitment to meeting the applicable regulatory requirements during detailed design in Sections 2.7, 4.1.2, 4.1.3, 4.1.4, renumbered 4.1.6, renumbered 4.1.7, renumbered 4.1.8, renumbered 4.1.9, renumbered 4.1.10, renumbered 4.1.12, 4.2.1, 4.2.2, 4.2.3, and 4.2.4.

Revision Number	Description of Change
Supplement 1 (continued)	Revised to incorporate the following responses to NRC Requests for Additional Information (eRAIs): • NRC eRAI 9730, Question 03.09.06-6 supplemental response, revised Section 4.1.2 to reflect [[ ]] and to include compliance with the requirements of the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(5) using a long-term cooling timeframe of [[ ]]. • NRC eRAI 9730, Question 03.09.06-8, revised Section renumbered 4.1.11 to address compliance with 10 CFR 50, Appendix A, GDC 33. • NRC eRAI 9730, Question 03.09.06-9, added new Section 4.1.5 to address compliance with 10 CFR 50, Appendix A, GDC 2. • NRC eRAI 9730, Question 03.09.06-11, added new Sections 4.2.5 and 4.2.6 to address conformance with the regulatory guidance of RG 1.147 and RG 1.192. • NRC eRAI 9730, Question 03.09.06-12, added new Section 4.3.1 to address conformance with the regulatory guidance of SRP 3.9.6. • NRC eRAI 9730, Question 03.09.06-13, revised Sections 4.4 and 4.5 identifying the limited scope of the evaluation of generic issues, and operational experience and generic communications, respectively, and committing to the up-to-date evaluation of these issues to be provided during future licensing activities. • NRC eRAI 9731, Question 03.06.02-3, and supplemental response, revised Section 2.4.1 to address material of construction for the [[ ]] which will be determined during detailed design and addressed in future licensing activities.

Revision Number	Description of Change
Supplement 1 (continued)	Revised to incorporate the following responses to NRC Requests for Additional Information (eRAIs): • NRC eRAI 9731, Question 03.06.02-1 supplemental response, revised Section 2.4.1 to describe the [[ ]] consistent with the discussions in NEDC-33911P, BWRX-300 Containment Performance. • NRC eRAI 9732, Question NONE-1, revised Section renumbered 4.1.11 to address compliance with 10 CFR 50, Appendix A, GDC 33. • NRC eRAI 9732, Question NONE-2, and supplemental response, revised Sections 1.1., 2.1, 2.7, 2.8, 4.1.1, 4.1.2, renumbered 4.1.11, renumbered 4.1.12, new 4.1.13, renumbered 4.3.4, and renumbered 4.3.5 to include compliance with the requirements of 10 CFR 50.46(b) by use of BWRX-300 acceptance criteria in response to a LOCA including maintaining reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range, which bound the acceptance criteria in 10 CFR 50.46(b). • NRC eRAI 9732, Question NONE-3, and supplemental response, revised Sections 2.1, 2.2, 2.4, 2.7, 4.1.2, and renumbered 4.1.12, and Table 2-1, to reflect [[ ]] and to include compliance with the requirements of 10 CFR 50.46(b)(5) using a long-term cooling timeframe of [[ ]].

Acronyms and Abbreviations

Term	Definition
ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ADS	Automatic Depressurization System
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
B&PV	Boiler & Pressure Vessel
BTP	Branch Technical Position
BWR	Boiling Water Reactor
COL	Combined Operating License
CP	Construction Permit
CRD	Control Rod Drive
DCA	Design Certification Application
DCD	Design Control Document
DPV	Depressurization Valve
ECCS	Emergency Core Cooling System
ESBWR	Economically Simplified Boiling Water Reactor
EQ	Environmental Qualification
FMCRD	Fine Motion Control Rod Drive
GDC	General Design Criteria
GEH	GE Hitachi Nuclear Energy
HGNE	Hitachi-GE Nuclear Energy Ltd.
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
I&C	Instrumentation and Control
IC	Isolation Condenser
ICS	Isolation Condenser System
ISI	Inservice Inspection
IST	Inservice Testing

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 Reactor Pressure Vessel (RPV) isolation and overpressure protection design functions, specifically for the following areas:

- Design requirements are specified for the RPV isolation valves and configuration with the function to close to limit the loss of coolant from large and medium pipe breaks, and design requirements are specified for automatic actuation of the Isolation Condenser System (ICS) to remove decay heat from large, medium, and small pipe breaks, to meet the BWRX-300 acceptance criteria in response to a Loss-Of-Coolant Accident (LOCA) which bound the acceptance criteria in 10 CFR 50.46(b). [[ ]] The design of the RPV isolation valves and ICS meet the requirements of 10 CFR 50.46(b) and 10 CFR 50 Appendix A, General Design Criteria, GDC 1, GDC 2, GDC 4, GDC 14, GDC 30, GDC 31, GDC 33, GDC 35, and GDC 37.
- Design requirements are specified for the Reactor Protection System (RPS) and ICS for overpressure protection. [[ ]] The design of the RPS and ICS meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 15, GDC 30, and GDC 31.

1.2 Scope

The scope of this report includes the following:

- A technical evaluation of the BWRX-300 RPV isolation and overpressure protection design features and design functions, 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 RPV isolation and overpressure protection design features and design functions to describe compliance with regulatory requirements and to describe the bases for any exemptions to regulatory requirements or approaches to regulatory guidance that may be referenced during 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 EVALUATION OF RPV ISOLATION

2.1 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 basic BWRX-300 safety design philosophy for the mitigation of LOCAs is built on utilization of inherent margins (e.g., larger water inventory) to eliminate system challenges, reduce number and size of RPV nozzles as compared to predecessor designs[[ ]]. The relatively large RPV volume of the BWRX-300, along with the relatively tall chimney region, provides a substantial reservoir of water above the core. This ensures the reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range following transients involving feedwater flow interruptions or LOCAs. [[ ]] These design features preserve reactor coolant inventory to ensure that adequate core cooling is maintained.

The large RPV volume also reduces the rate at which reactor pressurization occurs if the reactor is suddenly isolated from its normal heat sink. If isolation should occur, RPS is initiated to shut down the reactor and ICS is initiated to remove heat from the reactor. Heat from the reactor is rejected to the Isolation Condenser (IC) heat exchangers located within separate, large pools of water (the IC pools) positioned immediately above (and outside) the containment. [[ ]]

2.1.1 Reactor Pressure Vessel

The BWRX-300 RPV assembly consists of the pressure vessel, removable head, and its appurtenances, supports and insulation, and the reactor internals. The RPV instrumentation to monitor the conditions within the RPV is designed to cover the full range of reactor power operation. The RPV, together with its internals, provides guidance and support for the Fine Motion Control Rod Drives (FMCRDs).

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.

The ICS is initiated automatically on high RPV pressure indicating an overpressure event or on signals indicating a LOCA. To start an IC train, the IC condensate return valve is opened whereupon the standing condensate drains into the reactor and the steam water interface in the IC tube bundle moves downward below the lower headers. [[

]] The IC pools are interconnected and 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.

2.2 General Overview of the Reactor Pressure Vessel Isolation Concept

[[

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[[

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Figure 2-3: RPV Isolation Valve Assembly (Example)

One of the design objectives of the BWRX-300 Reactor Coolant Pressure Boundary (RCPB) is to minimize the risks associated with LOCAs relative to the ESBWR design. Risk is minimized by the following:

- Reducing the number of nozzles,
- Reducing pipe lengths and nominal pipe diameters,
- Maximizing the elevation of the nozzles, and
- [[

]]

For the BWRX-300 the RPV nozzles are placed as high on the RPV as possible to limit the effect of a potential pipe break. [[

]] The main steam line nozzles are placed as high as possible on the RPV. The total number of nozzles are reduced from the ESBWR to the BWRX-300. [[

]]

Figure 2-4 shows a representation of the preliminary RPV assembly arrangement for the BWRX-300 and summarizes the relative locations of nozzles on the RPV assembly.

2.4 Reactor Pressure Vessel Nozzle Design Requirements

The RPV nozzle design requirements for the BWRX-300 use the same design codes and standards, except for issue date, as the ESBWR, which are documented in the ESBWR DCD, Tier 2, Subsection 5.3.3.2.2, Reactor Vessel Design Data, for Reactor Vessel Nozzles [Reference 5.2].

There are some differences in the RPV nozzle designs between the BWRX-300 and the ESBWR. [[

]]

Design Requirements:

- All piping and valves connected to the nozzles shall be designed not to exceed the allowable loads on any nozzle.
- The feedwater inlet nozzles and IC condensate return nozzles shall be designed to account for stresses caused by cooler injection water.
- All nozzles shall be low alloy steel forgings; except the water level instrumentation nozzles.
- The design of the nozzles shall be in accordance with ASME Section III, Subsection NB and meet the applicable requirements of the vessel design documents.
- [[

]]

2.4.1 Connection of Reactor Pressure Vessel Isolation Valves to Reactor Vessel

[[

]] The BWRX-300 design requirements for identifying postulated pipe rupture locations and configurations inside containment conform to the guidance in Branch Technical Position (BTP) 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment." However, [[

]], most of the BTP 3-4, Part B, Item 1(ii) criteria do not apply. However, BTP 3-4, Part B. Item 1(ii) criteria generically involving design stress and fatigue limits and inservice inspection (ISI) guidelines are applicable.

[[

]] These [[

]] will be established during detailed design of the [[ ]] and provided during future licensing activities.

BTP 3-4, Part B, Item 1(ii)(1) specifies more conservative stress and fatigue limits for ASME Class 1 piping in containment penetration areas than those required for piping by ASME Code Section III, Paragraph NB-3653. The bases for these more conservative limits include limiting the stresses resulting from service loads (excluding those due to peak stresses) to within the material yield strength (i.e., elastic strains), and to ensure that the cumulative usage factor calculation account for the possibility of a faulty design or improperly controlled fabrication, installation errors, and unexpected modes of operation, vibration, and other structural degradation mechanisms.

[[

]] the design criteria given in ASME Code Section III, Paragraph NB-3230, provides greater margin against yielding due to service loads than do the rules of Paragraph NB-3653 for typical piping system materials, even when using the more restrictive limits of BTP 3-4, Part B, Item 1(ii)(1). Therefore, the imposition of more conservative stress limits of BTP 3-4, Part B, Item 1(ii)(1) is not justified.

[[

]]

[[

]] extending to the containment wall, the BWRX-300 design requirements include identifying postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, Part B, Item 1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2).

Where break locations are selected without the benefit of stress calculations, breaks are postulated at the piping welds to each fitting, valve, or welded attachment. Additionally, for a structure credited with separating a high-energy line from essential structures, systems, and components (SSCs), the separating structure is designed in accordance with BTP 3-4, Part B, Item 1(iii)(4) to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect on the structure. This is true even though the criteria described in BTP 3-4, Part B, Items 1(iii)(1) through (3) might not require the postulation of a break at that location.

2.5 Reactor Pressure Vessel Isolation Valve Design Requirements

GEH applies the following key factors to the selection of valves for the BWRX-300 RPV isolation valves design:

- The valve closure is a safety-related function.
- Compact valve and actuator assemblies are selected.
- Required Primary Containment Vessel (PCV) space allocation is minimized in proximity of the RPV.
- Electrical and digital controls are minimized inside the PCV.

Design Requirement:

- All BWRX-300 RPV isolation valves shall have a proven low leakage potential.

Design and administrative leakage limits are applied to valve selection during the BWRX-300 preliminary design and are based on plant design and event evaluations using offsite dose consequences compared to regulatory limits as well as containment design limits. The leakage criteria are analyzed as part of the plant safety analysis.

For BWRX-300 design, the application of motor-operated valves is constrained because there is no safety-related power supply other than limited Class 1E battery-stored power. Thus, motor-operated valves other than fail-closed magnetic-motor valves (i.e., solenoid operated valves) are not used for any RPV isolation valve applications.

Design Requirements:

- The RPV isolation valves for main steam line, feedwater, shutdown cooling, and reactor water cleanup shall fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means.
- [[
  ]] with valve actuators designed to maintain the valves in their as-is position by positive mechanical means.

A critical aspect of the valve and actuator selection to evaluate is the failure mode. The failure mode of the RPV isolation valves are determined based on the safety function of the connected system. [[

]]

The actuation signal for the RPV isolation valve closure is different for fail-close and fail-as-is isolation valves.

Design Requirements:

- The fail-close RPV isolation valves shall automatically close on high containment pressure indicating a LOCA.
- [[
  ]]

The ICS RPV isolation valves automatic isolation function uses the same logic and functionality as the ESBWR ICS containment isolation valves, which is described in the last two paragraphs of

ESBWR DCD Tier 2 Subsection 5.4.6.2.2 [Reference 5.2]. However, the BWRX-300 Instrumentation and Control (I&C) system has three divisions of safety-related I&C. [[

]]

2.6 Reactor Pressure Vessel Isolation Valve Actuator Design Requirements

Design Requirement:

- The RPV isolation valves and actuators shall be operable during events when the containment pressure and temperature are elevated.

A key design requirement is control of the temperature at the valve-actuator interface in order to limit thermal effects on the actuator assembly. The RPV isolation valves are heated by process water or steam, which also elevates valve actuator temperatures above the local ambient. Valve and valve actuator designs are qualified in accordance with ASME QME-1 [Reference 5.5] to include evaluation of the local environmental conditions, including evaluation of the effects of heat transfer from the process water or steam and Design Basis Events. [[ ]] The stem connection and actuator mounting method are studied to determine if thermal isolation needs to be implemented. High-temperature seals or lubricants are used for the actuators.

Design Requirement:

- Control devices (e.g., pilots) that rely on electric power may be located outside the PCV when practical.

Locating the control devices for the RPV isolation valve actuators that rely on electric power outside the PCV eliminates harsh Environmental Qualification (EQ) requirements.

2.7 Categories of Pipe Breaks

Steam and liquid line breaks are evaluated. The pipe breaks evaluated in the safety analysis are divided into two size categories:

- [[
-]]

The largest steam line break is a main steam line break. The largest liquid line break is the feedwater line break. [[

]]

Design Requirements:

- [[

-

]]

For a postulated pipe break, the RPS performs the control of reactivity function by shutting down the core. [[

]]

The emergency core cooling system (ECCS) evaluation model for the BWRX-300 is to be developed using previously approved methodologies used in the ECCS performance analyses for the ESBWR as modified using BWRX-300 specific design requirements and parameters. Final ECCS performance analyses are to be completed during future licensing activities using an NRC-approved BWRX-300 ECCS evaluation model. Methodology for containment response is described in LTR NEDC-33911P, BWRX-300 Containment Performance [Reference 5.6].

Table 2-1 summarizes the pipe break categories and the key assumptions for each case.

Table 2-1: Pipe Break Categories

Break Type	[[ ]] Small Breaks	[[ ]] Large Breaks
Steam	[[ ]]	[[ ]]
Liquid	[[ ]]	[[ ]]

Small leaks in the RCPB that are [[
]] (e.g., leakage from flanges or cracks in piping or other components) do not exceed the capability of the nonsafety-related high-pressure CRD system used as normal reactor coolant

makeup during power operations. The maximum allowed leakage rate for continuing power operation is stipulated in the plant Technical Specifications. For leakage greater than the maximum allowed leakage rate, automatic reactor scram and automatic actuation of the [[ ]] is not anticipated for most of these small leaks because the normal means of makeup from the high-pressure CRD system and feedwater maintains the level in the normal operating range. However, these small leaks which do not exceed the capability of the nonsafety-related high-pressure CRD system are evaluated using specified acceptable fuel design limits rather than the BWRX-300 acceptance criteria in response to a LOCA. See Subsection 4.1.11 for further discussion of these small leaks.

2.8 LOCA Acceptance Criteria

The BWRX-300 acceptance criteria in response to a LOCA are that the reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, which bound the acceptance criteria of 10 CFR 50.46(b). Maintaining reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range ensures that:

- No significant fuel cladding heatup occurs.
- No significant fuel cladding oxidization occurs.
- No significant fuel cladding hydrogen generation occurs.
- No significant changes in core geometry occurs.
- Long term cooling to remove decay heat and maintain the core temperature to acceptably low values occurs.

3.0 TECHNICAL EVALUATION OF OVERPRESSURE PROTECTION

3.1 General Overview of the Overpressure Protection Concept

The BWRX-300 integrated overpressure protection during operation at power is ensured by application of the RPS to shut down the reactor [[
]]

As with other BWRs, the BWRX-300 does not operate in water-solid conditions and therefore is not subject to low-temperature operation requiring special overpressure protection. Additionally, for periodic leak testing while shutdown, the system is not subject to pressurization from the reactor, and special test conditions are established to allow for pressure control.

3.1.1 Reactor Protection System Design Requirements

The BWRX-300 RPS is based on the ESBWR RPS design [Reference 5.7]. The safety-related RPS performs the control of reactivity function for overpressure protection by initiating an automatic reactor shutdown by rapid insertion of control rods (scram) if monitored system variables exceed pre-established limits. This action prevents fuel damage and limits system pressure, thus aiding in the containment of radioactive materials function for overpressure protection.

The RPS implements the reactor trip functions. The RPS is the overall collection of instrument channels, trip logics, trip actuators, manual controls, and scram logic circuitry that initiates rapid insertion of control rods to shut down the reactor to help ensure established safety criteria are met.

The RPS is based on a fail-safe design philosophy. The RPS design provides reliable, single failure proof capability to automatically or manually initiate a reactor scram while maintaining protection against unnecessary scrams resulting from single failures. This is accomplished through the combination of fail-safe and fault-tolerant equipment design, and a two-out-of-three voting logic algorithm.

Design Requirements:

- RPS shall shutdown the reactor to ensure overpressure protection design requirements are met.
- RPS scram signals shall be established to ensure overpressure protection design requirements are met.
- RPS trip function performance shall be established to ensure overpressure protection design requirements are met.
- RPS functions to ensure overpressure protection design requirements are met shall be single failure proof.

3.1.2 Isolation Condenser System Design Requirements

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 and with a minimal loss of reactor coolant following reactor shutdown and isolation. [[

]] These functions aid in the containment of radioactive materials function for overpressure protection.

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. [[
]] with valve actuators designed to maintain the valves in their open position by positive mechanical means.

Design Requirements:

- [[
-
-
-
]]

3.2 ASME Requirements for Overpressure Protection

Overpressure protection for the RCPB is in compliance with ASME B&PV Code, Section III, Article NB-7000 [Reference 5.1]. Paragraph NB-7120 requires that overpressure protection of the components shall be provided by any of the following as an integrated overpressure protection:

a. The use of pressure relief devices and associated pressure sensing elements

b. The use of the reactor shutdown system

c. A design without pressure relief devices such that for each component in the protected system

   1. The overpressure does not exceed 1.1 times design pressure for the expected system pressure transient condition

   2. The calculated stress intensity and other design limitations for Service Level C are not exceeded for the unexpected system excess pressure transient condition

Overpressure protection for the BWRX-300 is provided in accordance with ASME B&PV Code, Section III, subparagraphs [[ ]].

4.0 REGULATORY EVALUATION

4.1 10 CFR 50 Regulations

4.1.1 10 CFR 50.34(f)

10 CFR 50.34(f), Additional Three Mile Island (TMI) related requirements, requires that each applicant for a design certification, design approval, combined license, or manufacturing license under Part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section, except for paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Although it is not yet determined whether a 10 CFR 52 license application may be submitted for a BWRX-300, these requirements are evaluated herein. 10 CFR 50.34(f)(1) states that to satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of these studies are factored into the final design of the facility, and that the studies must be submitted as part of the final safety analysis report. 10 CFR 50.34(f)(2) states that to satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. The following requirements are evaluated as they are related to [[

]], as being required following the worst-case LOCA to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b):

- Regulatory Requirement: 10 CFR 50.34(f)(1)(v) requires that an evaluation be performed of the safety effectiveness of providing for separation of High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with High Pressure Core Spray (HPCS) systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (Applicable to BWR's only). (II.K.3.13)

  Statement of Compliance: The BWRX-300 does not include safety-related high-pressure injection systems, including RCIC, HPCI, and HPCS systems, because [[

  ]] Therefore, this requirement is not technically relevant to the BWRX-300.

- Regulatory Requirement: 10 CFR 50.34(f)(1)(vi) requires that a study be performed to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWR's only). (II.K.3.16)

Statement of Compliance: The design features of the BWRX-300 RCPB include the use of the RPS and [[ ]] for overpressure protection, [[

]]. The intent of this requirement is to minimize the potential for loss of reactor coolant through inadvertent operation of relief and safety valves, [[

]] Therefore, this requirement is not technically relevant to the BWRX-300.

- Regulatory Requirement: 10 CFR 50.34(f)(1)(vii) requires that a feasibility and risk assessment study be performed to determine the optimum ADS design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWR's only). (II.K.3.18)

  Statement of Compliance: Automatic actuation of [[
  ]] performs the function to mitigate the effects of a LOCA. [[ ]] is a one-time action to open or close appropriate valves during accident response which may also be initiated manually as a one-time action if necessary. However, manual actuation is not assumed in the accident analysis. Therefore, this requirement is not technically relevant to the BWRX-300.

- Regulatory Requirement: 10 CFR 50.34(f)(1)(viii) requires that a study be performed of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (Applicable to BWR's only). (II.K.3.21)

  Statement of Compliance: The BWRX-300 does not include core spray and low pressure coolant injection systems, because automatic actuation of [[
  ]] are sufficient to mitigate the effects of a LOCA. [[ ]] is a one-time action to open or close appropriate valves during accident response and does not require the use of active pumps. Therefore, this this requirement is not technically relevant to the BWRX-300.

- Regulatory Requirement: 10 CFR 50.34(f)(1)(ix) requires that a study be performed to determine the need for additional space cooling to ensure reliable long-term operation of the RCIC and HPCI systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, “high pressure core spray” for “high pressure coolant injection” and “HPCS” for “HPCI”) (Applicable to BWR's only). (II.K.3.24)

  Statement of Compliance: The BWRX-300 does not include RCIC and HPCI systems, because automatic actuation of [[
  ]] are sufficient to mitigate the effects of a LOCA. [[
  ]] is a one-time action to open or close appropriate valves located inside the PCV during accident response which are environmentally qualified to operate under post-accident conditions, and does not require the use of active pumps or other equipment requiring space cooling. Therefore, this requirement is not technically relevant to the BWRX-300.

- Regulatory Requirement: 10 CFR 50.34(f)(1)(x) requires that a study be performed to ensure that the ADS, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for nonsafety-related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWR's only). (II.K.3.28)

  Statement of Compliance: The BWRX-300 does not include [[ ]], because automatic actuation of [[ ]] are sufficient [[ ]] to mitigate the effects of a LOCA. [[ ]] is a one-time action to open or close appropriate valves located inside the PCV during accident response taking no credit for nonsafety-related equipment or instrumentation. The valves and actuators are environmentally qualified to operate under post-accident conditions. Therefore, this requirement is not technically relevant to the BWRX-300.

- Regulatory Requirement: 10 CFR 50.34(f)(2)(x) requires that a test program and associated model development be provided and tests conducted to qualify RCS relief and safety valves and, for PWRs, Power-Operated Relief Valve (PORV) block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of ATWS conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. (II.D.1)

  Statement of Compliance: The design features of the BWRX-300 RCPB include the use of the RPS [[ ]] for overpressure protection, [[ ]]. Because the [[ ]], this requirement is not technically relevant to the BWRX-300. However, the [[ ]] perform a similar function during an ATWS event in providing overpressure protection and will be qualified by testing to perform their safety-related design function for fluid conditions expected under operating conditions, transients and accidents including ATWS events.

- Regulatory Requirement: 10 CFR 50.34(f)(2)(xi) requires that direct indication of relief and safety valve position (open or closed) be provided in the control room. (II.D.3)

  Statement of Compliance: The design features of the BWRX-300 RCPB include the use of the RPS and [[ ]] for overpressure protection, [[ ]]. The intent of this requirement is to provide indication to the operator [[ ]]. Therefore, direct indication of relief and safety valve position (open or closed) provided in the control room is not technically relevant to the BWRX-300. However, direct position indication of the [[ ]] are provided in the BWRX-300 design.

Based on the above discussions, these requirements are not technically relevant to the BWRX-300. These statements of compliance may be used as the bases for this conclusion during 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.

4.1.2 10 CFR 50.46

10 CFR 50.46, Acceptance criteria for Emergency Core Cooling Systems (ECCS) for light-water nuclear power reactors, includes the following requirements:

- Regulatory Requirement: 10 CFR 50.46(a)(1)(i) requires that each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy or ZIRLO cladding must be provided with an emergency core cooling system that must be designed so that its calculated cooling performance following postulated LOCAs conforms to the criteria set forth in paragraph (b) of this section. As further defined in 10 CFR 50.46(c)(1), LOCAs are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

  Statement of Compliance: The design features of the BWRX-300 used to comply with this requirement include [[ ]] for all postulated LOCA break sizes, in conjunction with the [[ ]] for postulated LOCA break sizes [[ ]].

  - The required ECCS design functions of the [[ ]] to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(4) as further described below. Another design function of the [[ ]] is for overpressure protection which is further required to maintain a coolable geometry to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance 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 BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria of 10 CFR 50.46(b)(5) and only requires operator action to [[ ]] after approximately seven days.

  - [[ ]] for any breaks

that would result in a loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system, which for the BWRX-300 includes LOCA break sizes [[ ]]. The worst-case single failure of [[ ]] or the worst-case single failure affecting the [[ ]] does not prevent fulfillment of the required ECCS design functions. The [[ ]] are to be determined during the final ECCS performance analyses to be completed during future licensing activities.

Based on the above evaluation, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS consistent with the definition of a LOCA in 10 CFR 50.46(c)(1).

Although this is a non-traditional approach for the design of the ECCS for past LWRs, no [[ ]] is needed following the worst-case LOCA to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b). The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46.

- Regulatory Requirement: 10 CFR 50.46(a)(1)(i) requires that ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model.

  As further required in 10 CFR 50.46(a)(1)(ii), alternatively an ECCS evaluation model may be developed in conformance with the required and acceptable features of Appendix K ECCS Evaluation Models. 10 CFR 50.46(a)(2) defines an evaluation model as the calculational framework for evaluating the behavior of the reactor system during a postulated LOCA. It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

Statement of Compliance: The ECCS evaluation model for the BWRX-300 is to be developed using previously approved methodologies used in the ECCS performance analyses for the ESBWR as modified using BWRX-300 specific design requirements and parameters. The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities using an NRC-approved BWRX-300 ECCS evaluation model which includes reasonably conservative methods. Because of the BWRX-300 acceptance criteria being applied to bound the 10 CFR 50.46(b) acceptance criteria, uncertainties will be addressed in the BWRX-300 ECCS evaluation model to verify that there is a high level of probability that the BWRX-300 acceptance criteria would not be exceeded rather than the 10 CFR 50.46(b) acceptance criteria. The BWRX-300 evaluation model will not use the alternatives provided in 10 CFR 50 Appendix K.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(a)(1) and 10 CFR 50.46(a)(2).

- Regulatory Requirement: 10 CFR 50.46(b)(1), Peak cladding temperature, requires that the calculated maximum fuel element cladding temperature shall not exceed 2200°F.

  Statement of Compliance: The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant fuel cladding heatup occurs, such that the calculated maximum fuel element cladding temperature does not exceed the acceptance criterion of 2200°F.

  The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(1).

- Regulatory Requirement: 10 CFR 50.46(b)(2), Maximum cladding oxidation, requires that the calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

Statement of Compliance: The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant fuel cladding oxidization occurs, such that the calculated total oxidation of the cladding nowhere exceeds the acceptance criterion of 0.17 times the total cladding thickness before oxidation.

The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(2).

- Regulatory Requirement: 10 CFR 50.46(b)(3), Maximum hydrogen generation, requires that the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

  Statement of Compliance: The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant fuel cladding hydrogen generation occurs, such that the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed the acceptance criterion of 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

  The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(3).

- Regulatory Requirement: 10 CFR 50.46(b)(4), Coolable geometry, requires that the changes in core geometry shall be such that the core remains amenable to cooling.

  Statement of Compliance: The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that no significant changes in core geometry occur, such that the acceptance criterion of the core remaining amenable to cooling is met.

  The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(4).

- Regulatory Requirement: 10 CFR 50.46(b)(5), Long-term cooling, requires that after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Statement of Compliance: The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, and that long-term cooling removes decay heat and maintains the core temperature to acceptably low values, such that after any calculated successful initial operation of the ECCS, the acceptance criteria of the calculated core temperature being maintained at an acceptably low value and decay heat being removed for the extended period of time required by the long-lived radioactivity remaining in the core are met.

For the BWRX-300, [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed power without any further need of onsite or offsite electric power system operation.

The selected long-term cooling timeframe of [[ ]] is sufficient to maintain reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range for both [[

]] Following [[

]], the [[ ]] continues to provide long-term cooling to meet the BWRX-300 acceptance criteria in response to a LOCA and only requires operator action to [[ ]] after approximately seven days.

The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.46(b)(5).

- Regulatory Requirement: 10 CFR 50.46(d) states that the requirements of this section are in addition to any other requirements applicable to ECCS set forth in this part. The criteria set forth in paragraph (b), with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of Appendix A.

  Statement of Compliance: Compliance with these additional requirements are addressed in the discussions below.

4.1.3 10 CFR 50.55a

10 CFR 50.55a, Codes and standards, in 10 CFR 50.55a(a), Documents approved for incorporation by reference, lists the standards that have been approved for incorporation by reference by the Director of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51.

- Regulatory Requirement: 10 CFR 50.55a(a) includes standards that are required for evaluation of the RPV isolation valves. This rule establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components of BWR nuclear power plants by requiring conformance with appropriate editions of specified published industry codes and standards.

Statement of Compliance: The BWRX-300 RPV isolation valves and overpressure protection design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, including any application for a construction permit under 10 CFR 50 or design certification application under 10 CFR 52. These requirements are to be implemented during detailed design of the safety-related components of the [[ ]].

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.55a.

4.1.4 10 CFR 50 Appendix A, GDC 1

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 1, Quality standards and records, requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

  Statement of Compliance: The BWRX-300 RPV isolation and overpressure protection design features, [[ ]] are to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed in accordance with generally recognized codes and standards, and under an approved quality assurance program with approved control of records.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 1.

4.1.5 10 CFR 50 Appendix A, GDC 2

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 2, Design bases for protection against natural phenomena, requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Statement of Compliance: The BWRX-300 RPV isolation and overpressure protection design features, [[ ]] are to be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. Specific design requirements for the [[ ]] used to verify the capability to perform their safety functions, and the natural phenomena and effects evaluated, will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 2.

4.1.6 10 CFR 50 Appendix A, GDC 4

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 4, Environmental and dynamic effects design bases, requires that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

  Statement of Compliance: As stated in this LTR, a design requirement of the BWRX-300 is that the SSCs required to mitigate a LOCA shall be operable in the environmental conditions (PCV pressure, temperature, radiation, etc.) following a LOCA. In addition, the dynamic effects of postulated pipe breaks are to be evaluated in the BWRX-300 design. As described in this LTR, [[ ]] the BWRX-300 design requirements include evaluation of the acceptable criteria to identify postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," as discussed in Subsection 2.4.1 of this LTR.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 4.

4.1.7 10 CFR 50 Appendix A, GDC 14

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 14, Reactor coolant pressure boundary, requires that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Statement of Compliance: [[

]] This results in an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture, in compliance with this criterion. For piping that have [[

]]. Further design details are to be described during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 14.

4.1.8 10 CFR 50 Appendix A, GDC 15

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 15, RCS design, requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

Statement of Compliance: Overpressure protection for the BWRX-300 is provided in accordance with ASME B&PV Code, Section III, Paragraph NB-7120 Subparagraphs [[ ]]. The combination of RPS and [[ ]] design features ensure that the acceptance criteria for each component in the protected system are met including 1) the overpressure does not exceed 1.1 times design pressure for the expected system pressure transient condition, and 2) the calculated stress intensity and other design limitations for Service Level C are not exceeded for the unexpected system excess pressure transient condition, with further design details to be described during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 15.

4.1.9 10 CFR 50 Appendix A, GDC 30

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 30, Quality of reactor coolant pressure boundary, requires that components which are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Statement of Compliance: The components of the RCPB, including the ICS and RPV isolation valves, and the overpressure protection equipment [[ ]] are to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed in accordance with generally recognized codes and standards, and under an approved quality assurance program with approved control of records, as required by 10 CFR 50.55a and 10 CFR 50 Appendix A, GDC 1. In addition, means are to be provided to detect and identify the location of the source of reactor coolant leakage, including the components of the ICS and

RPV isolation valves, for components of the RCPB, with further design details to be described during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 30.

4.1.10 10 CFR 50 Appendix A, GDC 31

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 31, Fracture prevention of reactor coolant pressure boundary, requires that the RCPB shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

  Statement of Compliance: The components of the RCPB, including the ICS and RPV isolation valves, are to be designed with sufficient margin to assure that these requirements are met, with further design details to be described during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 31.

4.1.11 10 CFR 50 Appendix A, GDC 33

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 33, Reactor coolant makeup, requires that a system to supply reactor coolant makeup for protection against small breaks in the RCPB shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.
- Statement of Compliance: GDC 33 applies to small leaks in the RCPB that are [[ ]] (e.g., leakage from flanges or cracks in piping or other components), and which do not exceed the capability of the nonsafety-related high-pressure CRD system used as normal reactor coolant makeup during power operations. The maximum allowed leakage rate for continuing power operation is stipulated in the plant Technical Specifications. For leakage greater than the maximum allowed leakage rate, automatic reactor scram and automatic actuation of the [[ ]] is not anticipated for most of these small leaks because the normal means of makeup from the high-pressure CRD system and feedwater maintains the level in the normal operating range. However, these small leaks which do not exceed the capability of the nonsafety-related high-pressure CRD system are evaluated using specified acceptable fuel design limits rather than the BWRX-300 acceptance criteria in response to a LOCA.

The high-pressure CRD system pumps can operate using either onsite electric power system operation (assuming offsite power is not available) backed up by nonsafety-related standby diesel generators and offsite electric power system operation (assuming onsite power is not available).

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 33.

4.1.12 10 CFR 50 Appendix A, GDC 35

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 35, Emergency core cooling, requires a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Statement of Compliance: As previously described, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS in compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling, which is assured for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS through the use of [[ ]].

The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range, such that the performance of the [[ ]] is sufficient to ensure that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

[[

]]

The analyses to demonstrate compliance with the BWRX-300 acceptance criteria in response to a LOCA (i.e., reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range) will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 35.

4.1.13 10 CFR 50 Appendix A, GDC 37

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 37, Testing of emergency core cooling system, requires that the emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

  Statement of Compliance: As previously described, the combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b). In addition, the [[ ]] are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS in compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The [[ ]] has the capability to provide more than sufficient emergency core cooling, which is assured for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS through the use of [[

  ]].

Specific requirements for periodic pressure and functional testing of the [[ ]] to assure (1) the structural and leaktight integrity of these components, (2) the operability and performance of these active components, and (3) the operability of the systems as a whole containing the [[ ]] will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 37.

4.2 Regulatory Guides

4.2.1 Regulatory Guide 1.26

Regulatory Guide (RG) 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, Rev. 5, describes methods acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50 Appendix A, GDC 1, Quality Standards and Records, with regard to a quality classification system related to specified national standards that may be used to determine quality standards acceptable to the NRC Staff for components containing water, steam, or radioactive material in light-water-cooled nuclear power plants. The design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50 Appendix A, GDC 1. The RPV isolation valves and the components of the ICS are classified in conformance with the guidance provided in RG 1.26.

Therefore, the BWRX-300 design conforms to the guidance for the RPV isolation valves and the ICS, including regulatory positions of RG 1.26.

4.2.2 Regulatory Guide 1.29

RG 1.29, Seismic Design Classification, Rev. 5, describes methods acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50.55a(h), 10 CFR 50 Appendix A, GDC 2, Design Bases for Protection Against Natural Phenomena, and 10 CFR 50 Appendix S, Earthquake Engineering Criteria for Nuclear Power Plants, for use in identifying and classifying those features of LWR nuclear power plants that must be designed to withstand the effects of the Safe-Shutdown Earthquake (SSE).

The design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a(h), 10 CFR 50 Appendix A, GDC 2, and 10 CFR 50 Appendix S. The components of the ICS and RPV isolation valves are classified as Seismic Class I in conformance with the guidance provided in RG 1.29.

Therefore, the BWRX-300 design conforms to the guidance, including regulatory positions of RG 1.29.

4.2.3 Regulatory Guide 1.45

RG 1.45, Guidance on Monitoring and Responding to RCS Leakage, Rev. 1, describes methods acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50 Appendix A, GDC 14, RCPB, and 10 CFR 50 Appendix A, GDC 30, Quality of RCPB, with regard to selecting reactor coolant leakage detection systems, monitoring for leakage, and responding to leakage for light-water-cooled reactors. This guidance additionally cites 10 CFR 50.55a, Codes and Standards, which requires the performance of ISI and testing of nuclear power plant components so that the concept of defense in depth is applied to provide assurance that the structural integrity of the RCPB is maintained.

Subsections 4.1.4 and 4.1.7 describe how the design of the [[ ]] complies with the requirements of 10 CFR 50 Appendix A, GDC 1 and GDC 14. Additionally, the means for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage in compliance with the requirements of 10 CFR 50 Appendix A, GDC 30, and the requirements for ISI and inservice testing (IST) of the [[ ]] in compliance with the requirements of 10 CFR 50.55a, are to be demonstrated during future licensing activities.

Therefore, the BWRX-300 design conforms to the guidance, including regulatory positions of RG 1.45.

4.2.4 Regulatory Guide 1.84

RG 1.84, Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, Rev. 37, lists the ASME B&PV Section III Code Cases that the NRC has approved for use as voluntary alternatives to the mandatory ASME B&PV Code provisions that are incorporated by reference into 10 CFR 50. This applies to reactor licensees subject to 10 CFR 50.55a, Codes and Standards. These ASME B&PV Section III Code Cases are acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50 Appendix A, GDC 1, Quality standards and records, and 10 CFR 50 Appendix A, GDC 30, Quality of reactor coolant pressure boundary, and the requirements of 10 CFR 50.55a(c), Reactor Coolant Pressure Boundary, which requires, in part, that components of the RCPB must be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of the ASME B&PV Section III Code or equivalent quality standards.

Subsections 4.1.4 and 4.1.9 describe how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50 Appendix A, GDC 1, and 10 CFR 50 Appendix A, GDC 30, respectively. Subsection 4.1.3 describes how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a. Compliance with the requirements of 10 CFR 50.55a, including the use of ASME B&PV Section III Code Cases endorsed in RG 1.84 where necessary, is to be demonstrated during future licensing activities.

Therefore, the BWRX-300 design will conform to the guidance, including regulatory positions of RG 1.84.

4.2.5 Regulatory Guide 1.147

RG 1.147, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, Rev. 19, lists the ASME B&PV Section XI Code Cases that the NRC has approved for use as voluntary alternatives to the mandatory ASME B&PV Code provisions that are incorporated by reference into 10 CFR 50. This applies to reactor licensees subject to 10 CFR 50.55a, Codes and Standards. These ASME B&PV Section XI Code Cases are acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50 Appendix A, GDC 1, Quality standards and records, and 10 CFR 50 Appendix A, GDC 30, Quality of reactor coolant pressure boundary, and the requirements of 10 CFR 52.79(a)(11) which requires the final safety analysis report to include “a description of the program(s), and their implementation, necessary to ensure that the systems and components meet the requirements of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants in accordance with 50.55a of this chapter.” In 10 CFR 50.55a(a)(1)(ii), the NRC references the latest editions and addenda of ASME B&PV Code Section XI that the agency has approved for use.

Section 4.1.3 describes how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a. However, the requirements of ASME B&PV Code Section XI, Division 1, specifically apply during construction and operation activities of nuclear power plants for performance of ISI activities, and do not apply during the design of the BWRX-300. Therefore, compliance with the requirements of 10 CFR 50.55a for the conduct of ISI activities, including the use of ASME B&PV Section XI Code Cases endorsed in RG 1.147 where necessary, is to be demonstrated during future licensing activities.

Based on this discussion, the guidance of RG 1.147 does not apply to the BWRX-300 design phase in meeting the requirements of 10 CFR 50.55a.

4.2.6 Regulatory Guide 1.192

RG 1.192, Operation and Maintenance Code Case Acceptability, ASME OM Code, Rev. 3, lists Code Cases associated with the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) that the NRC has approved for use as voluntary alternatives to the mandatory ASME OM Code provisions that are incorporated by reference into 10 CFR 50. This applies to reactor licensees subject to 10 CFR 50.55a, Codes and Standards. These ASME OM Code Cases are acceptable to the NRC Staff for use in implementing the regulatory requirements of 10 CFR 50 Appendix A, GDC 1, Quality standards and records, and 10 CFR 50 Appendix A, GDC 30, Quality of reactor coolant pressure boundary, the requirements of 10 CFR 50.55a(f) which requires, in part, that Class 1, 2, and 3 components and their supports meet the requirements of the ASME OM Code or equivalent quality standards, and the requirements of 10 CFR 52.79(a)(11) which requires the final safety analysis report to include “a description of the program(s), and their implementation, necessary to ensure that the systems and components meet the requirements of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants in accordance with 50.55a of this chapter.” In 10 CFR 50.55a(a)(1)(iv), the NRC references the latest editions and addenda of ASME OM Code that the agency has approved for use.

Section 4.1.3 describes how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a. However, the requirements of ASME OM Code specifically apply during operation and maintenance activities of nuclear power plants for performance of IST activities, and do not apply during the design of the BWRX-300. Therefore, compliance with the requirements of 10 CFR 50.55a for the conduct of IST activities, including the use of ASME OM Code Cases endorsed in RG 1.192 where necessary, is to be demonstrated during future licensing activities.

Based on this discussion, the guidance of RG 1.192 does not apply to the BWRX-300 design phase in meeting the requirements of 10 CFR 50.55a.

4.3 NUREG-0800 Standard Review Plan Guidance

4.3.1 Standard Review Plan 3.9.6

Standard Review Plan (SRP) 3.9.6, Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints, Rev. 4, states that the areas of review include the functional design and qualification provisions and IST programs for safety-related pumps, valves, and dynamic restraints (snubbers) designated as Class 1, 2, or 3 under ASME B&PV Code Section III.

As described in Section 4.1, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 2, GDC 4, GDC 14, GDC 15, and GDC 37, and the requirements of 10 CFR 50.55a, during detailed design activities with specific requirements for the [[ ]] to be provided during future licensing activities. In addition, Section 4.1.3 describes that the BWRX-300 RPV isolation valves and overpressure protection design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, including any application for a construction permit under 10 CFR 50 or design certification application under 10 CFR 52. These requirements are to be implemented during detailed design of the safety-related components of the [[ ]]. Therefore, the existing SRP provides adequate guidance to use 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.

4.3.2 Standard Review Plan 5.2.2

SRP 5.2.2, Overpressure Protection, Rev. 3, states that the areas of review include the application of relief and safety valves and the RPS that ensures overpressure protection for the RCPB during operation at power. SRP 5.2.2 also discusses the application of pressure-relieving systems that function during low-temperature operation ensures overpressure protection for the RCPB during low-temperature operation of the plant (startup, shutdown).

The design features of the BWRX-300 RCPB include the use of the RPS and [[ ]] for overpressure protection, [[ ]]. As this is a non-traditional approach to meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 15, GDC 30, and GDC 31, alternate guidance applicable to SRP 5.2.2 for the BWRX-300 is recommended to provide guidance to use 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.

Specific discussions under Section I, Areas of Review, that are affected by the [[ ]] include the following:

- I.1.A. - For BWRs, the area of review for operation at power includes relief and safety valves on the main steamlines and piping from these valves to the suppression pool. The BWR design also may incorporate interfacing systems, such as an IC, to prevent challenges to the relief and safety valves during normal operations. The BWR description of the basic design concept; the systems, subsystems, and support systems providing overpressure protection to the RCPB; the components and instrumentation employed in these systems; and process and instrumentation diagrams should be reviewed for power operation.

It is recommended that this area of review should include [[ ]].

- Review Interfaces, item 8 – For BWRs, review of the IC for sufficient capacity to preclude actuation of the overpressure protection system (under SRP Section 5.4.13).

  This area of review should be revised to require review of the [[ ]].

- Review Interfaces, item 9 - For BWRs, review of the suppression pool capability to condense and cool the discharge from the safety valves (under SRP Section 6.2.1.1.C).

  This area of review [[ ]].

Specific discussions under Section II, Acceptance Criteria, for generic acceptance criteria that are affected by the [[ ]], include the following:

- Requirements, item 3 - 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi) require that RCS relief and safety valves meet TMI Action Plan Items II.D.1 and II.D.3 of NUREG-0737.

  These acceptance criteria should be revised, because the BWRX-300 [[ ]]. Refer to Subsection 4.1.1 of this LTR regarding the requirements of 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).

- Requirements, item 4 - 10 CFR 52.47(a)(8) provides the requirement for design certification reviews to comply with the technically relevant portions of the TMI requirements in 10 CFR 50.34(f).

  These acceptance criteria may be retained, although a reference to revising the review guidance for 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi) should be added. Refer to Subsection 4.1.1 of this LTR regarding the requirements of 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).

- Requirements, item 5 - 10 CFR 52.79(a)(17) provides the requirement for COL applications to comply with the technically relevant information in 10 CFR 50.34. This includes the TMI-related requirements specified by 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).

  These acceptance criteria may be retained, although the second sentence should be deleted and replaced with a reference to revising the review guidance for 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi). Refer to Subsection 4.1.1 of this LTR regarding the requirements of 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).

Specific SRP acceptance criteria that are affected by the [[ ]] include the following:

5. The design of the ECCS is reviewed to determine that it is capable of performing all of the functions required by the design bases.

The design functions of the [[

]] to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(5). The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above TAF or fuel cladding temperature is maintained within normal operating temperature range. These BWRX-300 acceptance criteria ensure the following:

1. No significant fuel cladding heatup occurs in the short-term.
2. No significant fuel cladding oxidization occurs.
3. No significant fuel cladding hydrogen generation occurs.
4. No significant changes in core geometry occur.
5. No significant fuel cladding heatup occurs in the long-term.

Although this is a non-traditional approach for the design of the ECCS for past LWRs, no active or passive injection of additional water inventory is required following the worst-case LOCA to meet these BWRX-300 acceptance criteria and to meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 14, GDC 30, GDC 31, and GDC 35. Therefore, the second and third areas of review regarding ADS and passive ECCS injection are not applicable. Other than these areas of review, the review interfaces, acceptance criteria, review procedures, evaluation findings, and references are acceptable for use for the BWRX-300 recognizing that those discussions related to the ADS and active or passive ECCS injection are not applicable. Based on these factors, the existing SRP provides adequate guidance to use 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.

4.3.5 Standard Review Plan 15.6.5

SRP 15.6.5, LOCAs resulting from Spectrum of Postulated Piping Breaks Within the RCPB, Rev. 3, includes review for compliance with the requirements of 10 CFR 50 Appendix A, GDC 35, as well as 10 CFR 50.46 and 10 CFR 50 Appendix K, and the applicable general design requirements discussed in SRP Section 6.3.

The design functions of the [[

]] to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(5). [[

]] to meet the requirements of 10 CFR 50 Appendix A, GDC 35, 10 CFR 50.46, and 10 CFR 50 Appendix K, and the applicable general design requirements discussed in SRP Section 6.3. Based on these factors, the existing SRP provides adequate guidance to use 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.

4.4 Generic Issues

The following generic issues provided are based on their relevance to the scope of this LTR, and an up-to-date evaluation of generic issues is to be provided during future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant requesting a CP and OL under 10 CFR 50 or a COL under 10 CFR 52.

4.4.1 NUREG-0737

NUREG-0737, Clarification of TMI Action Plan Requirements, November 1980, contains requirements approved for implementation by the NRC Commissioners as a result of the accident at TMI Unit 2. The NRC Commission subsequently recommended that certain of these requirements be added to the 10 CFR 50 regulations, which were subsequently implemented in 10 CFR 50.34(f). Compliance with the items that are related to [[ ]] are discussed in Subsection 4.1.1 of this LTR.

4.5 Operational Experience and Generic Communications

The operational experience and generic communications provided are based on their relevance to the scope of this LTR, and an up-to-date evaluation of operational experience and generic communications is to be provided during future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant requesting a CP and OL under 10 CFR 50 or a COL under 10 CFR 52.

4.5.1 Generic Letter 83-02

Generic Letter 83-02, NUREG-0737 Technical Specifications, dated January 10, 1983, contains a request for information for the current BWR licensees regarding NUREG-0737 items for which technical specifications are required, including guidance on the scope of a specification which the staff would find acceptable and sample technical specifications. This includes NUREG-0737 item II.K.3.3 for reporting relief and safety valve failures. This requirement is not applicable because [[ ]]. In addition, this requirement was not subsequently implemented in 10 CFR 50.34(f). However, technical specifications for the [[ ]] are to be proposed during future licensing activities.

4.5.2 Generic Letter 95-07

Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, dated August 17, 1995, contains a request to ensure that safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding are evaluated to ensure they are capable of performing the safety functions described in the licensing basis. This guidance will be evaluated for applicability during future licensing activities.

5.0 REFERENCES

5.1 ASME Boiler and Pressure Vessel Code Section III Rules for Construction of Nuclear Facility Components, Division 1 – Subsection NB Class 1 Components

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 ASME B16.5-2017 "Pipe Flanges and Flanged Fittings NPS ½ Through NPS 24 Metric/Inch Standard," American Society of Mechanical Engineers, 2017

5.4 ASME B16.34-2017 "Valves-Flanged, Threaded, and Welding End," American Association of Mechanical Engineers, 2017

5.5 ASME QME-1-2007 "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," American Society of Mechanical Engineers, 2007

5.6 NEDC-33911P, "BWRX-300 Containment Performance"

5.7 26A66412AW, Rev 10, "ESBWR Design Control Document, Tier 2, Chapter 7 Instrumentation and Control Systems," GE Hitachi Nuclear Energy, April 2014

Appendix D

Replaced Pages from NEDC-33910P Revision 0, Supplement 2

Revision Number	Description of Change
Supplement 1 (continued)	Revised to incorporate the following responses to NRC Requests for Additional Information (eRAIs): • NRC eRAI 9731, Question 03.06.02-1 supplemental response, revised Section 2.4.1 to describe the [[ ]] consistent with the discussions in NEDC-33911P, BWRX-300 Containment Performance. • NRC eRAI 9732, Question NONE-1, revised Section renumbered 4.1.11 to address compliance with 10 CFR 50, Appendix A, GDC 33. • NRC eRAI 9732, Question NONE-2, and supplemental response, revised Sections 1.1., 2.1, 2.7, 2.8, 4.1.1, 4.1.2, renumbered 4.1.11, renumbered 4.1.12, new 4.1.13, renumbered 4.3.4, and renumbered 4.3.5 to include compliance with the requirements of 10 CFR 50.46(b) by use of BWRX-300 acceptance criteria in response to a LOCA including maintaining reactor water level at or above TAF or fuel cladding temperature within normal operating temperature range, which bound the acceptance criteria in 10 CFR 50.46(b). • NRC eRAI 9732, Question NONE-3, and supplemental response, revised Sections 2.1, 2.2, 2.4, 2.7, 4.1.2, and renumbered 4.1.12, and Table 2-1, to reflect [[ ]] and to include compliance with the requirements of 10 CFR 50.46(b)(5) using a long-term cooling timeframe of [[ ]].
Supplement 2	Section 2.5 is revised to clarify that the ICS RPV isolation valves automatic isolation function uses logic and functionality similar to, rather than the same as, the ESBWR ICS containment isolation valves, the [[ ]] is removed from the design, and the [[ ]] is clarified as an example of the ICS RPV isolation valves automatic isolation function logic and functionality.

- The valve closure is a safety-related function.
- Compact valve and actuator assemblies are selected.
- Required Primary Containment Vessel (PCV) space allocation is minimized in proximity of the RPV.
- Electrical and digital controls are minimized inside the PCV.

Design Requirement:

- All BWRX-300 RPV isolation valves shall have a proven low leakage potential.

Design and administrative leakage limits are applied to valve selection during the BWRX-300 preliminary design and are based on plant design and event evaluations using offsite dose consequences compared to regulatory limits as well as containment design limits. The leakage criteria are analyzed as part of the plant safety analysis.

For BWRX-300 design, the application of motor-operated valves is constrained because there is no safety-related power supply other than limited Class 1E battery-stored power. Thus, motor-operated valves other than fail-closed magnetic-motor valves (i.e., solenoid operated valves) are not used for any RPV isolation valve applications.

Design Requirements:

- The RPV isolation valves for main steam line, feedwater, shutdown cooling, and reactor water cleanup shall fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means.
- [[
  ]] with valve actuators designed to maintain the valves in their as-is position by positive mechanical means.

A critical aspect of the valve and actuator selection to evaluate is the failure mode. The failure mode of the RPV isolation valves are determined based on the safety function of the connected system. [[

]]

The actuation signal for the RPV isolation valve closure is different for fail-close and fail-as-is isolation valves.

Design Requirements:

- The fail-close RPV isolation valves shall automatically close on high containment pressure indicating a LOCA.
- [[
  ]]

The ICS RPV isolation valves automatic isolation function uses logic and functionality similar to the ESBWR ICS containment isolation valves, which is described in the last two paragraphs of

ESBWR DCD Tier 2 Subsection 5.4.6.2.2 [Reference 5.2]. However, the BWRX-300 Instrumentation and Control (I&C) system has three divisions of safety-related I&C. [[

]]

2.6 Reactor Pressure Vessel Isolation Valve Actuator Design Requirements

Design Requirement:

- The RPV isolation valves and actuators shall be operable during events when the containment pressure and temperature are elevated.

A key design requirement is control of the temperature at the valve-actuator interface in order to limit thermal effects on the actuator assembly. The RPV isolation valves are heated by process water or steam, which also elevates valve actuator temperatures above the local ambient. Valve and valve actuator designs are qualified in accordance with ASME QME-1 [Reference 5.5] to include evaluation of the local environmental conditions, including evaluation of the effects of heat transfer from the process water or steam and Design Basis Events. [[ ]] The stem connection and actuator mounting method are studied to determine if thermal isolation needs to be implemented. High-temperature seals or lubricants are used for the actuators.

Design Requirement:

- Control devices (e.g., pilots) that rely on electric power may be located outside the PCV when practical.

Locating the control devices for the RPV isolation valve actuators that rely on electric power outside the PCV eliminates harsh Environmental Qualification (EQ) requirements.

2.7 Categories of Pipe Breaks

Steam and liquid line breaks are evaluated. The pipe breaks evaluated in the safety analysis are divided into two size categories:

- [[
-]]

The largest steam line break is a main steam line break. The largest liquid line break is the feedwater line break. [[

]]

Design Requirements:

- [[