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.