SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT NEDC-33911P, REVISION 0, SUPPLEMENT 1

BWRX-300 CONTAINMENT PERFORMANCE

GE-HITACHI NUCLEAR ENERGY

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1.0 INTRODUCTION

The purpose of the GE-Hitachi Nuclear Energy Americas, LLC (GEH), Licensing Topical Report (LTR) NEDC-33911P, "BWRX-300 Containment Performance," Revision 0, submitted on March 31, 2020 (Agencywide Documents Access and Management System (ADAMS) under Accession No. ML20091S340), with Supplement 1, submitted on September 4, 2020 (ADAMS Accession No. ML20248H570); is to provide the design requirements, analytical methods, acceptance criteria, and regulatory bases for the containment performance design functions of the BWRX-300 small modular reactor. Specifically, the LTR addresses design requirements for the containment and the passive containment cooling system (PCCS), the containment isolation valves (CIVs), the analytical methods for evaluating containment performance, and the acceptance criteria for BWRX-300 containment performance.

In addition, the LTR provides: (1) a technical description of the BWRX-300 containment, PCCS, CIV design features and design functions, acceptance criteria, regulatory bases, and references to existing proven design concepts; (2) a technical description of the BWRX-300 analytical methods to be used to demonstrate compliance with containment, PCCS, and CIV acceptance criteria; and (3) a regulatory review of the BWRX-300 containment, PCCS, and CIV design features and design functions and the BWRX-300 analytical methods to be used to demonstrate compliance with containment, PCCS, and CIV acceptance criteria. This safety evaluation (SE) describes consistency with regulatory requirements and alternative approaches to regulatory guidance that may be referenced in future licensing activities, either in support of a design certification application (DCA) under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, certifications, and approvals for nuclear power plants," or by an applicant requesting a construction permit (CP) and operating license (OL) under 10 CFR Part 50, "Domestic licensing of production and utilization facilities," or a design certification and a combined license (COL) under 10 CFR Part 52.

In this SE, the U.S. Nuclear Regulatory Commission (NRC) staff details its review of NEDC-33911P and the acceptability of the LTR provisions for the BWRX-300 containment performance design functions. In response to NRC staff requests for additional information, GEH submitted responses dated June 26, July 21, July 24, August 7, and September 4, 2020 (ADAMS Accession Nos. ML20178A706, ML20213C745, ML20206L386, ML20220A581, and ML20248H570 respectively). The NRC staff will evaluate the compliance of the final design of the BWRX-300 containment performance functions during future licensing activities in accordance with 10 CFR Part 50 or 10 CFR Part 52. In this SE, double brackets indicate proprietary information.

2.0 TECHNICAL EVALUATION OF CONTAINMENT PERFORMANCE

2.1 General Introduction

2.1.1 Reactor Pressure Vessel (RPV)

NEDC-33911P, Section 2.1.1, “Reactor Pressure Vessel,” describes the RPV for the GEH BWRX-300. The RPV is a vertical, cylindrical pressure vessel. The height of the RPV design permits natural circulation driving forces to produce abundant reactor core coolant flow.

2.1.2 Isolation Condenser System (ICS)

NEDC-33911P, Section 2.1.2, “Isolation Condenser System,” describes the ICS for the BWRX-300. The ICS passively removes heat from the reactor when the normal heat removal system is unavailable. Passive removal means that heat transfer from the isolation condenser (IC) heat exchanger tubes to the surrounding IC pool water is accomplished by condensation and natural circulation, and no forced circulation equipment is required. The ICS consists of three independent trains, each containing an IC heat exchanger that condenses steam on the tube side and transfers heat by heating/evaporating water in the IC pool, which is vented to the atmosphere. The IC pools have a total installed capacity that provides approximately seven days of reactor heat removal capability. The heat rejection process can be continued by replenishing the IC pool inventory.

The NRC staff’s review of the ICS for the heat removal function is described in the SE Section 2.1.2 for GEH LTR NEDC-33910, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection.” (ADAMS Accession No. ML20091S367).

In addition, the NRC staff finds that NEDC-33911P, Figure 2-7, “Isolation Condenser CIVs Connected to the RPV Boundary,” shows that the ICS piping layout penetrating containment for the BWRX-300 is different from the piping layout for the previously-certified economic simplified boiling water reactor (ESBWR). Therefore, the containment isolation function of the ICS containment penetration is governed by the regulatory requirements of General Design Criterion (GDC) 55, “Reactor Coolant Pressure Boundary Penetrating Containment,” is subject to more detailed review, and is described in Section 5.1.22, “10 CFR Part 50, Appendix A, GDC 55,” of this SE.

2.2 Overview of Containment

NEDC-33911P, Section 2.2.1, “Containment Design Functions,” presents an overview of the BWRX-300 containment design that is based on GEH boiling water reactor (BWR) experience and fleet performance. The BWRX-300 containment is an underground subterranean steel or reinforced concrete primary containment vessel (PCV) or a combination steel and reinforced concrete vessel of similar size and functional features. With a containment size comparable to a small BWR drywell, NEDC-33911P states that the PCV peak accident pressure and temperature are within the existing BWR experience base. Even though the BWRX-300 containment does not have a suppression pool, its atmosphere is initially nitrogen-inerted like that of BWR Mark I/Mark II. The containment pressure and temperature are maintained by fan coolers during normal operation, and heat removal is achieved by the PCCS upon loss of active containment cooling, as described in NEDC-33911P, Section 2.2.8, “Passive Containment Cooling System.” The reactor cavity pool for the PCCS heat removal during design basis events (DBEs) is located above the containment and is vented to the atmosphere.

NEDC-33911P states that the BWRX-300 containment subcompartments include the volume below the RPV, the space between the RPV and the biological shield, and the containment head area above the refueling bellows. Section 5.3.6, “Standard Review Plan 6.2.1.2,” of this SE summarizes the NRC staff’s evaluation of the information provided in this LTR regarding the pressure differentials across the subcompartment walls due to pipe breaks.

NEDC-33911P, Section 2.2, “Overview of Containment,” states that the BWRX-300 does not need to have combustible gas control for design-basis accidents (DBAs) because the BWRX-300 containment atmosphere is well mixed due to the open connections between containment and the volume below the RPV and containment and the space between the RPV and the biological shield, and because the containment atmosphere is initially nitrogen-inerted. However, 10 CFR 50.44(c)(1), requires that all containments must have a capability for ensuring a mixed atmosphere during DBAs and significant beyond design-basis accidents (BDBAs). In NEDC-33911P, Section 5.1.2, GEH has indicated that LTR NEDC-33921P, “BWRX-300 Severe Accident Management,” will address compliance with this requirement for beyond-design-basis (BDB) events and severe accident management.

2.2.1 Containment Design Functions

NEDC-33911P, Section 2.2.1 specifies the primary design functions of the BWRX-300 PCV. The primary design functions include:

- Enclosing and supporting the Nuclear Boiler System (NBS) RPV and its connected piping systems;
- Providing associated radiation shielding; and,
- Providing a boundary for radioactive contamination released from the NBS or from portions of systems connected to the NBS that are located inside the PCV.

NEDC-33911P states that the PVC design uses a nitrogen-inerted containment atmosphere that provides dilution of hydrogen and oxygen gases released in a post-accident condition, and that dilution provides protection to the PCV and its internal components from hydrogen combustion or detonation.

2.2.2 Containment Design Requirements

NEDC-33911P, Section 2.2.2 provides a list of containment design requirements for BWRX-300. The containment, known as PCV, is classified as a Safety Class 1, safety-related, and Seismic Category I structure. The list includes applicable American Society of Mechanical Engineers (ASME) and ANSI/AISC N690 codes and NRC review guidance in SRP Branch Technical Position (BTP) 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment.”

The NRC staff noted that additional regulatory requirements are described in NEDC-33911P, Chapter 5, “Regulatory Evaluation,” and the corresponding SE sections are in Chapter 5, “Regulatory Evaluation,” of this SE.

2.2.3 Containment Performance Requirements

NEDC-33911P, Section 2.2.3, “Containment Performance Requirements,” describes the PCV’s performance to contain the loss-of-coolant accident (LOCA) mass and energy release and, as a backup discharge volume, accommodate the additional non-condensable (NC) gas from the ICS vents. The PCV design anticipates a service life of 60 years.

2.2.4 Containment Boundary

NEDC-33911P, Section 2.2.4, “Containment Boundary,” describes the PCV physical design boundary being used to interpret design code applicability to the PCV.

2.2.5 Access and Maintenance

NEDC-33911P, Section 2.2.5, “Access and Maintenance,” describes the design for refueling access and periodic inspection, personnel hatches, and installed crane rails and cart tracks for access and maintenance.

2.2.6 Containment Penetrations

NEDC-33911P, Section 2.2.6, “Containment Penetration,” describes the BWRX-300 containment penetration design. The PCV structure, in conjunction with concurrent operation of containment isolation function(s) limit fission product leakage during and following the postulated DBA. Hydraulic lines for the fine motion control rod drive (FMCRD) scram function use penetrations without isolation valves based on being closed-system piping outside the PCV and having integral reactor coolant pressure boundary (RCPB) isolation in the design of the drives. NEDC-33911P states that the PCV design has provisions for periodic testing to measure the integrated leakage rate from the PCV structure to confirm the leak-tight integrity of the pressure boundary.

The NRC staff’s review of the containment penetrations is described in SE Section 5.1.21, “10 CFR Part 50, Appendix A, GDC 54,” and Section 5.3.12, “Standard Review Plan Section 6.2.4,” pertaining to the containment isolation system.

2.2.7 Containment Isolation Valves

NEDC-33911P, Section 2.2.7, “Containment Isolation Valves,” states that CIVs provide the necessary isolation of the containment in the event of accidents or other conditions and prevent the unfiltered release of containment contents that would exceed 10 CFR 50.34(a)(1)(ii)(D) limits. NEDC-33911P, Section 2.2.7, describes how the design will comply with GDC 55, GDC 56, “Primary containment isolation,” and GDC 57, “Closed system isolation valves,” and includes information in the following figures.

- Figure 2-4, “RPV Isolation Valve assembly (Example),” shows an example of RPV isolation valves. Figure 2-5, “Main Steam and Feedwater CIVs Connected to RPV Boundary,” and Figure 2-6, “CIVs Connected to RPV Boundary,” show the systems that are connected to the RPV boundary and CIVs to meet GDC 55.

- Figure 2-7, “Isolation Condenser CIVs Connected to the RPV Boundary,” shows the ICS connections to the RPV boundary and other ICS CIVs and process valves.

- Figure 2-8, “FMCRD CIVs Connected to RPV Boundary,” shows the lines to the FMCRDs.

- Figure 2-9, “CIVs Connected to Containment Atmosphere,” and Figure 2-10, “CIVs Connected to Closed Systems,” show CIVs that are connected to containment atmosphere and closed systems in order to meet GDC 56 and GDC 57, respectively.

NEDC-33911P, Section 2.2.7, states that leak-tightness of CIVs is verified by 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” Type C tests. NEDC-33911P states that leak-tightness of the containment is verified by 10 CFR Part 50, Appendix J, Type A testing. NEDC-33911P also states that leak-tightness of other containment penetrations is verified by 10 CFR Part 50, Appendix J, Type B testing. The discussion of applicable GDCs are in Section 5.1.22, “10 CFR 50 Appendix A, GDC 55,” Section 5.1.23, “10 CFR 50 Appendix A, GDC 56,” and Section 5.1.24, “10 CFR 50 Appendix A, GDC 57,” respectively.

The NRC staff finds that NEDC-33911P, Figure 2-7, indicates that the ICS piping layout penetrating containment for the BWRX-300 is different from the ESBWR piping layout. The containment isolation function of the ICS containment penetration is governed by the regulatory requirements of GDC 55 and is subject to a more detailed review, and is described in Section 5.1.22 of this SE.

The NRC staff’s review pertaining to 10 CFR Part 50, Appendix J testing is described in SE Section 5.1.26, “10 CFR Part 50, Appendix J.” The review of the CIVs pertaining to the containment isolation system is described in SE Section 5.3.12. The NRC staff’s evaluation of consistency with applicable GDCs is described in SE Section 5.1.22 for GDC 55, Section 5.1.23, “10 CFR 50 Appendix A, GDC 56,” and Section 5.1.24, “10 CFR 50 Appendix A, GDC 57,” respectively.

2.2.7.1 Containment Isolation Valves Connected to RPV Boundary

NEDC-33911P, Section 2.2.7.1, “Containment Isolation Valves Connected to RPV Boundary,” states that [[ ]] NEDC-33911P, Section 5.3.12, states that [[ ]] Small pipes for level instruments use excess flow check valves (EFCVs) to conform to the provisions of Regulatory Guide (RG) 1.11, “Instrument Lines Penetrating the Primary Reactor Containment.” Section 2.2.7.1 describes that the closed-loop reactor coolant piping outside containment in lieu of outboard CIVs are designed for the ICS and FMCRD. Section 2.2.7.1 states that [[ ]] and that the ICS provides emergency core cooling functions, and the ICS RPV isolation valves, [[ ]], will close if a pipe break is detected. However, the NRC staff noted that these two RPV isolation valves do not provide single-failure proof containment isolation, which requires two CIVs: one inboard and one outboard. [[ ]]

The NRC staff’s review of the CIVs pertaining to the containment isolation system is described in SE Section 5.3.12, and consistency with the requirements in GDC 55 for containment isolation is described in SE Section 5.1.22. Specifically, the NRC staff reviewed the isolation of ICS containment penetration using [[ ]] and using closed-loop reactor coolant piping outside the containment in lieu of outboard CIVs for ICS and FMCRD as described in Section 5.1.22 of this SE.

2.2.7.2 Containment Isolation Valves Connected to Containment Atmosphere

NEDC-33911P, Section 2.2.7.2, “Containment Isolation Valves Connected to Containment Atmosphere,” states that the BWRX-300 CIVs that are attached directly to the containment atmosphere include the integrated leak rate testing system, the emergency purging system, the containment inerting system nitrogen supply, the process gas and radiation monitoring system, and the floor drain sump system.

The NRC staff’s review of the CIVs pertaining to the containment isolation system is described in SE Section 5.3.12, and consistency with the requirements in GDC 56 for the containment isolation is described in Section 5.1.23 of this SE.

2.2.7.3 Containment Isolation Valves Connected to Closed Systems

NEDC-33911P, Section 2.2.7.3, “Containment Isolation Valves Connected to Closed Systems,” states that the BWRX-300 CIVs connected to the closed system, inside the containment, include the pneumatic nitrogen or air system, the service and breathing air system, the quench tank supply system, chilled water supply and return, and demineralized water system.

The NRC staff’s review of the CIVs pertaining to the containment isolation system is described in SE Section 5.3.12, and consistency with the requirements in GDC 57 for containment isolation is described in Section 5.1.24 of this SE.

2.2.8 Passive Containment Cooling System (PCCS)

NEDC-33911P, Section 2.2.8, “Passive Containment Cooling System (PCCS),” describes the conceptual design of the PCCS used in BWRX-300 for containment heat removal during accident conditions or upon the loss of active containment cooling during the DBEs. NEDC-33911P, Section 2.2.8.1, “PCCS Design Functions,” summarizes the PCCS design functions. The reactor cavity pool for the PCCS heat removal during DBEs is located above the containment and is vented to the atmosphere. As described in NEDC-33911P, Section 2.2.8.2, “PCCS Design Requirements,” the PCCS is designed in accordance with the containment design requirements as outlined in NEDC-33911P, Section 2.2.2. As the PCCS design is not finalized yet, NEDC-33911P, Section 2.2.8.3, “PCCS and Containment Boundary,” presents two example configurations of the PCCS geometry and the containment boundary.

3.0 TECHNICAL EVALUATION OF TRACG AND GOTHIC COMPUTER CODES FOR CONTAINMENT PERFORMANCE

3.1 Scope of the Evaluation Model

NEDC-33911P, Section 3.1, “Scope of the Evaluation Model,” describes the BWRX-300 containment DBEs as anticipated operational occurrences (AOOs), station blackouts (SBOs), Anticipated Transient without Scram (ATWS), and large break and small break LOCAs inside the containment. The large break LOCA events inside the containment are the double-ended guillotine break of one of the following pipes: main steam pipe, feedwater pipe, IC steam pipe, or IC condensate return pipe. At least one of the two RPV isolation valves installed on the broken line is closed during the large break LOCA, subject to the single failure criterion. Small 78/ inside the containment are assumed to remain un-isolated. These small pipes include instrument lines. As the PCCS does not rely on any active components to operate, SBO events are no different than long term AOO or ATWS events where the reactor is isolated with respect to the containment response. The only potential challenge to the containment in an SBO event is the long-term heat up of the reactor cavity pool.

Regulations in 10 CFR Part 50, Appendix A, GDC 38 and GDC 50, require the evaluation model to demonstrate that the design pressure and structure temperature bound the accident peak pressure and structure temperature, and that the heat removal systems reduce the containment pressure rapidly. NEDC-33911P states that the target for rapid depressurization is to reduce the pressure to the 50 percent of the peak accident pressure of the most limiting LOCA in 24 hours. Section 5.3.6 of this SE discusses the Standard Review Plan 6.2.1.2 acceptance criteria and evaluation modeling that is needed for the pressure differential across the subcompartment walls resulting from the postulated high-energy pipe breaks, to analyze the structural integrity of the BWRX-300 containment subcompartments.

3.2 Overview of the Evaluation Model

The BWRX-300 containment evaluation model utilizes the applicable parts of the ESBWR evaluation methods, which have been previously reviewed and approved for the ESBWR Design Certification. The BWRX-300 RPV is similar to the ESBWR RPV, but the BWRX-300 containment is different from the ESBWR containment. The BWRX-300 containment does not have the challenging modeling features of the ESBWR containment, such as the wetwell, suppression pool, a more complicated PCCS design, and the annulus between the RPV and the biological shield.

The BWRX-300 containment evaluation model uses the Transient Reactor Analysis Code General Electric (TRACG) ESBWR RPV model described in Section 3.3, "TRACG Mass and Energy Releases for Containment." The containment is modeled separately using Generation of Thermal-Hydraulic Information for Containments (GOTHIC). As described in Sections 3.4.2, 3.4.2.1, 3.5, "TRACG and Gothic Analyses Numerical Convergence," and 3.6, "Summary of the Containment Evaluation Method," of NEDC-33911P, the development of the BWRX-300 containment evaluation model follows the structure of RG 1.203, "Transient and Accident Analysis Methods." Conservative temperature and steam/NC gas composition distributions can be calculated for the BWRX-300 containment using an appropriate model with nodalization. Conservatism in the evaluation model is achieved by biasing the inputs and modeling parameters to bound the uncertainties, rather than performing a statistical analysis. GEH indicated its intent to demonstrate the conservatism of the evaluation model by benchmarking with the available test data as part of the application methodology described in LTR NEDC-33922P, "BWRX-300 Containment Evaluation Method." The NRC staff will conduct a detailed evaluation of the validation against the test data to confirm that the BWRX-300 containment evaluation methodology follows RG 1.203, for a conservative analysis utilizing mature computer codes with an extensive qualification base, during future BWRX-300 licensing activities.

3.3 TRACG Mass and Energy Releases for Containment

GEH plans to use TRACG to model the BWRX-300 RPV neutronics and thermal-hydraulics, and to perform the containment mass and energy release calculations. NEDC-33911P, Section 3.3 lists four previously submitted GEH LTRs to describe the TRACG model and qualification. The NRC staff will conduct a detailed evaluation of the applicability of these previous TRACG submittals to the BWRX-300 design, during the future BWRX-300 licensing activities.

The NRC staff reviewed the analytical details and technical evaluation provided in NEDC-33911P, Section 3.0, "Technical Evaluation of TRACG and GOTHIC Computer Codes for

Containment Performance," to evaluate their consistency with GDCs 16, 38, and 50. The NRC staff also evaluated the information presented in LTR Section 3.0 against review guidance in NUREG- 0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Sections 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments," and 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)." As described in the LTR, the containment design will be based on consideration of a full spectrum of postulated DBEs that would result in the release of reactor coolant to the containment. These containment DBEs include liquid, steam, and partial (both steam and liquid) breaks and will be evaluated using the [[ ]]. The evaluation of the containment design will be based on enveloping the results of the range of analyses, plus provision for sufficient margin. The most limiting short-term and long-term pressure and temperature responses will be assessed to verify the integrity of the containment structure.

GEH provided a high-level overview of the containment DBEs for the mass and energy release from the RPV into the containment. The LTR describes the development of the overall evaluation model based on the use of TRACG for predicting mass and energy release to the containment and use of GOTHIC for predicting the resulting containment thermal-hydraulic response. The LTR recognizes that the evaluation model objective is to show that the containment design pressure and temperature bound the accident peak pressure and temperature, and that the heat removal systems reduce the containment pressure rapidly, to demonstrate compliance with GDC 38 and GDC 50. These results will also be used for equipment environmental qualification. However, the LTR provides no further modeling details and assumptions about the conservatism, correlations, uncertainty, biases, nodalization, and validation used in the licensing basis containment safety analyses. Therefore, the NRC staff determined that the information provided in the present LTR, NEDC-33911P, is inadequate to review and make a regulatory finding regarding the proposed TRACG/GOTHIC containment analysis methodology. However, GEH indicated its intent in the LTR, to provide further information needed to demonstrate regulatory compliance during future BWRX-300 licensing activities.

To ensure that GEH will provide the remaining information for the NRC staff's review and approval before use of the present LTR as a reference in future licensing actions, the NRC staff is imposing a limitation and condition on this LTR, as documented in Section 4.0, "Containment Performance Acceptance Criteria," of this SE. The NRC staff will also conduct a detailed evaluation of the containment safety analyses to confirm that the final BWRX-300 containment design satisfies GDC 16, 38, and 50, during future BWRX-300 licensing activities.

3.4 GOTHIC Containment Model

3.4.1 Overview of the GOTHIC Computer Code

GOTHIC is a thermal-hydraulics code specifically developed for nuclear power plant containments and similar confinements. GOTHIC solves the mass, momentum and energy conservation equations in multi-dimensional and/or lumped-parameter volumes. GOTHIC allows for steam/gas mixture, continuous liquid, and liquid droplets, as well as the secondary fields for mist and liquid components, and multiple species of NC gases. GEH intends to use GOTHIC for modeling the BWRX-300 containment without any code modifications.

3.4.2 Evaluation Model Development for GOTHIC

The methodology utilizes the code, scaling, applicability and uncertainty (CSAU) described in NUREG/CR-5249, Revision 4, and RG 1.203, with containment pressure and structure temperature being the two figures of merit.

3.4.2.1 Requirements of the Model

Following RG 1.203, NEDC-33911P, Section 3.1, the requirements of the model regarding the purpose of the evaluation method and analysis, transient and power plant classes, and figures of merit are established. NEDC-33911P, Section 3.4.2.1, "Requirements of the Model," identifies the systems, components, phases, geometries, fields and processes that must be modeled. Systems, subsystems, modules and components included in the BWRX-300 containment GOTHIC model are as follows (the ones modeled by TRACG are indicated within parentheses):

- Primary containment, including enclosed volume, heat sinks and heat transfer surfaces
- Reactor vessel, including internals which serve as heat sinks (TRACG)
- RPV isolation valves, their actuators and the control systems (TRACG)
- Fuel (TRACG)
- RPS and ICS initiation control system(s) (TRACG)
- Piping systems
- ICS (TRACG)
- PCCS
- Reactor cavity pool
- Feedwater and control rod drive (CRD) systems which may add water from outside containment (TRACG)

The GOTHIC model uses a small dry containment geometry with water, nitrogen, hydrogen, and oxygen as constituents/chemical forms of the fluids, and steel and concrete structures/heat slabs, while uranium dioxide fuel and zircalloy cladding are used in the RPV TRACG model. The phenomena identified involve the transport of, and interactions between, constituent phases throughout the system. In future licensing activities, the NRC staff will evaluate the safety significance of the LTR assertion that "The geometrical shapes/configurations defined for a given transfer process (e.g., pool, drop, bubble, film, etc.) are enveloped by ESBWR design for TRACG, because the reactor, fuel, isolation condenser, isolation valves and control systems are like ESBWR."

3.5 TRACG and GOTHIC Analyses Numerical Convergence

NEDC-33911P, Section 3.5, "TRACG and GOTHIC Analyses Numerical Convergence," states that ensuring the individual numerical convergences of TRACG and GOTHIC and the overall convergence of the iteration between the two codes, is part of the application method. The

TRACG and GOTHIC analyses iteration continues until there is no significant change in the containment pressure and temperature, which is done by automatically limiting the time step size to maintain the numerical error below the internal convergence criteria for both the codes. The NRC staff will review the acceptance criteria for the sufficiency of convergence to be established as part of NEDC-33922P. NEDC-33922P will also include a BWRX-300 containment nodalization sensitivity study to support the nodalization used in the application method. The NRC staff will also evaluate the safety significance of the LTR assertion that "Nodalization of the BWRX-300 RPV is consistent with and as fine as the ESBWR RPV nodalization, which was successfully demonstrated in the ESBWR application methodology."

3.6 Summary of the Containment Evaluation Method

As described in NEDC33911P, Sections 3.1, 3.4.2, 3.4.2.1, and 3.5, the applicable steps in RG 1.203 are followed to establish a conservative containment evaluation method and define the purpose of the evaluation method, figures of merit, and convergence of the evaluation method and the transient analyses.

During the review, GEH made significant changes to NEDC-33911P, Revision 0, on the GOTHIC PIRT. GEH moved Section 3.4.2.2, "GOTHIC Phenomenon Identification and Ranking Table (PIRT)," Section 3.4.2.3, "PIRT Survey," Section 3.4.2.4, "Development of the Assessment Base," Table 3-1, "Phenomena Ranking Criteria," and Table 3-2 to NEDC-33922P for review with the TRACG/GOTHIC methodology. GEH plans to further discuss the PIRT phenomena in NEDC-33922P and present the sensitivity and demonstration cases to provide greater technical justification for the rankings. Therefore, the NRC staff will not make an overall finding about the GOTHIC PIRT in this SE. The NRC staff will need to conduct a detailed evaluation of the GOTHIC PIRT, along with the other details of the TRACG/GOTHIC methodology, to confirm that all phenomena related to the containment evaluations for the DBEs are covered in TRACG and GOTHIC.

This would complete the remaining RG 1.203 elements up to the PIRT. The other elements of the method, including the demonstration analyses and the specifics of the application method are planned to be presented in LTR NEDC-33922P, BWRX-300 Containment Evaluation Method. A key specific that will be reviewed by the NRC staff is the use of the TRACG ESBWR model for the mass and energy release from the BWRX-300 RPV as boundary conditions to the GOTHIC containment model.

4.0 CONTAINMENT PERFORMANCE ACCEPTANCE CRITERIA

Section 4.0, "Containment Performance Acceptance Criteria," of NEDC-33911P, specifies the following BWRX-300 containment performance acceptance criteria.

- NEDC-33911P states that the containment pressure boundary and penetrations are designed for the design pressure and temperature to be established for DBAs during future licensing activities in accordance with 10 CFR Part 50, Appendix A, GDC 2, GDC 4, GDC 16, GDC 38, GDC 41, GDC 50, and GDC 51.

- NEDC-33911P states that, in accordance with 10 CFR Part 50, Appendix A, GDC 4, GDC 16, GDC 38, GDC 41, GDC 50, and GDC 51, containment design pressure will be evaluated during future licensing activities to bound the peak accident containment pressure resulting from the most limiting large break LOCA with margin, with no less than 10 percent margin during the Preliminary Safety Analysis Report (PSAR) phase in order to conform to SRP Section 6.2.1.1.A, Acceptance Criteria.

- NEDC-33911P states that, in accordance with 10 CFR Part 50, Appendix A, GDC 16, GDC 38, and GDC 50, the BWRX-300 containment design features establish an essentially leak-tight barrier, and will be demonstrated during future licensing activities to reduce containment pressure and temperature rapidly, and maintain them at acceptably low levels following a LOCA; and the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a LOCA.

5.0 REGULATORY EVALUATION

5.1 10 CFR Part 50 Regulations

5.1.1 10 CFR 50.34(f)

The NRC regulations in 10 CFR 50.34(f)(2)(xv), require the design to provide the capability to purge or vent the containment to minimize the purging time, consistent with the principle of keeping occupational exposure as-low-as-reasonably achievable (ALARA) for occupational exposure, and to provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. NEDC-33911P, Section 5.1.1, "10 CFR 50.34(f)," goes on to describe in detail the Three Mile Island Unit 2 (TMI Unit 2) containment requirements.

NEDC-33911P, Section 5.1.1, states that all nonessential systems automatically isolate with two isolation barriers in series except for nonessential instrument lines. None of the nonessential systems reopen on containment isolation reset signals and have a set point pressure for initiating containment isolation as low as compatible with normal operation. NEDC-33911P states that automatic closing on a high radiation signal is provided where required to meet the regulations in 10 CFR Part 100, "Reactor site criteria." Therefore, GEH states that the BWRX-300 design will meet the requirements of 10 CFR 50.34(f)(2)(xiv).

NEDC-33911P, Section 5.1.1, states that the BWRX-300 containment emergency purge system is designed to reliably isolate under accident conditions and is capable of purging and venting in consideration of ALARA occupational exposure. Therefore, GEH states that the BWRX-300 design will meet the requirements of 10 CFR 50.34(f)(2)(xv).

NEDC-33911P, Section 5.1.1, states that the BWRX-300 design includes instrumentation to measure, record and readout in the control room for containment pressure, containment water level, containment hydrogen and oxygen concentration, containment radiation level, and noble gas effluents at release points to the environment with continuous sampling capability for radioactive iodines and particulates in gaseous effluents and onsite capability to analyze and measure these samples accordingly. Therefore, GEH states that the BWRX-300 design will meet the requirements of 10 CFR 50.34(f)(2)(xvii).

NEDC-33911P, Section 5.1.1, states that the ASME B&PV Code, Section III, Division 1 or Division 2 requirements and additional requirements specified are to be met for the design of the BWRX-300 containment depending on whether a steel or concrete containment, or a combination of steel and concrete containment design, is chosen. Therefore, GEH states that the BWRX-300 design will meet the requirements of 10 CFR 50.34(f)(3)(v)(A)(1).

The NRC staff finds the approach, as described in NEDC-33911P, Section 5.1.1, to be consistent with 10 CFR 50.34(f) and, therefore, is acceptable. The NRC staff will conduct a detailed evaluation to confirm that 10 CFR 50.34(f)(2)(xiv), 10 CFR 50.34(f)(2)(xv), 10 CFR 50.34 (f)(2)(xvii), and 10 CFR 50.34(f)(3)(v)(A)(1) are satisfied when it receives an application for a BWRX-300.

5.1.2 10 CFR 50.44

The NRC regulations in 10 CFR 50.44(c) set forth combustible gas control requirements for future water-cooled nuclear power reactor designs. Section 2.2.1 of NEDC-33911P describes how BWRX-300 satisfies the 10 CFR 50.44 requirement with respect to the containment design function. It states that the BWRX-300 PCV design uses a nitrogen-inerted containment atmosphere during operating modes and that the inerted atmosphere provides dilution of hydrogen and oxygen gases released in a post-accident condition by radiolytic decomposition of water and the released hydrogen from water and fuel cladding (zirconium) reaction during a severe accident condition. It also states that the dilution protects the PCV and its internal components from hydrogen combustion or detonation.

In accordance with SRP Section 6.2.5, the NRC staff reviewed the BWRX-300 containment design for consistency with 10 CFR 50.44. Specifically, the NRC staff reviewed the report to determine whether the proposed containment design will include: (1) the capability to mix the combustible gases with the containment atmosphere and prevent high concentrations of combustible gases in local areas, (2) the capability to monitor combustible gas concentrations within containment and for inerted containments, and (3) the capability to reduce combustible gas concentrations within containment by suitable means such as igniters.

NEDC-33911P, Section 5.3.13, addresses the functional capability of the BWRX-300 combustible gas control systems to ensure that containment integrity is maintained. It states that the BWRX-300 design includes a dry, inerted containment that does not rely upon combustible gas control to keep concentrations of hydrogen and oxygen below combustible levels and maintain containment structural integrity following a DBA, which the staff finds to be consistent with 10CFR50.44(c)(2). NEDC-33911P, Section 5.3.13, states that the LTR evaluated herein does not cover Beyond Design Basis (BDB) events and severe accidents; nor does it describe how the design will comply with the requirements of 10 CFR 50.44(c)(1) and 10 CFR 50.44(c)(3). The applicant indicated these matters will be addressed in NEDC-33921P, “BWRX300 Severe Accident Management.”

The regulations in 10 CFR 50.44(c)(4) require reliable equipment for monitoring oxygen and hydrogen concentrations in inerted containments during and following a significant beyond-design-basis accident (BDBA). The design feature of the BWRX-300 used to comply with this regulation includes the requirement for oxygen and hydrogen analyzers to monitor oxygen and hydrogen concentrations, which the staff finds to be consistent with 10 CFR 50.44 (c)(4). NEDC-33911P, Section 5.1.1, includes the following statement:

> The BWRX-300 design includes instrumentation to measure, record and readout in the control room containment pressure, containment water level, containment hydrogen and oxygen concentration, containment radiation level, and noble gas effluents at specified release points to the environment with continuous sampling capability for radioactive iodines and particulates in gaseous effluents with onsite capability to analyze and measure these samples accordingly.

The regulations in 10 CFR 50.44(c)(5) require that a structural analysis be performed that demonstrates that containment structural integrity is maintained during accident conditions in which hydrogen is released from a 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. The demonstration must be an analytical technique acceptable to the NRC with supporting justification to show that the technique describes the containment response to the structural loads involved with systems necessary to ensure containment integrity to perform under these accident conditions.

NEDC-33911P, Section 5.1.2 states that the design requirement for the BWRX□300 containment structural integrity analysis is to demonstrate during future licensing activities the survivability of the containment to the structural loads generated from an accident where a 100 percent fuel clad□coolant reaction accompanied by hydrogen burning occurs.

The NRC staff finds that the NEDC-33911P approach for combustible gas control for the BWRX-300 is consistent with the requirements of 10 CFR 50.44(c)(2) and 10 CFR 50.44(c)(4) and, therefore, is acceptable for normal operating and DBA conditions. However, GEH indicated that while -NEDC33911P does not address BDB events and severe accidents or compliance with the requirements of 10 CFR 50.44(c)(1) and 10 CFR 50.44(c)(3), NEDC-33921P will address them. Additionally, GEH indicated that while NEDC-33911P does not address the containment structural integrity under structural loads generated from an accident in which a 100 percent fuel clad-coolant reaction accompanied by hydrogen burning occurs, GEH will address this analysis during a future licensing activity. The NRC staff will conduct a detailed evaluation to confirm compliance with 10 CFR 50.44(c) when it reviews NEDC-33921P or other future licensing activities.

5.1.3 10 CFR 50.55a

NEDC-33911P, Section 5.1.3, “10 CFR 50.55a,” states that the NRC regulations in 10 CFR 50.55a, “Codes and standards,” establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components of BWR nuclear power plants by requiring conformance with appropriate editions of specified published industry codes and standards. Section 5.1.3 states that the BWRX-300 containment and CIV design features are to be designed using the standards approved in 10 CFR 50.55a(a), “Documents approved for incorporation by reference,” in effect within six months of any license application, including any application for a CP under 10 CFR Part 50 or DCA under 10 CFR Part 52. Section 5.1.3 states that the BWRX-300 design will meet the requirements of 10 CFR 50.55a.

The NRC regulations in 10 CFR 50.55a(a) incorporate 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. The NRC staff finds that GEH’s plans stated in NEDC-33911P for the BWRX-300 design to satisfy the requirements of 10 CFR 50.55a are acceptable. The NRC staff will conduct a detailed evaluation to confirm that 10 CFR 50.55a is satisfied when it receives an application for a BWRX-300.

5.1.4 10 CFR 50.63

NEDC-33911P, Section 5.1.4, "10 CFR 50.63," states that the BWRX-300 design includes Class 1E battery-backed direct current (DC) power supplied to the safety related containment design features necessary for coping with an SBO. The operation of the ICS for RPV depressurization and decay heat removal does not require offsite electric power system operation, only requires one-time automatic actuation using onsite Class 1E battery-backed DC power, and then remains in service for at least 72 hours without any further need of onsite or offsite electric power system operation.

The NRC regulations in 10 CFR 50.63 require that light-water-cooled nuclear power plants licensed under 10 CFR Part 50 or Part 52 be able to withstand, for a specified duration, and recover from a station blackout. A station blackout is the complete loss of alternating current electric power to the essential and nonessential switchgear buses in the nuclear power plant. The NRC staff finds that the approach, as described in NEDC-33911P, is consistent with 10 CFR 50.63(a)(2) and, therefore, is acceptable. The NRC staff will conduct a detailed evaluation to confirm that 10 CFR 50.63(a)(2) is satisfied when it receives an application for a BWRX-300.

5.1.5 10 CFR Part 50, Appendix A, GDC 1

GDC 1, "Quality standards and records," establishes requirements for design, fabrication and construction of structures, systems, and components important to safety. NEDC-33911P, Section 5.1.5, "10 CFR Part 50, Appendix A, GDC 1," indicates that containment isolation penetration and CIVs are designed to meet the GDC 1 requirements. Section 5.1.5 states that the BWRX-300 containment and CIV design features, including the ICS, RPV isolation valve assemblies, PCCS, CIVs, containment structure, containment penetrations, piping, and instrumentation lines, 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.

The NRC staff finds that these provisions, as described in NEDC-33911P, are consistent with GDC 1, and therefore, are acceptable. When the NRC staff receives an application for a BWRX-300, it will conduct a detailed evaluation to confirm that it satisfies GDC 1.

5.1.6 10 CFR Part 50, Appendix A, GDC 2

GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that SSCs important to safety be designed to withstand the effects of natural phenomena. NEDC-33911P, Section 5.1.6, "10 CFR Part 50, Appendix A, GDC 2," indicates that containment isolation penetration and CIVs are designed to meet the GDC 2 requirements. Section 5.1.6 states that the BWRX-300 containment and CIV design features, including the ICS, RPV isolation valve assemblies, PCCS, CIVs, containment structure, containment penetrations, piping, and instrumentation lines, are to be designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches, without loss of capability to perform their safety functions.

The NRC staff finds that these provisions, as described in NEDC-33911P, are consistent with GDC 2, and therefore, acceptable. When the NRC staff receives an application for a BWRX-300, it will conduct a detailed evaluation to confirm that it satisfies GDC 2.

5.1.7 10 CFR Part 50, Appendix A, GDC 4

[[ ]]
SE Section 5.1.22 includes the details of the NRC staff's evaluation of consistency with GDC 55. This section of the SE specifically addresses how the BWRX-300 design provisions are consistent with the requirements of GDC 4.

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.

In NEDC-33911P, Section 2.2.2, Section 2.2.7, Section 2.2.7.1, "Containment Isolation Valves Connected to RPV Boundary," Section 3.1, and Section 5.1.7, GEH describes how the BWRX-300 design provisions will be consistent with the relevant NRC staff's guidelines as delineated in BTP 3-4 and, therefore, will meet the pertinent GDC 4 requirements.

In NEDC-33911P, Section 5.1.7, GEH states that the BWRX-300 containment and CIV design features are to be designed to the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. In addition, the BWRX-300 design will evaluate the dynamic effects of postulated pipe breaks. GEH further stated that the BWRX-300 design requirements include applying the design criteria from BTP 3-4, Part B, items 1(ii)(1)(d) and (e), to eliminate postulating breaks and cracks in those portions of piping from the containment wall to, and including, the outboard CIVs. Breaks and cracks will be postulated in those portions of piping from the RPV isolation valves [[ ]] to the containment wall, and the dynamic effects of those postulated pipe breaks will be evaluated in the BWRX-300 design.

Moreover, GEH stated that [[ ]] extending to the containment wall, the BWRX-300 design requirements will 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). Internal containment flooding will be evaluated during future licensing activities. Therefore, GEH concluded that the BWRX-300 design will meet the requirements of GDC 4.

In NEDC 33911P, Section 2.2.2 and Section 5.1.7, GEH stated that, in addition to ASME *Boiler and Pressure Vessel Code* (BPV Code), Section III, "Rules for Construction of Nuclear Power Plant Components," Division 1, Subarticle NE-1120, the design criteria from BTP 3-4, items 1(ii)(1)(d) and (e) and items 1(ii)(2) through (7), will also be applied to eliminate postulated breaks and cracks in those portions of piping from the containment wall to, and including, the outboard CIVs. The NRC staff found the pertinent BWRX-300 design criteria to be acceptable because the BWRX-300 design criteria will be consistent with the pertinent BTP 3-4 guidelines for eliminating postulated breaks and cracks in those portions of piping. The NRC staff will conduct a detailed evaluation to confirm that the design criteria for postulating breaks and cracks are consistent with the pertinent BTP 3-4 guidelines when it receives the application for a BWRX-300.

Regarding potential dynamic effects on the functionality of those outboard CIVs resulting from postulated pipe breaks beyond those portions of piping from the containment wall to, and including, the outboard CIVs, the CIV design and qualification will comply with ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants" (or a later edition), as endorsed by RG 1.100, "Seismic Qualification of Electrical and Active

Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Revision 3, issued September 2009. GEH will address the compliance with ASME Standard QME-1-2007 (or later edition), as endorsed by RG 1.100, in the detailed design and procurement process for the valves and will specify it during future licensing activities. The NRC staff found this to be acceptable because the CIV design and qualification will comply with ASME Standard QME 1-2007 (or later edition) as endorsed by RG 1.100. As stated in SE Section 5.1.22, the NRC staff will conduct a detailed evaluation to confirm that the final design of the CIVs satisfies the NRC regulations when it receives an application for a BWRX-300.

In NEDC-33911P, Section 3.1, GEH stated that the methodology for assessing jet loads resulting from pipe breaks are not in the scope of the evaluation method described in this section for the BWRX-300 containment response. GEH further stated that the jet loads and zone of influence are evaluated using a separate structural method that will be described during future licensing activities. Consideration of jet loads and zone of influence is safety significant because it provides assurance that a breach in the containment of the BWRX-300 will not occur and cause a radioactive release to the environment that exceeds regulatory requirements. Moreover, in NEDC 33911P, Section 3.1 and Section 5.1.7, GEH stated that the BWRX-300 design will consider all of the dynamic effects resulting from postulated high-energy pipe breaks, including the effects of jet loads, pipe whipping, missiles, and discharging fluids in the containment design and associated piping, valves, penetrations, and instrument lines in future licensing activities. The NRC staff found this to be acceptable because the BWRX-300 containment response to all of the dynamic effects resulting from postulated high-energy pipe breaks, including the effects of missiles, pipe whipping, and discharging fluids, if applicable, will be evaluated and described during future licensing activities to comply with the pertinent GDC 4 requirements. The NRC staff will conduct a detailed evaluation to confirm that the GDC 4 requirements are satisfied when it receives an application for a BWRX-300.

In NEDC-33911P, Section 5.1.7, GEH stated that breaks and cracks in those portions of piping from the RPV isolation valves [[ ]] to the containment wall remain postulated to occur, and the dynamic effects of those postulated pipe breaks will be evaluated in the BWRX-300 design. The capability of the CIVs to perform their design-basis functions is safety significant because it provides assurance that the containment of the BWRX-300 can be safely isolated and prevent radioactive release to the environment that exceeds regulatory requirements. GEH states that the BWRX-300 design will meet the requirements of GDC 4. GEH specified that the CIV qualification, such as compliance with ASME Standard QME-1-2007 (or a later edition) as accepted in RG 1.100, will be addressed in the detailed design and the procurement process of the CIVs and will be specified during future licensing activities. As stated in SE Section 5.1.22, the NRC staff will conduct a detailed evaluation to confirm that the final design of the CIVs satisfies the NRC regulations when it receives an application for a BWRX-300.

5.1.8 10 CFR Part 50, Appendix A, GDC 5

GDC 5 "Sharing of structures, systems, and components," places requirements on the sharing of SSCs important to safety among nuclear plant units. In Section 5.1.8, "10 CFR 50 Appendix A, GDC 5," of NEDC-33911P, the statement of compliance with GDC 5 states that the BWRX-300 design does not include sharing SSCs important to safety among units at multiunit sites. Therefore, GEH states that the BWRX-300 design will meet the requirements of GDC 5.

The NRC staff finds that the BWRX-300 containment design is consistent with the requirements of GDC 5 and is, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm that the BWRX-300 design conforms to GDC 5 when it receives an application for a BWRX-300.

5.1.9 10 CFR Part 50, Appendix A, GDC 13

GDC 13, “Instrumentation and control,” places requirements on instrumentation and controls. Section 5.1.9, “10 CFR 50 Appendix A, GDC 13,” of NEDC-33911P, describes the BWRX-300 instrumentation and controls that will be provided to monitor variables and systems important to the containment and associated systems over their anticipated ranges for normal operation for AOOs, and for accident conditions as appropriate to assure adequate safety. GEH intends to describe these instrumentation and control systems during future licensing activities. The NRC staff finds that the GEH description and plan to providing the monitoring instrumentation and controls for the containment and associated systems at the licensing stage, as documented in NEDC-33911P, are consistent with GDC 13 and are, therefore, acceptable. The NRC staff will conduct a detailed technical evaluation to confirm that the final BWRX-300 containment design satisfies the requirements of GDC 13 during future licensing activities of the BWRX-300.

5.1.10 10 CFR Part 50, Appendix A, GDC 16

GDC 16, “Containment design,” requires that reactor containment and associated systems be provided to establish an essentially leak-tight containment barrier against the uncontrolled release of radioactivity to the environment. NEDC-33911P, Section 5.1.10, “10 CFR Part 50, Appendix A, GDC 16,” describes a leak-tight BWRX-300 PCV that encloses the RPV and includes the RCPB as well as leak-tight containment isolation design features and maintenance and refueling provisions. Temperature and pressure conditions inside the PCV are controlled and maintained below acceptance criteria following an accident for at least 72 hours by with RPV decay heat removal using the ICS and condensation on the PCV walls with containment heat removal using the PCCS. The NRC staff finds that the leak-tight PCV/associated systems description for BWRX-300, and GEH’s intent to provide the analyses in the future to demonstrate the compliance, as documented in NEDC-33911P are consistent with GDC 16 and are, therefore, acceptable. The NRC staff will conduct a detailed evaluation of the additional analyses to confirm that the final BWRX-300 containment design satisfies the requirements of GDC 16 during future licensing activities of the BWRX-300.

5.1.11 10 CFR Part 50, Appendix A, GDC 38

GDC 38, “Containment heat removal,” requires that a system to remove heat from the reactor containment be provided. NEDC-33911P, Section 5.1.11, “10 CFR Part 50, Appendix A, GDC 38,” describes the containment peak pressure and temperature as being limited by condensation on containment walls and containment heat removal by the PCCS using condensation and natural convection, and by RPV decay heat removal by the ICS. Heat is rejected from the containment to the reactor cavity pool located above the containment by natural circulation using water jackets covering sections of the containment shell or concentric pipes. Unisolated small breaks are not limiting for containment peak pressure or temperature. The operation of the ICS does not require offsite electric power system operation, and only requires one-time actuation using onsite Class 1E battery-backed DC power. Additionally, suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished in the event of a single failure.

NEDC-33911P describes the PCCS design, recognizing its safety function in rapidly reducing containment pressure and temperature and maintaining them at acceptably low levels during the most limiting BWRX-300 DBE, a large-break LOCA with loss of offsite power and a single active failure. The LTR specifies a rapid depressurization target of reducing the pressure to below 50 percent of the peak accident pressure within 24 hours and indicated its intent to provide the analyses to demonstrate compliance during future licensing activities. The NRC staff finds that the PCCS description and intent to provide for the rapid reduction of BWRX-300 containment peak pressure at the licensing stage, as documented in NEDC-33911P are consistent with GDC 38 and are, therefore, acceptable. The NRC staff will conduct a detailed evaluation of the additional analyses to confirm that the final BWRX-300 containment design satisfies the requirements of GDC 38 during future licensing activities of the BWRX-300.

5.1.12 10 CFR Part 50, Appendix A, GDC 39

GDC 39, "Inspection of containment heat removal system," requires that the containment heat removal system be designed to permit appropriate periodic inspection of its important components. Section 5.1.12, "10 CFR 50 Appendix A, GDC 39," of NEDC-33911P describes that the components of the PCCS used to remove heat from the containment during DBEs will be designed, fabricated, erected, and tested in accordance with ASME BPV Code, Section III, Class MC and Section XI, IWE requirements for design accessibility of welds during in-service inspection to meet GDC 16, and under an approved quality assurance program with approved control of records, as required by 10 CFR 50.55a, "Codes and standards," and GDC 1. The LTR also describes GEH's intent to explain, in the future, the means that will be used to detect and identify the location of the source of containment leakage, including the CIVs, PCCS, nonessential and closed systems, and components of the ICS and RPV isolation valves, for components of the RCPB.

The NRC staff finds that the GEH approach to the PCCS construction and accessibility, and intent regarding the provisions for detection and identification of the containment leakage source at the licensing stage, as documented in NEDC-33911P are consistent with GDC 39 and are, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm that the final BWRX-300 containment design satisfies the requirements of GDC 39 during future licensing activities of the BWRX-300.

5.1.13 10 CFR Part 50, Appendix A, GDC 40

GDC 40, "Testing of containment heat removal system," requires that the containment heat removal system be designed to permit appropriate periodic pressure and functional testing. Section 5.1.13, "10 CFR Part 50, Appendix A, GDC 40," of NEDC-33911P describes how the PCCS and its components that accomplish the containment heat removal function are designed to be periodically pressure tested as part of the overall containment leakage rate testing program to demonstrate structural and leaktight integrity, during maintenance or inservice inspection using various nondestructive methods. The NRC staff noted that the PCCS has no active components and its operation does not require offsite electric power. GEH indicated its intent to design the PCCS components with sufficient margin to meet the leaktight integrity and operational performance requirements for periodic pressure and functional testing for the range of in-containment design conditions under normal operations and DBEs, using normal and emergency power.

The NRC staff finds that GEH's description of the approach to the PCCS periodic pressure testing, and the intent at the licensing stage to demonstrate sufficient margin, as documented in

NEDC-33911P are consistent with GDC 40 and are, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm that the final BWRX-300 containment design satisfies the requirements of GDC 40 during future licensing activities of the BWRX-300.

5.1.14 10 CFR Part 50, Appendix A, GDC 41

GDC 41, “Containment atmosphere cleanup,” requires that there be systems to control fission and other products following postulated accidents. In Section 5.1.14, “10 CFR 50 Appendix A, GDC 41,” of NEDC-33911P, the statement of compliance with GDC 41 states that the BWRX-300 dry containment is nitrogen inerted and maintained during operation by a containment inerting system. Fission products, hydrogen, oxygen and other substances released from the reactor are contained within the low-leakage containment, and oxygen monitors are installed for monitoring during and after a DBA. However, Section 5.1.14 contains no specific discussion on how hydrogen produced in a severe accident will be monitored. Hydrogen and oxygen monitoring systems provide the capability to continuously measure the appropriate parameter in the BDBA environment. NEDC-33911P, Section 5.1.14, states that NEDC-33921P will address instrumentation requirements for BDB events and severe accidents. In regard to compliance with 10 CFR 50.44(c)(4), Section 5.1.2, of NEDC-33911P, supplement 1, includes the following in its statement of compliance:

> 10 CFR 50.44(c)(4), Monitoring, requires reliable equipment for monitoring oxygen and hydrogen concentrations in inerted containments during and following a significant Beyond Design Basis Accident (BDBA). The design feature of the BWRX-300 used to comply with this requirement includes the requirement for oxygen and hydrogen analyzers for monitoring containment oxygen and hydrogen concentrations. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.44(c)(4).

The NRC staff finds the BWRX-300 design, as described in NEDC-33911P, will provide the plant with the capability to monitor and control the concentration of hydrogen or oxygen in the containment atmosphere following postulated accidents to assure that containment integrity is maintained. Therefore, the NRC staff finds that the BWRX-300 design is consistent with the requirements of GDC 41 for monitoring combustible gases in the containment. However, full compliance with GDC 41 must be based on the final design of the containment and the combustible gas control and monitoring system and must consider the results for BDB events and severe accidents. NEDC-339211P does not provide design details and indicates that GEH will address BDB events and severe accidents in NEDC-33921P. The NRC staff will conduct a detailed evaluation to confirm GDC 41 is met when it receives an application for a BWRX-300.

5.1.15 10 CFR Part 50, Appendix A, GDC 42

GDC 42, “Inspection of containment atmosphere cleanup system,” requires that containment atmosphere cleanup systems be designed to permit periodic inspections of important components. In Section 5.1.15, “10 CFR 50 Appendix A, GDC 42,” of NEDC-33911P, the statement of compliance with GDC 42 states that the BWRX-300 design for the containment atmosphere cleanup systems will permit appropriated periodic inspection of important components, such as filter frames, ducts, and piping, to assure the integrity and capability of the systems.

The NRC staff finds the BWRX-300 design, as described in NEDC-33911P, will provide for the containment inerting system to be periodically tested and will provide the capability to permit periodic inspections of important components, such as filter frames, ducts, and piping. The NRC staff will conduct a detailed evaluation to confirm that appropriate inspection and testing of the containment atmosphere cleanup systems can be performed and that GDC 42 is satisfied for the specific final plant design when it receives an application for a BWRX-300.

5.1.16 10 CFR Part 50, Appendix A, GDC 43

GDC 43, "Testing of containment atmosphere cleanup systems," requires that containment atmosphere cleanup systems be designed to permit appropriate periodic pressure and functional testing. In Section 5.1.16, "10 CFR 50 Appendix A, GDC 43," of NEDC-33911P, the statement of compliance with GDC 43 states that the containment atmosphere is provided by the containment inerting system and is designed to be periodically tested.

The NRC staff finds the approach, as described in NEDC-33911P, will provide for containment inerting system to be periodically tested consistent with GDC 43 and is, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm that GDC 43 is satisfied when it receives an application for a BWRX-300.

5.1.17 10 CFR Part 50, Appendix A, GDC 50

GDC 50, "Containment design basis," requires the containment structure and associated heat removal systems be designed to accommodate the calculated pressure and temperature conditions resulting from any loss-of-coolant accident without exceeding the containment design leakage rate and with sufficient margin. Meeting GDC 50 will ensure that containment integrity is maintained under the most limiting accident conditions, thus precluding the release of radioactivity to the environment. Section 5.1.17, "10 CFR Part 50, Appendix A, GDC 50," of NEDC-33911P describes that the BWRX-300 containment design is based upon consideration of the full spectrum of postulated accidents that would result in the release of reactor coolant to the containment. These accidents are evaluated using the TRACG code to generate mass and energy release that serves as a boundary condition to GOTHIC to calculate the containment response. These accidents include liquid, steam and partial (both steam and liquid) breaks. The evaluation of the containment design NEDC-33911P is based upon enveloping the results of this range of analyses, plus provision for appropriate margin. The most-limiting short-term and long-term pressure and temperature responses are assessed to verify the integrity of the containment structure. GEH stated that the GOTHIC computer methodology for measuring containment response will be provided in LTR NEDC-33922P BWRX-300 Containment Evaluation Method, and the analyses to demonstrate compliance will be provided in the future BWRX-300 licensing activities.

GEH chose the guidance and acceptance criteria in SRP Section 6.2.1.1.A, Revision 3, issued March 2007, for the BWRX-300 design. One of the six specific areas of review in SRP Section 6.2.1.1.A pertains to the maximum expected external pressure to which the containment may be subjected. To satisfy the requirements of GDC 38 and 50, in part, with respect to the functional capability of the containment heat removal systems and containment structure under LOCA conditions, provisions would be needed to protect the containment structure against possible damage from external pressure conditions. Pursuant to the SRP, the provisions should include either a conservative structural design to assure that the containment structure can withstand the maximum expected external pressure, or interlocks in the plant protection system combined with administrative controls to prevent inadvertent operation of the containment heat removal systems. If it is designed to withstand the maximum expected external pressure, the containment should provide an adequate margin above the maximum expected external pressure to account for uncertainties in the analysis of the postulated event.

GEH provided additional information on the demonstration methodology and evaluation model to satisfy the external pressure acceptance criterion to assure the functional capability and safety margin of the BWRX-300 containment. GEH intends to identify the design pressure limit as part of the containment structural design and to providing analyses to demonstrate compliance during future licensing activities. The BWRX-300 containment design will be evaluated against the maximum expected external pressure with sufficient margin to account for uncertainties from a full spectrum of postulated accidents that would result in the release of reactor coolant to the containment.

Future licensing activities will include a containment structural evaluation of the maximum expected external pressure to demonstrate compliance with GDC 38 and 50. GEH also submitted markups to revise NEDC-33911P, Section 5.1.17 to reflect the future submittal of the design evaluation of the maximum external containment pressure, accordingly. The NRC staff finds that, with GEH's changes to NEDC-33911P, the maximum expected external pressure aspects of the BWRX-300 containment design, as incorporated in Supplement 1 to the LTR, are consistent with GDC 38 and 50 and are, therefore, acceptable. The NRC staff will conduct a detailed evaluation of the maximum expected external pressure analyses to confirm that the final BWRX-300 containment design satisfies GDC 38 and 50, in part, during future licensing activities of the BWRX-300.

5.1.18 10 CFR Part 50, Appendix A, GDC 51

GDC 51, "Fracture prevention of containment pressure boundary," requires that the reactor containment boundary shall be designed with sufficient margin to avoid brittle fracture. NEDC-33911P, Section 5.1.16 states that the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 51.

SRP Section 6.2.7 states that the PCV includes the functional capability of enclosing the reactor system and of providing a final barrier against the release of radioactive fission products. Fracture of the containment pressure boundary should be prevented for it to fulfill its design function. The BWRX-300 design must address GDC 51, "Fracture prevention of containment pressure boundary," as it relates to design considerations to ensure against fracture of the containment pressure boundary.

In SE Section 5.3.15, the NRC staff finds that the use of SRP Section 6.2.7 during a future review of a BWRX-300 license application is acceptable in that it is consistent with NRC practice. The NRC staff notes that the applicant will be expected to address GDC 51 and the embrittlement concerns to support a future NRC staff finding concerning fracture prevention of the containment pressure boundary.

5.1.19 10 CFR Part 50, Appendix A, GDC 52

GDC 52, "Capability for containment leakage rate testing," requires that the reactor containment and other equipment that may be subjected to containment test conditions be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

NEDC-33911P, Section 5.1.19, "10 CFR 50 Appendix A, GDC 52," states that the BWRX-300 containment and other equipment that may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure to comply with 10 CFR Part 50, Appendix J, and the guidance of RG 1.163, "Performance-Based Containment Leak-Test Program."

The NRC staff finds that the approach, as described in NEDC-33911P, is consistent with GDC 52 and is, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm that GDC 52 is satisfied when it receives an application for a BWRX-300.

5.1.20 10 CFR Part 50, Appendix A, GDC 53

GDC 53, "Provisions for containment testing and inspection," requires that the reactor containment be designed to permit: (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations that have resilient seals and expansion bellows.

NEDC-33911P, Section 5.1.20, "10 CFR 50 Appendix A, GDC 53," states that the BWRX-300 containment and associated penetrations have provisions for conducting individual leakage rate tests on applicable penetrations. Penetrations are visually inspected and pressure tested for leaktightness at periodic intervals, in accordance with 10 CFR Part 50, Appendix J.

The NRC staff finds that the approach, as described in NEDC-33911P, is consistent with GDC 53 and, therefore, is acceptable. The NRC staff will conduct a detailed evaluation to confirm GDC 53 is satisfied when it receives an application for a BWRX-300.

5.1.21 10 CFR Part 50, Appendix A, GDC 54

GDC 54, "Piping systems penetrating containment," requires that piping systems penetrating containment shall be provided with specific leak detection, isolation, and containment capabilities. NEDC-33911P, Section 2.2.6 states that the PCV structure, in conjunction with concurrent operation of containment isolation function(s) limit fission product leakage during and following the postulated DBA. Containment isolation function is applied to all mechanical penetrations of the PCV pressure boundary installed for piping systems and ducts carrying process or service system fluids into or out of the PCV. Containment isolation function is applied to all mechanical instrument sensing line penetrations of the PCV boundary in a manner that provides the highest reliability of maintaining instrument function while limiting potential radioactive release if an instrument line is ruptured outside the PCV boundary. The PCV design has provisions for periodic testing to measure the integrated leakage rate from the PCV structure to confirm the leak-tight integrity of the pressure boundary.

NEDC-33911P, Section 5.1.21, "10 CFR Part 50, Appendix A, GDC 54," states that the BWRX-300 containment is designed to provide the required isolation and testing capabilities. These piping systems have test connections to allow periodic leak detection as necessary to determine whether valve leakage is within acceptable limits.

The NRC staff finds that the approach, as described in NEDC-33911P, is consistent with GDC 54 and, therefore, is acceptable. The NRC staff will conduct a detailed evaluation to confirm GDC 54 is satisfied when it receives an application for a BWRX-300.

5.1.22 10 CFR Part 50, Appendix A, GDC 55

GDC 55, "Reactor coolant pressure boundary penetrating containment," requires that reactor coolant pressure boundary line penetrating containment be provided with specific isolation features. NEDC-33911P, Section 5.1.22, states that GDC 55 requires that each line that is part of the RCPB and that penetrates primary reactor containment be provided with CIVs as required in GDC 55, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis.

NEDC-33911P, Section 2.2.7.1, states that [[ ]] Small pipes for level instruments use excess flow check valves (EFCVs) to conform to the provisions of RG 1.11, "Instrument Lines Penetrating the Primary Reactor Containment."

NEDC-33911P, Section 2.2.7.1 describes that [[ ]] and the ICS provides emergency core cooling function, and the ICS RPV isolation valves, [[ ]], will close if a pipe break is detected. The closed-loop reactor coolant piping outside containment in lieu of outboard CIVs are designed for the ICS and FMCRD.

The NRC staff finds that the BWRX-300 RPV isolation valves, which isolate the RPV to preserve reactor coolant system inventory for large and medium pipe break LOCAs, [[ ]] to prevent releases from containment, consistent with GDC 55. The RPV isolation valve is attached to the RPV, not to the containment wall. In case of pipe failure outside containment, which is discussed later in this SE section, the fluid between the RPV isolation valve and the location of pipe failures may contribute to the inventory for radioactive releases, because the RPV isolation valve may not be located nearby the containment wall. In addition, the NRC staff finds that the following BWRX-300 containment penetrations: main steam line, feedwater line, level instrument, shutdown cooling suction line, and reactor water cleanup, are consistent with GDC 55, based on one inboard and one outboard isolation valves for each penetration line shown in Figures 2-5 and 2-6 of NEDC-33911P.

NEDC-33911P, Figure 2-7, indicates that the design of ICS does not comply with the explicit requirements of GDC 55, because it does not have outboard CIVs. NEDC-33911P, Section 2.2.7.1, states that [[ ]] Section 2.2.7.1 states that the ICS provides emergency core cooling function. The ICS RPV isolation valves, [[ ]], will close in a short time if a pipe break is detected. Based on this, GEH justified that the ICS containment penetrations meet GDC 55 on the "other defined basis."

The NRC staff reviewed GEH's justification that maintaining the ICS fluid line open without outboard containment isolation will provide higher reliability for performing the safety function of emergency core cooling. The ICS piping penetrating containment performs dual safety functions to deliver ECCS for core cooling and to provide isolation function for containment isolation. These two safety functions have opposite objectives of either maintaining an open flow path for ECCS or isolating flow to prevent release of fission products that may result from a postulated accident. By comparing the risk between the ECCS core cooling function and the limited consequences without the outboard CIV, the NRC staff determines that keeping flow path open without isolation is of higher risk significance. Therefore, the NRC staff finds GEH's justification to be consistent with GDC 55 on the "other defined basis." GEH indicates that for a postulated ICS piping failure inside containment, two RPV isolation valves [[ ]] and the closed-loop piping outside containment can prevent radioactive releases outside containment to satisfy GDC 55 requirement.

However, NRC staff noted that the lack of outboard CIVs could introduce the consideration of potential postulated pipe failures outside containment, because this section of high energy line piping is not included in the break exclusion areas being identified in NEDC-33911P. The postulated high energy line pipe failures in accordance with NRC BTP 3-4 must be considered.

The closing of single-failure-proof RPV isolation valve inside containment can limit, but not eliminate, the consequences outside containment. The mitigation of dynamic and environmental effects including radioactive fission product releases of the pipe failures must be considered. For a large break of ICS pipe failure outside containment, the radioactive fluid inside the closed piping, the fluid between the break location and the PCV isolation valve, and reactor coolant inventory prior to the RPV valve closure provide source term for the radioactive releases. For a small break that is below the RPV isolation setpoint could provide prolonged radioactive releases from reactor coolant. Therefore, potential pipe failures outside containment must be reviewed separately in the future licensing activities. This is identified in Section 6.0, "Limitations and Conditions," of this SE.

NEDC-33911P, Section 2.2.7.1, states that the containment penetrations of the FMCRD hydraulic lines do not have outboard CIVs based on the closed-system piping outside the PCV and RCPB isolation using internal ball check valves in the design of the drives. The CRD system and the associated hydraulic insertion line performs a safety critical function by providing the high-pressure water to implement a reactor scram as needed. The hydraulic control units (HCU) of the FMCRD meet GDC 55 on the "other defined basis."

The NRC staff finds that GEH's justification for the FMCRD is consistent with GDC 55 on the "other defined basis," because keeping the flow path open is necessary for the CRD system to perform its scram function. Comparing the risk between the degraded scram function and the radioactive fission product releases without an outboard CIV, the scram function demonstrates higher risk significance.

The NRC staff noted that the lack of an outboard CIV could introduce potential postulated pipe failures outside containment, but the consequences would be limited, not eliminated, by the closure of internal ball check valves. Potential pipe failures outside containment must be reviewed separately in the future licensing activities. This is identified in Section 6.0, "Limitations and Conditions," of this SE.

Subject to confirmation in future licensing activities, the NRC staff finds the BWRX-300 design of the CIVs is consistent with the requirements of GDC 55. Specifically, the NRC staff finds the ICS and FMCRD containment isolation meeting GDC 55 on the "other defined basis" to be acceptable. However, this design, using a closed loop outside containment instead of an outboard CIV, could introduce a postulated LOCA outside containment that must be evaluated in future licensing activities. The NRC staff will conduct a detailed evaluation to confirm that GDC 55 is satisfied when it receives an application for a BWRX-300.

5.1.23 10 CFR Part 50, Appendix A, GDC 56

GDC 56, "Primary containment isolation," requires that each line connecting directly to the containment atmosphere and penetrating containment be provided with CIVs. NEDC-33911P, Section 5.1.23, "10 CFR Part 50, Appendix A, GDC 56," describes the approach taken by the BWRX-300 design to comply with GDC 56. The BWRX-300 CIVs governed by GDC 56 that are attached directly to the containment atmosphere include the integrated leak rate testing system, the emergency purging system, the containment inerting system nitrogen supply, the process gas and radiation monitoring system, and the floor drain sump system. The integrated leak rate testing system and the emergency purging system have two manual CIVs outside containment that are normally closed. The integrated leak rate testing system and the emergency purging system CIVs are both outside containment, as they are required to be accessed for manual operations when containment access is not possible, and then only when containment integrity is not required to be automatically assured. The containment inerting system nitrogen supply has normally closed automatic CIVs inside and outside containment. The process gas and radiation monitoring system is a closed system outside containment and has normally open automatic CIVs outside containment because it is an essential system following BDB events and severe accidents. The floor drain sump line has two normally closed automatic CIVs outside containment, because it is not practicable to include an automatic CIV inside containment to allow draining all the water accumulated in the sump. However, these CIVs, being at the bottom of the containment, are not subject to damage due to external effects.

The NRC staff finds that the approach described in NEDC-33911P, Section 5.1.23, Section 2.2.7.2, and Figure 2-9 to be consistent with GDC 56 and, therefore, is acceptable. The NRC staff will conduct a detailed evaluation to confirm that GDC 56 is satisfied when it receives an application for a BWRX-300.

5.1.24 10 CFR Part 50, Appendix A, GDC 57

GDC 57, "Closed system isolation valves," requires that lines penetrating the primary containment boundary and neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere have at least one locked-closed, remote-manual, or automatic isolation valve outside containment.

NEDC-33911P, Section 5.1.24, describes the BWRX-300 design approach to complying with GDC 57. BWRX-300 "GDC 57" CIVs, which are for a closed system that penetrates primary reactor containment and are neither part of the RCPB nor connected directly to the containment atmosphere, have at least one CIV, either automatic, locked closed, or capable of remote manual operation. The BWRX-300 closed system CIVs include the pneumatic nitrogen or air system, the service and breathing air system, the quench tank supply system, the chilled water supply and return, and the demineralized water system. The pneumatic nitrogen or air system and the quench tank supply system have either normally open or normally closed automatic CIVs inside and outside containment. The service and breathing air system and demineralized water system have normally closed manual CIVs inside and outside containment. The chilled water supply and return have normally open automatic CIVs outside containment. [[ ]]

The NRC staff finds that the approach described in NEDC-33911P Section 5.1.24, Section 2.2.7.3, and Figure 2-10, is consistent with GDC 57 and, therefore, is acceptable. The NRC staff will conduct a detailed evaluation to confirm that GDC 57 is satisfied when it receives an application for a BWRX-300.

5.1.25 10 CFR Part 50, Appendix A, GDC 64

GDC 64, "Monitoring radioactivity releases," requires that means be provided for monitoring the reactor containment atmosphere and the plant environs for radioactivity that may be released from normal operations and postulated accidents. Section 5.1.25, "10 CFR 50 Appendix A, GDC 64," of NEDC-33911P states that the BWRX-300 has a process gas and radiation monitoring system that monitors radioactivity in containment for normal operations, AOOs, infrequent events, and DBAs. The NRC staff finds that the GEH indicated its intent to provide radiation monitoring in the containment atmosphere for normal operations and postulated DBEs, as documented in NEDC-33911P is consistent with GDC 64 and is, therefore, acceptable. The NRC staff will conduct a detailed technical evaluation to confirm that the final BWRX-300 containment design satisfies the requirements of GDC 64 during future licensing activities of the BWRX-300. The NRC staff will further confirm that the BWRX-300 radiation monitoring system also covers the spaces containing components for LOCA fluid recirculation, effluent discharge paths, and the plant environs for radioactivity that may be released.

5.1.26 10 CFR Part 50, Appendix J

Appendix J specifies containment leakage testing requirements, including the types of tests required to ensure the leak-tight integrity of the primary reactor containment and systems and components that penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing, and reporting requirements for each type of test.

In Section 5.1.26, "10 CFR 50 Appendix J," of NEDC-33911P, the statement of compliance with the regulatory requirement in 10 CFR Part 50, Appendix J, states that the BWRX-300 containment and other equipment that may be subjected to containment test conditions are designed so that periodic integrated leakage rate testing can be conducted at containment design pressure to comply with 10 CFR Part 50, Appendix J, and the guidance of RG 1.163.

RG 1.163 provides guidance on an acceptable performance-based leak-test program, leakage-rate test methods, procedures, and analyses that may be used to comply with the performance-based Option B in 10 CFR Part 50, Appendix J, and endorsed Nuclear Energy Institute 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0. Section 5.2.7, "Regulatory Guide 1.163," of NEDC-33911P addresses RG 1.163. In Section 5.2.7, GEH stated that the BWRX-300 design will include a containment leak test program that addresses integrated containment leakage rate (Type A tests), containment penetration leakage tests (Type B tests), and CIV leakage rates (Type C tests) and complies with 10 CFR Part 50, Appendix J, Option A or B, in accordance with RG 1.163 and GDC 52, GDC 53, and GDC 54. Type A, B, and C tests are performed before operations and periodically thereafter to assure that the leakage rates through the containment and through systems or components that penetrate containment do not exceed their maximum allowable rates. Maintenance of the containment, including repairs on systems and components penetrating the containment, is performed as necessary to maintain leakage rates at or below acceptable values. GEH concluded that the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.163.

The NRC staff finds the BWRX-300 design, as described in NEDC-33911P, would accommodate periodic integrated leakage rate testing and local leak rate test for CIVs, and containment penetrations; thus, the NRC staff finds that the design is consistent with 10 CFR Part 50, Appendix J and, therefore, is acceptable. The NRC staff will conduct a detailed evaluation to confirm that the 10 CFR Part 50, Appendix J requirements are met for the specific final plant design when it receives an application for a BWRX-300.

5.2 Regulatory Guides

5.2.1 Regulatory Guide 1.7

RG 1.7, "Control of Combustible Gas Concentrations in Containment," Revision 3, describes methods that are acceptable to the NRC with regard to control of combustible gases generated by beyond-design-basis accidents. NEDC-33911P, Section 5.2.1, states that the BWRX-300 design includes a dry, inerted containment that does not rely upon combustible gas control to maintain concentrations of hydrogen and oxygen below combustible levels and maintain containment structural integrity following a DBA. Consistency with 10 CFR 50.44 is addressed in Section 5.1.2 of this SE and, as indicated in that section, consistency with the requirements of

10 CFR 50.44(c)(1), 10 CFR 50.44(c)(3), and 10 CFR 50.44(c)(5) for beyond design basis events and severe accident management is not addressed in NEDC-33911P, but will be addressed in LTR NEDC-33921P, "BWRX-300 Severe Accident Management."

The NRC staff finds the BWRX-3000 design for control of combustible gas concentrations in containment to be consistent with the guidance in RG 1.7, "Control of Combustible Gas Concentrations in Containment," Revision 3, issued March 2007, and the acceptance criteria associated with the guidance in SRP Section 6.2.5. The NRC staff will conduct a detailed evaluation to confirm that the BWRX-300 design conforms to the guidance in RG 1.7 when it receives an application for a BWRX-300.

5.2.2 Regulatory Guide 1.11

NEDC-33911P, Section 5.2.2, "Regulatory Guide 1.11," states that, in RG 1.11, instrument lines that penetrate the primary reactor containment and that are part of the RCPB or that penetrate the primary reactor containment and connect directly to the containment atmosphere should be chosen with consideration of the importance of the following two safety functions: (1) the function that the associated instrumentation performs and (2) the need to maintain containment leaktight integrity.

NEDC-33911P states that BWRX-300 instrument lines penetrating primary reactor containment that are part of the RCPB or penetrate the primary reactor containment and connect directly to the containment atmosphere comply with Regulatory Position C.3 of RG 1.11 by providing excess flow check valves, and they also comply with the requirements of GDC 55 and GDC 56. Each line has a self-actuated excess flow check valve located outside containment, as close as practical to the containment. These check valves are designed to remain open as long as the flow through the instrument lines is consistent with normal plant operation. However, if the flow rate is increased to a value representative of a loss of piping integrity outside containment, the valves close. These valves reopen automatically when the pressure in the instrument line is reduced. The instrument lines are Quality Group B up to and including the isolation valve, located and protected to minimize the likelihood of damage, protected or separated to prevent the failure of one line from affecting the others, accessible for inspection, and not so restrictive that the response time of the connected instrumentation is affected. GEH stated that the BWRX-300 design therefore conforms to the guidance, including regulatory positions in RG 1.11.

The NRC staff finds that the BWRX-300 design is consistent with the guidance in RG 1.11 for the instrument lines penetrating the primary reactor containment 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 1.11 when it receives an application for a BWRX-300.

5.2.3 Regulatory Guide 1.84

NEDC-33911, Revision 0, Supplement 1, in Section 5.2.3, "Regulatory Guide 1.84," states that RG 1.84, "Design, Fabrication and Materials Code Case Acceptability, ASME Section III," Revision 38, describes 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 into 10 CFR 50.55a. NEDC-33911P, Revision 0, Supplement 1, states that the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.84, Revision 38, and uses the guidance in conformance to RG 1.84, Revision 33, as described in ESBWR Design Control Document (DCD) Tier 2, 26A6642AD, Revision 10, Section 1.9.2, Table 1.9-21, and Table 5.2-4. ASME BPV Code Case N-782 is also applied to the BWRX-300 design. Code Case N-782 endorses the use of the Edition and Addenda of ASME BPV Code, Section III, Division 1, as an alternative to the requirements of Code paragraphs NCA-1140(a)(2)(a) and NCA-1140(a)(2)(b). Justification for application of Code Case N-782 will be provided in the BWRX-300 PSAR or future licensing activities.

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 is received.

5.2.4 Regulatory Guide 1.141

NEDC-33911P, Section 5.2.4, "Regulatory Guide 1.141," states that RG 1.141 recommendations for the containment isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors, as specified in American National Standards Institute N271-1976, "Containment Isolation Provisions for Fluid Systems," are generally acceptable and provide an adequate basis for use. Section 2.2.8, "Passive Containment Cooling System," Section 5.1.22, "10 CFR 50 Appendix A, GDC 55," Section 5.1.23, "10 CFR 50 Appendix A, GDC 56," and Section 5.1.24, "10 CFR 50 Appendix A, GDC 57," of this LTR describe how the design of the BWRX-300 CIVs complies with the requirements of GDC 55, GDC 56, and GDC 57. GEH states that the BWRX-300 design therefore conforms to the guidance, including regulatory positions in RG 1.141.

The NRC staff finds that the BWRX-300 design is consistent with the guidance in RG 1.141 for the containment isolation provisions 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 1.141 when it receives an application for a BWRX-300.

5.2.5 Regulatory Guide 1.147

NEDC-33911P, Revision 0, Supplement 1, in Section 5.2.5, "Regulatory Guide 1.147," states that RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 19, lists the ASME BPV Code, Section XI, Code Cases that the NRC has approved for use as voluntary alternatives to the mandatory ASME BPV Code provisions that are incorporated by reference into 10 CFR Part 50. GEH states that this applies to reactor licensees subject to 10 CFR 50.55a. The ASME BPV Code, Section XI, Code Cases in RG 1.147 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 the requirements of 10 CFR 52.79(a)(11), which requires the Final Safety Analysis Report to include "a description of the program(s), and their implementation, necessary to ensure that the systems and components meet the requirements of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants in accordance with 50.55a of this chapter." In 10 CFR 50.55a(a), the NRC references the latest editions and addenda of ASME BPV Code, Section XI, that the agency has incorporated by reference.

Section 4.1.3, "10 CFR 50.55a," of NEDC-33910P on RPV isolation and overpressure protection for the BWRX-300 describes how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a. The requirements of ASME BPV Code, Section XI, Division 1, specifically apply during construction and operation activities of nuclear power plants for performance of inservice inspection (ISI) activities. The GEH design process and associated administrative controls consider operating plant conformance to RG 1.147 guidance in performing examinations, inspections and tests of installed systems and components. GEH indicates that RG 1.147 guidance is incorporated in the design review process to support plant operation and maintenance best practices. Therefore, compliance with the requirements of 10 CFR 50.55a for the conduct of ISI activities, including the use of ASME BPV Code, Section XI, Code Cases accepted in RG 1.147 where necessary, is to be demonstrated during future licensing activities. The guidance of RG 1.147 will be applied to the BWRX-300 design phase in considering the operating plant provisions to perform examinations, inspections and testing of installed systems and components to support plant operation and best maintenance practices to meet the requirements of 10 CFR 50.55a.

The NRC staff finds the plans to apply the Code Cases identified in RG 1.147 to be acceptable for the design phase of the BWRX-300. The NRC staff will conduct a detailed evaluation of Code Cases identified in RG 1.147 that are applied during the design phase of the BWRX-300 when an application for a BWRX-300 is received.

5.2.6 Regulatory Guide 1.155

RG 1.155, "Station Blackout," Revision 0, issued August 1988, describes methods for complying with 10 CFR 50.63, "Loss of all alternating current power," which requires nuclear power plants to be capable of coping with an SBO for a specified duration, so that SSCs important to safety continue to function. NEDC-33911P, Section 5.2.6, "Regulatory Guide 1.155," states that the BWRX-300 is designed to safely shut down without AC power. In case of an SBO, safety-grade control power, closure, and position indication provide safety related CIV position indication and closure. GEH stated that the BWRX-300 design, therefore, conforms to the guidance, including regulatory positions in RG 1.155.

The NRC staff finds that the BWRX-300 design is consistent with the guidance in RG 1.155 for an SBO 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 1.155 when it receives an application for a BWRX-300.

5.2.7 Regulatory Guide 1.163

Section 5.2.7 "Regulatory Guide 1.163," of NEDC-33911P addresses RG 1.163. In Section 5.2.7, GEH stated that the BWRX-300 design will include a containment leak test program that addresses integrated containment leakage rate (Type A tests), containment penetration leakage tests (Type B tests), and CIV leakage rates (Type C tests) and complies with 10 CFR Part 50, Appendix J, Option A or B, in accordance with RG 1.163 and GDC 52, GDC 53, and GDC 54. Type A, B, and C tests are performed before operations and periodically thereafter to assure that the leakage rates through the containment and through systems or components that penetrate containment do not exceed their maximum allowable rates. Maintenance of the containment, including repairs on systems and components penetrating the containment, is performed as necessary to maintain leakage rates at or below acceptable values. GEH concluded that the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.163.

The NRC staff finds the BWRX-300 design, as described in NEDC-33911P, would accommodate periodic integrated leakage rate testing and local leak rate test for CIVs, and containment penetrations; thus, the NRC staff finds the design to be consistent with 10 CFR Part 50, Appendix J and is, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm that the 10 CFR Part 50, Appendix J meets the requirements for the specific final plant design when it receives an application for a BWRX-300.

5.2.8 Regulatory Guide 1.192

NEDC-33911P, Revision 0, Supplement 1, states that RG 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code,” Revision 3, lists Code Cases associated with the ASME Operation and Maintenance of Nuclear Power Plants, Division 1, OM Code: Section IST (OM Code) that the NRC has approved for use as voluntary alternatives to the mandatory ASME OM Code provisions that are incorporated by reference into 10 CFR Part 50. GEH states that this applies to reactor licensees subject to 10 CFR 50.55a. These ASME OM Code Cases in RG 1.192 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, the requirements of 10 CFR 50.55a(f) which requires, in part, that Class 1, 2, and 3 components and their supports meet the requirements of the ASME OM Code or equivalent quality standards, and the requirements of 10 CFR 52.79(a)(11), which requires the Final Safety Analysis Report to include a description of the program(s), and their implementation, necessary to ensure that the systems and components meet the requirements of the ASME BPV Code and the ASME OM Code in accordance with 10 CFR 50.55a. In 10 CFR 50.55a(a)(1)(iv), the NRC specifies the latest editions and addenda of ASME OM incorporated by reference in 10 CFR 50.55a.

Section 4.1.3 of NEDC-33910P describes how the design of the RCPB, including the ICS and RPV isolation valves, complies with the requirements of 10 CFR 50.55a. The requirements of ASME OM Code specifically apply during operation and maintenance activities of nuclear power plants for performance of inservice testing (IST) activities. GEH design process and associated administrative controls consider operating plant conformance to RG 1.192 guidance in performing examinations, inspections and tests of installed systems and components. GEH indicates that RG 1.192 guidance is incorporated in the design review process to support plant operation and maintenance best practices. Therefore, compliance with the requirements of 10 CFR 50.55a for the conduct of IST activities, including the use of ASME OM Code Cases endorsed in RG 1.192 where necessary, is to be demonstrated during future licensing activities.

Section 5.2.8 in NEDC-33911P, Revision 0, Supplement 1, states that the guidance of RG 1.192 will be applied to the BWRX-300 design phase in considering the operating plant provisions to perform examinations, inspections and testing of installed systems and components to support plant operation and best maintenance practices to meet the requirements of 10 CFR 50.55a.

The NRC staff finds the plans for the use of RG 1.192 to be acceptable for the design phase of the BWRX-300 because the Code Cases are voluntary alternatives to the ASME OM Code. The NRC staff will conduct a detailed evaluation if any of the Code Cases identified in RG 1.192 are applied during the design phase of the BWRX-300 when an application for a BWRX-300 is received.

5.2.9 Regulatory Guide 1.203

RG 1.203, “Transient and Accident Analysis Methods,” describes a multi-step process for developing and assessing evaluation models to analyze transient NPP response during the postulated DBEs. NEDC-33911P, Sections 3.4.2, 3.4.2.1, 3.5, and 3.6, describe the GEH application of RG 1.203, Element 1, to the BWRX-300 evaluation model development, up to the GOTHIC PIRT development step. However, GEH has removed the GOTHIC PIRT related information from NEDC-33911P, Revision 0, as reflected in Supplement 1 (ML20248H570) to the LTR and plans to discuss the PIRT phenomena and provide the justification for their rankings in NEDC-33922P. The NRC staff will conduct a detailed evaluation of the GOTHIC PIRT, along with the other details of the TRACG/GOTHIC methodology, to confirm that all phenomena related to the containment evaluations for the design basis events are covered in TRACG and GOTHIC. The applicant plans to present the remaining elements of RG 1.203, including the demonstration analyses and the specifics of the application method, in LTR NEDC-33922P, "BWRX-300 Containment Evaluation Method." The staff concludes that the applicant's approach to qualify the BWRX-300 containment evaluation methodology, as defined in NEDC-33911P, Supplement 1, is consistent with RG 1.203, Element 1, and, therefore, is acceptable. The NRC staff will review the detailed descriptions of the remaining elements of Reg Guide 1.203 for a conservative GEH analysis during its review and evaluation of LTR NEDC-33922P. The NRC staff will conduct a detailed evaluation to confirm that the evaluation and analysis methods satisfy the NRC regulations during future licensing activities.

5.3 NUREG-0800 Standard Review Plan Guidance

Section 1.2, "Scope," of NEDC-33911P states that the LTR may be "referenced in future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant for requesting a CP and OL under 10 CFR 50 or a COL under 10 CFR 52." This implies that the scope of the present containment performance LTR regulatory review is intended to cover future applications pursuant to both 10 CFR Part 50 and 10 CFR Part 52. LTR Section 5.1, "10 CFR 50 Regulations," specifically addresses all the applicable SRP-referenced regulations in 10 CFR Part 50, including the GDC, and describes how these requirements will be met.

SE subsections in Section 5.3 discuss this staff review. The NRC staff reviewed the LTR in accordance with applicable SRP guidance and associated regulations, including those regulations listed in NEDC 33911P, Sections 5.1.1 through 5.1.26. The NRC staff noted that the LTR does not address the regulatory requirements in 10 CFR 52.47(b)(1) and 10 CFR 52.80(a) identified in these SRP Sections. If 10 CFR Part 52 is chosen for the future license application, the DCA and COL application must address these two additional regulatory requirements pertaining to ITAAC and COL information.

5.3.1 Standard Review Plan 3.6.2

NEDC-33911P, Revision 0, Supplement 1, includes Section 5.3.1, "Standard Review Plan 3.6.2," to describe the application of SRP Section 3.6.2 to the BWRX-300. SRP Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping", Revision 3, dated December 2016, states that dynamic effects of postulated accidents, including the appropriate protection against the dynamic effects of postulated pipe ruptures in accordance with the requirements of 10 CFR 50.55a, Appendix A, GDC 4 shall be considered in the design of structures, systems and components. GEH states that this SRP section provides guidance for ensuring that the appropriate protection of SSCs relied upon for safe shutdown or to mitigate the consequences of postulated pipe rupture are considered in the design.

Section 5.3.1 in NEDC-33911P states that the BWRX-300 containment isolation system SSCs will conform to the guidance of the SRP to meet the pertinent GDC 4 requirements. GEH stated that the design of the containment structure, component arrangement, pipe runs, pipe whip restraints, and compartmentalization will be done in concert with the acknowledgement of protection against the dynamic effects associated with a pipe break event. Analytically sized and positioned pipe whip restraints will be engineered to preclude damage based upon break evaluation. Moreover, in NEDC 33911P, Section 3.1 and Section 5.1.7, GEH stated that the BWRX-300 design will consider all of the dynamic effects resulting from postulated high-energy pipe breaks, including the effects of jet loads, pipe whipping, missiles, and discharging fluids in the containment design and associated piping, valves, penetrations, and instrument lines in future licensing activities. GEH also stated that a complete description of compliance with SRP Section 3.6.2 and the associated BTP 3-4, using many of the assumptions from ESBWR DCD Section 3.6.1.1 to determine the appropriate protection requirements for protection against dynamic effects, will be provided in future licensing activities.

The NRC staff finds that the design criteria of the BWRX-300 containment isolation system SSCs as described above in NEDC-33911P, are acceptable because the design criteria are consistent with the NRC staff's guidance in SRP Section 3.6.2 and associated BTP 3-4 and are therefore consistent with the requirements of 10 CFR 50.55a, Appendix A, GDC 4. The NRC staff will conduct a detailed evaluation to confirm that the final design of the containment isolation system SSCs satisfies the NRC regulations when it receives an application for a BWRX-300.

5.3.2 Standard Review Plan 3.9.6

Section 2.2.7 in NEDC-33911P describes the various CIVs used in the BWRX-300. The capability of CIVs to perform their design-basis functions is safety significant because it provides assurance that the containment of the BWRX-300 can be safely isolated and can prevent a radioactive release that exceeds regulatory requirements to the environment.

GEH described various aspects of the CIV design planned for the BWRX-300. For example, GEH stated that it does not anticipate any first-of-a-kind features for the BWRX-300 CIVs. The valve and actuator types, as well as valve size, will be addressed in the detailed design of the BWRX-300 and will be specified during future licensing activities. Further, GEH stated that the qualification of the CIVs, such as compliance with ASME Standard QME-1 as endorsed in RG 1.100, will be addressed in the detailed design and procurement process for the CIVs and will be specified during future licensing activities. GEH also stated that the detailed design of the CIVs will address lessons learned from international operating experience. GEH stated that the detailed system design layout of the CIVs will address the accessibility for IST activities in accordance with 10 CFR 50.55a. GEH indicated that the BWRX-300 CIVs are expected to have specific leak criteria under the Category A requirements of ASME OM Code.

NEDC-33911P, Revision 0, Supplement 1, specifies in Section 2.2.7 that valve qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as endorsed in RG 1.100, will be addressed in the detailed design and procurement process of the valves and will be specified in future licensing activities. The NRC staff notes that RG 1.100, Revision 4, issued May 2020, accepts ASME Standard QME-1-2017, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," with conditions. The NRC staff expects an applicant to specify the use of the most recent edition of ASME Standard QME-1 as accepted in RG 1.100 for the BWRX-300.

NEDC-33911P, Revision 0, Supplement 1, includes updated provisions with respect to CIV design features. For example, NEDC-33911P includes provisions in Section 2.2.7 for certain CIVs to fail in the closed position with valve actuators designed to maintain the valves closed by positive mechanical means. NEDC-33911P has provisions in Section 5.5, "Operational Experience and Generic Communications," for considering generic issues, operational experience, and generic communications related to CIVs. NEDC-33911P includes specific sections to address the applicability of RG 1.84, RG 1.141, “Containment Isolation Provisions for Fluid Systems,” and RG 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code,” with respect to containment design and performance.

NEDC-33911P, Revision 0, Supplement 1, includes Section 5.3.2, “Standard Review Plan 3.9.6,” to describe the application of SRP Section 3.9.6 to the BWRX-300. Section 5.3.2 specifies that the BWRX-300 CIVs 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 CP under 10 CFR Part 50 or a DCA under 10 CFR Part 52. Section 5.3.2 indicates that the requirements of 10 CFR 50.55a will be implemented during the detailed design of the safety related components for containment isolation. Section 5.3.2 concludes that SRP Section 3.9.6 provides adequate guidance to use during a future review of a 10 CFR Part 52 DCA for a BWRX-300, if pursued, or for future 10 CFR Part 50 license applications.

The NRC staff finds that the CIV design features as described in NEDC-33911P are consistent with the NRC regulations in Appendix A and Appendix B to 10 CFR Part 50, and in 10 CFR 50.55a, and are, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm that the final design of the CIVs satisfies the NRC regulations when it receives an application for a BWRX-300.

5.3.3 Standard Review Plan 6.2.1

Section 5.3.3, “Standard Review Plan 6.2.1,” of NEDC-33911P describes the application of SRP Section 6.2.1 to the BWRX-300 containment functional design. The BWRX-300 containment design is affected by the guidance provided in six SRP sections; SRP Section 6.2.1.1.A, “PWR Dry Containments, Including Subatmospheric Containments,” SRP Section 6.2.1.1.C, “Pressure Suppression Type BWR Containments,” SRP Section 6.2.1.2, “Subcompartment Analysis,” SRP Section 6.2.1.3, “Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs),” SRP Section 6.2.1.4, “Mass and Energy Release Analysis for Postulated Secondary System Pipe Rupture,” and SRP Section 6.2.1.5, “Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies.”

Section 5.3.3 of NEDC-33911P describes the design features of the BWRX-300 containment, that is, a nitrogen-inerted, dry, underground steel or reinforced concrete PCV, with no suppression pool. Containment heat is removed passively by the PCCS for design basis events, and by fan coolers for normal operations. The ICS pool and reactor cavity pool are located above the containment. The design and sizing of the BWRX-300 containment systems will be largely based on the pressure and temperature conditions that result from release of the reactor coolant during the large-break LOCA, which is the limiting DBE. Section 5.3.3 of NEDC-33911P mentioned the regulations in 10 CFR Part 50, Appendix K, stating: “The basic functional design requirements for containment are given in GDC 4, GDC 16, GDC 50, and 10 CFR 50, Appendix K.” However, the NRC staff noted that the LTR does not include a compliance statement for 10 CFR Part 50, Appendix K, as it did for GDC 4, GDC 16, and GDC 50.

To address Appendix K compliance, GEH stated that the mass and energy release rates used in the BWRX-300 containment analyses will be calculated accounting for all applicable sources of energy required for consideration in 10 CFR Part 50, Appendix K, using the assumptions and correlations similar to those used in LTR NEDC-33083P-A, “TRACG Application for ESBWR,” dated April 8, 2005. GEH stated it will describe these applicable energy sources, the correlations, and the conservative biases in detail in the forthcoming NEDC-33922P. Appendix K to 10 CFR Part 50 specifies analysis requirements for 10 CFR 50.46 acceptance criteria for the emergency core cooling system (ECCS) for light-water reactor compliance. GEH also stated in the response that NEDC-33910P, Revision 0, Supplement 1, Section 4.1.2, "10 CFR 50.46," demonstrates compliance with 10 CFR 50.46. The NRC staff finds that the details provided to account for all applicable energy sources to the mass and energy release for the BWRX-300 containment design as described in the LTR are consistent with 10 CFR Part 50, Appendix K, and are therefore, acceptable. The NRC staff will perform a detailed evaluation of the mass and energy release analyses for postulated high-energy pipe breaks to confirm that the final BWRX-300 containment design satisfies the required applicable aspects of 10 CFR Part 50, Appendix K, during future licensing activities of the BWRX-300.

5.3.4 Standard Review Plan 6.2.1.1.A

GEH identified SRP Section 6.2.1.1.A, "PWR Dry Containments, including Subatmospheric Containments," Revision 3, for use because the guidance and associated acceptance criteria contained therein are applicable to the BWRX-300 design. NEDC-33911P, Section 5.3.4, identified six review areas of SRP Section 6.2.1.1.A that are applicable to the BWRX-300 containment design.

NEDC-33911P states that the BWRX-300 design has a dry, nitrogen-inerted containment with no suppression pool; rather, it uses a PCCS to mitigate the dynamic effects of DBEs. In this respect, it is similar to several PWR containment designs. The NRC staff noted that the BWRX-300 containment does not employ an ice condenser or a pressure-suppression pool for maintaining containment pressure and temperature during the DBEs; thus, SRP Section 6.2.1.1.B, "Ice Condenser Containments," and SRP Section 6.2.1.1.C, "Pressure Suppression Type BWR Containments," are not applicable. While SRP Section 6.2.1.1.A better reflects the design of the BWRX-300, some of its portions are not applicable to the BWRX-300 design; specifically: (1) the BWRX-300 does incorporate the use of an ECCS inasmuch as the ICS system maintains RPV pressure at acceptable levels during any DBA and the PCCS maintains containment pressure during any DBA; (2) there are no subcompartments in containment with large bore high energy lines that could affect the dynamics of energy line breaks; and (3) there are no secondary systems utilized in the BWRX-300 design. As the areas of review, review interfaces, acceptance criteria, review procedures, evaluation findings, and references are applicable to the BWRX-300 based on the design description and design requirements discussed in Sections 2.0 through 4.0 of this LTR, the NRC staff agrees that the SRP Section 6.2.1.1.A guidelines are applicable to the BWRX-300 containment design.

5.3.5 Standard Review Plan 6.2.1.1.C

SRP Section 6.2.1.1.C provides guidance for evaluating the temperature and pressure condition effects in the drywell and wetwell of BWR containments incorporating a suppression pool. GEH states that the BWRX-300 design does not employ a drywell and wetwell incorporating a suppression pool. Therefore, subject to confirmation during licensing activities for a final design, the NRC staff agrees that the SRP Section 6.2.1.1.C guidelines are not applicable to the BWRX-300 containment design.

5.3.6 Standard Review Plan 6.2.1.2

SRP Section 6.2.1.2, "Subcompartment Analysis," Revision 3, includes guidance for verifying compliance with the requirements of GDC 4 and GDC 50 for subcompartments within primary containment that house high-energy piping and would limit the flow of fluid to the main containment volume in the event of a pipe rupture within the volume. In NEDC-33911P, Section 5.3.6, GEH stated that "The BWRX-300 design does not include any subcompartments with large bore high energy lines that would limit the flow of fluid to the containment in the event of a pipe rupture.

Because the BWRX-300 containment does not include subcompartments containing large high energy pipes, subcompartment pressurization and acoustic loads resulting from pipe breaks in subcompartments for the purposes of structural integrity do not apply to the BWRX-300 containment. Subcompartments are used in the model only to the extent to calculate containment atmosphere mixing. Therefore, the acceptance criteria associated with these guidelines are met without the need for specific analyses for the BWRX-300 design."

The BWRX-300 containment design is at the conceptual stage and, as described in NEDC-33911P Sections 2.2 and 2.2.5, does include as subcompartments the volume below the RPV, the space between RPV and the biological shield, and the containment head area above the refueling bellows. For the NRC staff to conclude that the SRP Section 6.2.1.2 review guidance for subcompartment pressurization would not apply to the BWRX-300 containment, GEH needs to demonstrate that the functional design of the subcompartments meets GDC 4 and GDC 50 requirements. This would require demonstrating that no significant pressure differentials are created across the subcompartment walls of the final containment design under the postulated DBAs, caused by line breaks either inside or outside the subcompartments.

To address this issue, GEH discussed how the design of the BWRX-300 containment subcompartments will comply with the safety regulations. GEH explained that the containment shell, as well as its internal structures and components, will be evaluated for the dynamic effects of jet impingement, missiles, postulated high-energy pipe breaks, discharging fluids, and pipe whipping as part of the detailed design, in accordance with GDC 4 and GDC 50. GEH will evaluate all these dynamic effects for compliance with the design requirements of GDC 4 and GDC 50. GEH also stated that the forthcoming NEDC-33922P will provide the detailed demonstration case results that show no significant pressure differential across subcompartment walls due to large breaks outside the subcompartments. The NRC staff finds that, with GEH's proposed changes to Section 3.1 and Section 5.1.7 of NEDC-33911P, the subcompartments' safety aspects of the BWRX-300 containment design as described in the LTR are consistent with GDC 4 and 50 and are, therefore, acceptable. The NRC staff will conduct a detailed evaluation of the design and analyses of pressure differential across the subcompartment walls due to postulated high-energy pipe breaks to confirm that the final BWRX-300 containment design satisfies GDC 4 and 50, in part, during future licensing activities of the BWRX-300.

5.3.7 Standard Review Plan 6.2.1.3

GEH identified SRP Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)," Revision 3, for use because the acceptance criteria, review areas/interfaces/procedures, and references are applicable to the BWRX-300 based on the design description and requirements discussed in this LTR. Mass and energy (M&E) release to the BWRX-300 containment will be calculated by using GEH's TRACG code for RPV neutronics and thermal-hydraulics calculations, with the modeling and plant parameters biased to account for the uncertainties. Containment back pressure and ingress of steam/gas mixture are specified as boundary conditions to the TRACG model. M&E release data are needed by the containment and subcompartment safety analyses to show that the BWRX-300 containment design satisfies the GDC 50 acceptance criteria, with sufficient margin to accommodate the calculated peak pressure and temperature resulting from the limiting M&E release event. The NRC staff will conduct a detailed evaluation of the applicability of the previous submittals of TRACG containment/LOCA models and qualification to the BWRX-300, during its future licensing activities.

5.3.8 Standard Review Plan 6.2.1.4

SRP Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," provides guidance for the review of the mass and energy release for secondary system pipe ruptures to evaluate the containment and subcompartment functional design for compliance with GDC 50 for the postulated PWR secondary system pipe ruptures. The BWRX-300 design does not employ any secondary systems for feedwater or steam production. Therefore, subject to confirmation during licensing activities for a final design, the NRC staff agrees that the acceptance criteria associated with SRP Section 6.2.1.4 are not applicable to the BWRX-300 design.

5.3.9 Standard Review Plan 6.2.1.5

SRP Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," provides guidance for verifying compliance with 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," for the performance of the ECCS in a PWR to reflood the core following a LOCA and the associated analyses of the minimum containment pressure possible until the core is reflooded. The BWRX-300 design does incorporate the use of ECCS design functions, inasmuch as the ICS maintains RPV pressure at acceptable levels during any DBE, as described in LTR NEDC-33910P, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection, Supplement 2 dated June 22, 2020. The NRC staff noted that the BWRX-300 design does not use the ECCS for mitigating containment thermal-hydraulic response, and the containment pressure is maintained by the PCCS during the DBEs. The NRC staff also considered GEH's assertion that the containment pressure does not affect the performance of the ECCS design functions for large-break LOCAs. The NRC SE (ADAMS Accession No. ML20176A449) on NEDC-33910P addresses consistency with the 10 CFR 50.46(a)(1)(i) acceptance criterion associated with SRP Section 6.2.1.5, with respect to an acceptable ECCS evaluation model that the NRC staff will confirm during future BWRX-300 licensing activities.

5.3.10 Standard Review Plan 6.2.2

GEH plans to follow SRP Section 6.2.2, "Containment Heat Removal Systems," Revision 5, as review guidance for containment heat removal under post-DBE conditions to ensure conformance with the requirements of GDC 38, GDC 39, GDC 40, and 10 CFR 50.46(b)(5). GDC 38, GDC 39, and GDC 40 involve demonstrating the capability of BWRX-300 containment heat removal systems to rapidly reduce containment pressure and temperature following the most severe LOCA with loss of offsite power (LOOP), assuming a single active failure and maintaining these indicators at acceptably low levels; as well as inspection and testing requirements.

GEH analyzed the following specific areas of review listed in SRP Section 6.2.2: (1) analyses of the consequences of single component malfunctions, (2) analyses of the available net positive suction head (NPSH) to the ECCS and containment heat removal system pumps, (3) analyses of the heat removal capability of the spray water system, (4) analyses of the heat removal capability of the residual heat removal (RHR) and fan cooler heat exchangers, (5) potential for surface fouling and flow blockage of the fan cooler, recirculation, and RHR heat exchangers and the effect on heat exchanger performance, (6) design provisions and proposed program for periodic inservice inspection and operability testing of each system or component, (7) design of sumps and water sources for ECCS and containment spray system (CSS) performance, and (8) effects of accident-generated debris, including an assessment for potential loss of long-term cooling capability resulting from LOCA-generated and latent debris. As described in the LTR, the BWRX-300 design does not use a spray water system, ECCS, or a sump to actively remove heat from the containment, and rather uses a PCCS. The NRC staff agrees that the above-listed review areas 2, 3, 7, and 8 are not applicable, as the BWRX-300 design does not use any active containment heat removal pumps, sprays, sumps, or ECCS. The NRC staff determined that GEH would therefore not have to address Generic Safety Issue 191, "Experimental Studies of Loss-of-Coolant-Accident-Generated Debris Accumulation and Head Loss with Emphasis on the Effects of Calcium Silicate Insulation" (NUREG/CR-6874, LA-UR-04-1227), on the effects of accident-generated debris.

Under 10 CFR 50.46(b)(5), the NRC requires that, after any calculated successful initial operation of the ECCS, the calculated core temperature be maintained at an acceptably low value and decay heat be removed for the extended period of time required by the long-lived radioactivity remaining in the core. As described by GEH in the LTR, the NRC staff confirmed that NEDC-33910P addresses BWRX-300 conformance to the requirements of 10 CFR 50.46(b)(5) for long-term core cooling. According the SE for NEDC-33910P (ADAMS Accession No. ML20176A449), the LTR identifies the BWRX-300 acceptance criteria to address 10 CFR 50.46(b)(5) and, therefore, no alternative approach, exception, or exemption from these requirements is required. As documented in the SE, the NRC staff found that the GEH's approach to meeting the regulation, as described in NEDC-33910P, is consistent with 10 CFR 50.46(b)(5) and is, therefore, acceptable. GEH will provide additional analysis to demonstrate compliance with the BWRX-300 acceptance criteria for long-term cooling, which may include the use of nonsafety-related equipment and operator actions, during future licensing activities. The NRC staff will conduct a detailed evaluation of the additional analysis to confirm that the final BWRX-300 design satisfies 10 CFR 50.46(b)(5) during future licensing activities of the BWRX-300.

5.3.11 Standard Review Plan 6.2.3

NEDC-33911P, Section 5.3.11, "Standard Review Plan 6.2.3," states that the BWRX-300 design does not use a secondary containment or dual containment. Therefore, the acceptance criteria associated with these guidelines are not applicable for the BWRX-300 design.

Subject to confirmation during licensing activities for a final design, the NRC staff has determined that the guidance in SRP Section 6.2.3 for the secondary containment functional design is not relevant to the BWRX-300, and the final design does not need to address it.

5.3.12 Standard Review Plan 6.2.4

SRP Section 6.2.4, "Containment Isolation System," Revision 3, issued March 2007, describes the regulatory requirements for the containment isolation systems. NEDC-33911P addresses containment isolation in multiple sections of the LTR. NEDC-33911P, Section 5.3.12, "Standard Review Plan 6.2.4," states that GEH recommends that the existing SRP Section 6.2.4 be used during future review of a 10 CFR Part 52 DCA for a BWRX-300 if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR Part 50 license applications. In addition, NEDC-33911P, Section 5.3.12, states that the design of isolation valves for lines penetrating containment follows the requirements of GDC 1, GDC 4, GDC 16, GDC 54, GDC 55, GDC 56, and GDC 57. NEDC-33911P discusses compliance with these GDCs in Section 2.2.6, "Containment Penetrations," Section 2.2.7, "Containment Isolation Valves," Section 5.1.5, "10 CFR 50 Appendix A, GDC 1," Section 5.1.7, "10 CFR 50 Appendix A, GDC 4," Section 5.1.10, "10 CFR 50 Appendix A, GDC 16," Section 5.1.21, "10 CFR 50 Appendix A, GDC 54," Section 5.1.22, "10 CFR 50 Appendix A, GDC 55," Section 5.1.23, "10 CFR 50 Appendix A, GDC 56," and Section 5.1.24, "10 CFR 50 Appendix A, GDC 57," respectively.

NEDC-33911P, Section 2.2.7.1, states that the automatic CIVs outside containment are not required to be fast closing because there is no credible scenario in which fission products can be released to the containment within a few hours of a DBA. As a result of the fast closing RPV isolation valves in conjunction with the large capacity of the ICS, [[ ]]. NEDC-33911P, Section 2.2.7.1, clarifies that the closure time of the outboard containment automatic CIVs will be established to assure that isolation occurs prior to the first fission product release and will be evaluated in source term evaluations in future licensing activities.

Subject to confirmation in future licensing activities, the NRC staff finds the above clarification on evaluating outboard CIV closure time acceptable, because valve closure time is to be established based on fission product release and source term evaluations, consistent with the guidance in SRP Section 6.2.4, Paragraph I, Item1.E regarding the basis for selection of closure times of isolation valves.

The NRC staff review of the containment penetrations pertaining to the applicable regulations is described in the corresponding SE Section 2.2.6, "Containment Penetrations," Section 2.2.7, "Containment Isolation Valves," Section 5.1.5, "10 CFR Part 50, Appendix A, GDC 1," Section 5.1.6, "10 CFR Part 50, Appendix A, GDC 2," Section 5.1.7, "10 CFR Part 50, Appendix A, GDC 4," Section 5.1.21, "10 CFR 50 Appendix A, GDC 54," Section 5.1.22, "10 CFR 50 Appendix A, GDC 55," Section 5.1.23, "10 CFR 50 Appendix A, GDC 56," and Section 5.1.24, "10 CFR 50 Appendix A, GDC 57," respectively.

5.3.13 Standard Review Plan 6.2.5

In accordance with SRP Section 6.2.5, the NRC staff reviewed the BWRX-300 containment design for compliance with 10 CFR 50.44. Specifically, the NRC staff reviewed the report to determine whether the proposed containment design will include: (1) the capability to mix the combustible gases with the containment atmosphere and prevent high concentrations of combustible gases in local areas, (2) the capability to monitor combustible gas concentrations within containment and for inerted containments, and (3) the capability to reduce combustible gas concentrations within containment by suitable means such as igniters.

NEDC-33911P, Section 5.3.13, addresses the functional capability of the BWRX-300 combustible gas control systems to ensure that the system maintains containment integrity. This section of the LTR states that the BWRX-300 design includes a dry, inerted containment that does not rely upon combustible gas control to keep concentrations of hydrogen and oxygen below combustible levels and maintain containment structural integrity following a DBA. NEDC-33911P does not cover BDB events and severe accident compliance; however, NEDC-33911P, Section 5.3.13, states that NEDC33911P does not address BDB events and severe accidents or compliance with the requirements of 10 CFR 50.44(c)(1), 10 CFR 50.44(c)(3) and 10 CFR 50.44(c)(5), but NEDC-33921P will address them.

The NRC staff finds that the NEDC-33911P approach for combustible gas control for the BWRX300 is consistent with the requirements of 10 CFR 50.44 and, therefore, is acceptable for normal operating and DBA conditions. However, GEH indicated that while NEDC-33911P does not address BDB events and severe accidents or compliance with the requirements of 10 CFR 50.44(c)(1), 10 CFR 50.44(c)(3) and 10 CFR 50.44(c)(5), NEDC-33921P will address them. Therefore, the NRC staff will conduct a detailed evaluation to confirm compliance with 10 CFR 50.44(c)(1), 10 CFR 50.44(c)(3) and 10 CFR 50.44(c)(5) when it reviews NEDC-33921P.

5.3.14 Standard Review Plan 6.2.6

SRP Section 6.2.6, "Containment Leakage Testing," Revision 3, provides guidance for reactor containment leakage rate testing in order to comply with the requirements of Appendix J to 10 CFR Part 50 and Appendix A to 10 CFR Part 50, GDC 52, GDC 53, and GDC 54 for containment leakage rate testing, inspection program, and ability to determine valve leakage rates for piping systems penetrating primary containment.

Section 5.3.14 states that the BWRX-300 design will conform to the guidance of SRP Section 6.2.6 in the same manner as described in the ESBWR DCD, and that GEH recommends that the existing SRP be used during future review of a BWRX-300 10 CFR Part 52 DCA if pursued.

Consistent with NRC practice, SRP Section 6.2.6 will be used during a future review of a BWRX-300 license application.

5.3.15 Standard Review Plan 6.2.7

The NRC staff focused its review pertaining to fracture prevention of the containment pressure boundary in Section 5.1.18, "10 CFR 50 Appendix A, GDC 51," of the LTR. This section includes a brief discussion on SRP Section 6.2.7, the components of the BWRX-300 to which SRP Section 6.2.7 may apply, and a recommendation that the existing SRP Section 6.2.7 be used during a "future review of a BWRX-300 [DCA or license application]."

Because NEDC-33911P does not describe the final BWRX-300 features (e.g., final containment size, volume, use of ferritic steel materials, etc.), exposure of metallic components to neutron fluence may warrant consideration of neutron embrittlement during future licensing activities. Consistent with SRP Section 6.2.7, the NRC staff would consider the effects of irradiation both near the core height and due to potential cavity streaming effects in reaching a safety conclusion. The NRC staff considers important topics for any future application to be the BWRX-300 containment, containment penetrations, and components within the containment exposed to neutron fluence. Therefore, the NRC staff would need to consider the potential effects of embrittlement, as well as SRP Section 6.2.7, as written, to reach a safety conclusion during future licensing activities.

The NRC staff notes that fracture toughness testing of ASME BPV Code, Section III, Class 2 materials for the BWRX-300 reactor containment and supporting components is mandatory and shall be performed as was first identified in the Summer 1977 Addenda Code Class 2 rules. The discussion in SRP Section 6.2.7 on materials that were not tested for fracture toughness refers to historical materials.

SRP Section 6.2.7 will be used during a future review of a BWRX-300 license application. The NRC staff notes that the applicant will be expected to address the embrittlement concerns as noted above to support the NRC staff's findings concerning fracture prevention of the containment pressure boundary.

5.4 Generic Issues

5.4.1 NUREG-0737

NEDC-33911P, Section 5.4.1, "NUREG-0737, Clarification of TMI Action Plan Requirements," November 1980, discusses requirements approved for implementation by the NRC as a result of the accident at TMI Unit 2. The NRC later codified some of these requirements in 10 CFR 50.34(f). Consistency with the requirements that are related to containment performance is discussed in Section 5.1.1.

The NRC staff's evaluation is described in SE Section 5.1.1, "10 CFR 50.34(f)."

5.5 Operational Experience and Generic Communications

5.5.1 Generic Letter 83-02

NEDC-33911P, Section 5.5.1, "Generic Letter 83-02," states that Generic Letter (GL) 83-02, NUREG-0737 Technical Specifications, dated January 10, 1983, contains a request for information for the current BWR licensees regarding NUREG-0737 items for which technical specifications are required, including guidance on the scope of a specification which the NRC staff would find acceptable and sample technical specifications. Technical specifications for the items related to containment and CIVs are to be proposed during future licensing activities.

The NRC staff finds the GEH plan to propose technical specifications related to containment and CIVs during future licensing activities to be acceptable.

5.5.2 Generic Letter 95-07

NEDC-33911P, Section 5.5.2, "Generic Letter 95-07," states that GL 95-07, "Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves," dated August 17, 1995, contains a request to ensure that safety related power operated gate valves that are susceptible to pressure locking or thermal binding are evaluated to ensure they are capable of performing the safety functions described in the licensing basis. This guidance will be evaluated for applicability during future licensing activities.

The NRC staff finds the GEH plan to evaluate the recommendations in GL 95-07 for applicability to the BWRX-300 design during future licensing activities to be acceptable.

6.0 LIMITATIONS AND CONDITIONS

This SE includes two limitations and conditions:

(1) Before use of the TRACG/GOTHIC containment safety analysis methodology as described in NEDC-33911P for the BWRX-300 design, the NRC staff must review and approve a detailed description of the methodology and modeling assumptions, including conservatism and uncertainty evaluation of its licensing model and its validation with the test data.

(2) As discussed in SE Section 5.1.22, the NRC staff must review and approve in future licensing activities the potential postulated breaks and cracks in the closed-loop piping being used for meeting GDC 55 on the “other defined basis.” In addition, the NRC staff will review the consequences of pipe failures outside containment resulting from ICS steam supply and condensate return piping and FMCRD hydraulic lines. The evaluation will include dynamic and environmental effects of pipe failures and fission product releases in the reactor building.

7.0 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 containment performance for the BWRX-300 as described in NEDC-33911P are acceptable. In particular, NEDC-33911P describes: (1) the BWRX-300 containment, containment functional design, containment heat removal systems, containment isolation system, combustible gas control in containment, containment leakage testing, and fracture prevention of the containment pressure boundary providing acceptance criteria, regulatory bases, and references to existing proven design concepts; (2) the BWRX-300 analytical methods to be used to demonstrate compliance with containment PCCS acceptance criteria; and (3) the BWRX-300 CIV design and regulatory bases to demonstrate compliance with regulations, including the “other defined basis” for compliance. If an applicant for a CP 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 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 containment performance for the BWRX-300 during future licensing activities in accordance with 10 CFR Part 50 or 10 CFR Part 52, as applicable. As discussed in this SE, GEH indicated that the detailed design of the BWRX-300 is not complete at this time. The NRC staff will make a final determination of the BWRX-300 acceptability when GEH completes the detailed design and the NRC staff reviews a BWRX-300 application during future licensing activities.