ENCLOSURE 2

M210056

NEDO-33911-A, Revision 2

Licensing Topical Report

BWRX-300 Containment Performance

Non-Proprietary Information

IMPORTANT NOTICE

This is a non-proprietary version of NEDC-33911P-A Revision 2, from which the proprietary information has been removed. Portions of the enclosure that have been removed are indicated by an open and closed bracket as shown here. [[ ]].

Note the NRC's Final Safety Evaluation is enclosed in NEDO-33911-A Revision 2. Portions of the Safety Evaluation that have been removed are indicated with double square brackets as shown here. [[ ]].

HITACHI

GE Hitachi Nuclear Energy

NEDO-33911-A
Revision 2
April 2021

Non-Proprietary Information

Licensing Topical Report

BWRX-300 Containment Performance

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

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

March 12, 2021

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

SUBJECT: FINAL SAFETY EVALUATION FOR GE-HITACHI LICENSING TOPICAL REPORT NEDC-33911P, REVISION 0, SUPPLEMENT 1, “BWRX-300 CONTAINMENT PERFORMANCE”

Dear Ms. Catts:

By letter dated March 31, 2020, GE-Hitachi Nuclear Energy Americas, LLC (GEH), submitted Licensing Topical Report (LTR) NEDC-33911P, Revision 0, “BWRX-300 Containment Performance” (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20091S340), to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval in support of a future licensing application for the GEH small modular reactor (SMR) under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” or Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” By letter dated September 4, 2020 (ADAMS Accession No. ML20248H570), GEH submitted Revision 0, Supplement 1 of the LTR.

The NRC staff has found LTR NEDC-33911P, Revision 0, as updated by the September 4, 2020, supplement, to be acceptable for referencing in licensing applications for the GEH SMR design to the extent specified in the enclosed safety evaluation (SE). The SE defines the basis for acceptance of the LTR. In addition, on March 1, 2021 (ADAMS Accession No. ML21049A340), the Advisory Committee on Reactor Safeguards concluded that the staff’s SE for LTR NEDC-33911P, Revision 0, Supplement 1, with the limitations and conditions imposed, is appropriate and should be issued.

In accordance with the guidance provided on the NRC’s LTR website (http://www.nrc.gov/about-nrc/regulatory/licensing/topical-reports.html), the NRC requests that GEH publish an accepted version of this LTR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE after the title page. Also, it must contain in its appendices historical review information, such as requests for additional information, accepted responses, and the actual revised pages (showing revision bars) that were included as part of NEDC-33911P Revision 0, Supplement 1. The accepted version shall include an "-A" (designated accepted) following the report identification symbol.

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

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

Sincerely,

/RA/

James J. Shea, Senior Project Manager
New Reactor Licensing Branch
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation

Docket No.: 99900003

Enclosure:
Safety Evaluation

cc: Listserv

SUBJECT: FINAL SAFETY EVALUATION FOR GE-HITACHI LICENSING TOPICAL REPORT NEDC-33911P, REVISION 0, "BWRX-300 CONTAINMENT PERFORMANCE" DATED: MARCH 12, 2021

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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

DOCKET NUMBER 99900034

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.

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LIST OF FIGURES

REVISION SUMMARY

Revision Number	Description of Change
0	Initial Issue
Supplement 1	Revised to incorporate the following changes and responses to NRC Requests for Additional Information (eRAIs): • [[ ]] from NEDC-33910P for Sections 2.1, 2.2, 2.2.3, 2.2.7.1, 3.1, 5.1.11, 5.3.8. • Sections 3.2, 3.3, 3.5, 3.6, 5.1.17, 5.3.3 and 6.5 are revised to reflect a title change for NEDC-33922P from GOTHIC Application for BWRX-300 Containment to BWRX-300 Containment Evaluation Method. • Changed Section Numbers as a result of response to NRC eRAI 9745: 5.2.3 to 5.2.4, 5.2.4 to 5.2.6 and 5.2.5 to 5.2.7. • Changed "Since" to "Because" in Sections 3.1 and 3.2. • Information that has been reclassified as non-proprietary is identified with for Section 2.1.2, last paragraph; 2.2, 2nd paragraph; 2.2.2, 4th bullet; 2.2.7.3; 3.1, 4th, 5th, and 6th paragraphs; 5.1.4; 5.1.5; 5.1.6; 5.1.7; 5.1.24; 5.3.3; 5.3.4, 2nd paragraph; 5.3.9, 2nd paragraph; and 5.4.1. • NRC Requests for Additional Information (eRAIs): – NRC eRAI 9745, Question 03.09.06-15, item (h) where Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves was added as Section 5.5.2 as this operating experience may be applicable to the detailed design of the CIVs to avoid thermal binding. – NRC eRAI 9745, Question 03.09.06-17, where Section 2.2.7 Containment Isolation Valves Design Requirements was revised to add two additional bullets that address CIV design to prevent valve movement from normal operating position, which is accomplished by positive mechanical means. – NRC eRAI 9745, Question 03.09.06-18, where new Sections 5.2.3, 5.2.5, and 5.2.8 for Regulatory Guides 1.84, 1.147 and 1.192, respectively, were added to reflect ASME Code Cases for design, inservice inspection and IST activities in satisfying 10 CFR50.55a. Renumbering of Section 5.2 occurred due to the insertion of these Regulatory Guides.

Revision Number	Description of Change
Supplement 1 (continued)	Revised to incorporate the following responses to NRC Requests for Additional Information (eRAIs):
	– NRC eRAI 9745, Question 03.09.06-19, where new Section 5.3.1 Standard Review Plan 3.9.6 was added to specify compliance with this guidance for functional design, qualification and IST programs for pumps, valves, and dynamic restraints for containment isolation valves. Renumbering of subsequent 5.3 sections follows. – NRC eRAI 9745, Question 03.09.06-20, where Sections 5.4 and 5.5 were revised to reflect that generic issues and operational experience would be provided in future licensing activities either by 10 CFR 50 or 10 CFR 52 licensing activities, and that the operational experience and generic communications provided in the LTR are based upon their relevance to the scope of the LTR only. – NRC eRAI 9745, Question 03.09.06-15, where Section 2.2.7 Containment Isolation Valve Design Requirements added a new bullet that addresses diversity for RPV isolation valve penetrations and where two containment isolation valves that have automatic isolation with RPV isolation valves ensure diversity; additionally, added new Section 5.5.2 Generic Letter 95-07 Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves that addresses applicability of pressure locking and thermal binding of valves during future licensing activities. – NRC eRAI 9745, Question 03.09.06-17, where Section 2.2.7 Containment Isolation Valve Design Requirements added three new bullets that addresses positive mechanical means in valve actuators to maintain these valves in their required post-accident valve positions. – NRC eRAI 9745, Question 03.09.06-18, where new Sections 5.2.3, 5.2.5 and 5.2.8 were added to reflect BWRX-300 compliance to the guidance of Regulatory Guides 1.84, 1.147, and 1.192, respectively for the acceptability of ASME Code Cases. Renumbering of previous and later sections in Section 5.2 occurred to reflect these new sections.

Revision Number	Description of Change
Supplement 1 (continued)	Revised to incorporate the following responses to NRC Requests for Additional Information (eRAIs):
	– NRC eRAI 9745, Question 03.09.06-19, where new Section 5.3.2 Standard Review Plan 3.9.6 was added to indicate BWRX-300 compliance to functional design, qualification and inservice testing for pumps, valves, and dynamic restraints guidance. Renumbered subsequent sections in Section 5.3 to reflect new Section 5.3.1 (see eRAI 9758) and 5.3.2. – NRC eRAI 9745, Question 03.09.06-20, where Section 5.4 Generic Issues and Section 5.5 Operational Experience and Generic Communications was modified to reflect that an up-to-date evaluation of relevant generic communications and experiences would be evaluated during future licensing activities in support of a 10 CFR 52 DCA or a 10 CFR 50 CP and OP application. – NRC eRAI 9746, Question 03.06.02-4, where bullet five was added to Subsection 2.2.2 Design Requirements that specified that BTP 3-4, Part B, Items 1(ii)(1)(d) and (e), and Items 1(ii)(2) through (7), are applied and Section 5.1.7 10 CFR 50 Appendix A, GDC 4 was modified to add compliance to BTP 3-4 Part B, Items 1(ii)(2) through (7) and the dynamic effects of those portions of the piping from the containment to the outboard CIVs. Section 5.1.7 was revised to reflect that the dynamic effects of postulated breaks and cracks in those portions of the piping beyond and including the outboard CIVs will be evaluated in future licensing activities. – NRC eRAI 9746, Question 03.06.02-5, where Section 3.1 and Section 5.1.7 was revised to show compliance to GDC 4 by evaluating the dynamic effects of jet loads, pipe whipping, postulated high-energy breaks, missiles and discharging fluids in the design of containment and the CIVs and described during future licensing activities to show compliance with GDC 4. Section 5.1.7 10 CFR 50, Appendix A, GDC 4 compliance was revised to indicate that the dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids will be included.

Revision Number	Description of Change
Supplement 1 (continued)	Revised to incorporate the following responses to NRC Requests for Additional Information (eRAIs):
	– NRC eRAI 9746, Question 03.06.02-6, where Section 2.2.7 Design Requirements was revised to add a new bullet that shows ASME Standard QME-1-20007 (or later edition) compliance for the design and procurement of the valves specified in future licensing activities. Section 5.1.7, 10 CFR 50 Appendix A, GDC 4 was also revised to comply with this same ASME Standard. – NRC eRAI 9760, Question 06.02.05-1, where Section 5.1.2 10 CFR 50.44 was revised to show that reliable equipment will be provided to monitor both oxygen and hydrogen concentrations in the BWRX-300 inerted containment during and following a BDBA. – NRC eRAI 9764, Question 06.02.01-1, where Section 5.1.17 10 CFR 50, Appendix A, GDC 50 compliance was revised to indicate that the containment structural design will be evaluated against the maximum expected external pressure with sufficient margin from a full spectrum of postulated accidents that would release reactor coolant to containment. – NRC eRAI 9766, Question 06.02.01-3, where Sections 3.4.2.2 GOTHIC Phenomenon Identification and Ranking Table (PIRT), 3.4.2.3 PIRT Survey and Tables 3-1 Phenomenon Ranking Criteria and 3-2 Phenomena Identification and Ranking Table for Containment (Excluding RPV) are moved to LTR NEDC-33922P BWRX-300 Containment Evaluation Method. The remaining sections were renumbered and references to the renumbered sections were amended accordingly. – NRC eRAI 9758, Question 06.02.04-1, where Section 2.2.7.1 was revised to state that the outside containment automatic CIVs closure time will be based upon the first fission product release greater than what is contained in the normal reactor coolant in source term evaluations and will be completed in future licensing activities. – NRC eRAI 9758, Question 06.02.04-2, where Section 2.2.7.1 was revised to reflect the design requirement of a closed loop outside containment plus two in-series automatic isolation valves inside containment meet the other defined basis provision of GDC 55.

Revision Number	Description of Change
Supplement 1 (continued)	Revised to incorporate the following responses to NRC Requests for Additional Information (eRAIs):
	– NRC eRAI 9758, Question 06.02.04-3, where Section 2.2.7.1 was revised to reflect that the HCUs meet GDC 55 as an "other defined alternative containment configuration design" by forming a closed system outside containment and is connected to an FMCRD inside containment with a normally open internal ball check valve to provide isolation from the RCPB; and new Section 5.3.1, Standard Review Plan 3.6.2 was added to reflect compliance to the provisions of SRP 6.2.4, Item 5, that requires compliance to SRP 3.6.2 for the BWRX-300 CRD system HCU piping and ball check valve alternative containment isolation valve arrangement.
1	Created "-A" version by adding the NRC's Final Safety Evaluation (Reference 6.14) and GEH's responses to the NRC's Requests for Additional Information (eRAIs) (References 6.10, 6.11, 6.12, 6.13). Renumbered to reference new Section 3.4.2.2 in 2nd paragraph of Section 3.6 due to response to eRAI 9766 where Sections 3.4.2.2 and 3.4.2.3 were removed from this LTR and relocated to NEDC-33922P. Added References 6.10 through 6.14.
2	Revised "-A" version by adding eRAIs 9760, 9764, 9765, 9766, 9767 and associated responses from GEH letter M200096 dated July 24, 2020 that were inadvertently omitted from Appendix A in Revision 1. Revision 1, Appendix B, did contain the associated LTR changes from these eRAIs.

Acronyms and Abbreviations

Term	Definition
ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
B&PV	Boiler & Pressure Vessel
BDBA	Beyond Design Basis Accident
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CCFL	Counter-current Flow Limitation
CIV	Containment Isolation Valve
COL	Combined Operating License
CP	Construction Permit
CRD	Control Rod Drive
CSAU	Code, Scaling, Applicability and Uncertainty
DBA	Design Basis Accident
DCA	Design Certification Application
ECCS	Emergency Core Cooling System
EFCV	Excess Flow Check Valve
EMDAP	Evaluation Model Development and Assessment Process
ESBWR	Economic Simplified Boiling Water Reactor
FMCRD	Fine Motion Control Rod Drive
GDC	General Design Criteria
GEH	GE Hitachi Nuclear Energy
GOTHIC	Generation of Thermal-Hydraulic Information for Containments
HGNE	Hitachi-GE Nuclear Energy Ltd.
IC	Isolation Condenser
ICS	Isolation Condenser System

Term	Definition
IE	Infrequent Event
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LTR	Licensing Topical Report
LWR	Light-Water-Reactor
NBS	Nuclear Boiler System
NC	Non-condensable
NDTT	Nil-Ductility Transition Temperature
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OL	Operating License
PCCS	Passive Containment Cooling System
PCV	Primary Containment Vessel
PIRT	Phenomenon Identification and Ranking Table
PSAR	Preliminary Safety Analysis Report
PWR	Pressurized Water Reactor
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SA	Severe Accident Management
SBO	Station Blackout
SMR	Small Modular Reactor
SRP	Standard Review Plan
SSC	Structure, System, and Component
TAF	Top of Active Fuel
TMI	Three Mile Island
TRACG	Transient Reactor Analysis Code General Electric

1.0 INTRODUCTION

1.1 Purpose

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

- Design requirements are specified for the containment and the Passive Containment Cooling System (PCCS). The design of the containment and PCCS meet the requirements of 10 CFR 50 Appendix A, General Design Criteria (GDC) 1, GDC 2, GDC 4, GDC 16, GDC 38, GDC 40, GDC 41, GDC 42, GDC 50, GDC 51, GDC 52, GDC 53, and GDC 54.
- Design requirements are specified for the containment isolation valves (CIVs). The design of the CIVs meet the requirements of 10 CFR 50 Appendix A, General Design Criteria, GDC 1, GDC 2, GDC 4, GDC 54, GDC 55, GDC 56, and GDC 57.
- Analytical methods are specified for evaluating containment performance, including acceptance criteria. The analytical methods are used to demonstrate compliance with the requirements of 10 CFR 50 Appendix A, GDC 16, GDC 38, GDC 40, and GDC 50.
- Acceptance criteria are defined for the BWRX-300 containment performance in accordance with the design requirements specified for the containment, PCCS, and CIVs.

1.2 Scope

The scope of this report includes the following:

- A technical description of the BWRX-300 containment, PCCS, and CIV design features and design functions, acceptance criteria, regulatory bases, and references to existing proven design concepts based upon previous Boiling Water Reactor (BWR) designs, including the Advanced Boiling Water Reactor (ABWR) and Economic Simplified Boiling Water Reactor (ESBWR).
- A technical description of the BWRX-300 analytical methods to be used to demonstrate compliance with containment, PCCS, and CIV acceptance criteria. Detailed descriptions, benchmarking, and demonstration analyses for the analytical methods, as well as the analyses for demonstrating compliance with the acceptance criteria, will be provided during future licensing activities.
- 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, to describe compliance with regulatory requirements and to describe alternative approaches to regulatory guidance that may be referenced in future licensing activities either by GEH in support of a 10 CFR 52 Design Certification Application (DCA) or by a license applicant requesting a Construction Permit (CP) and Operating License (OL) under 10 CFR 50 or a Combined Operating License (COL) under 10 CFR 52.

2.0 TECHNICAL EVALUATION OF CONTAINMENT PERFORMANCE

2.1 General Introduction

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

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

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

2.1.1 Reactor Pressure Vessel

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

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

Figure 2-1 shows a representation of BWRX-300 RPV and internals.

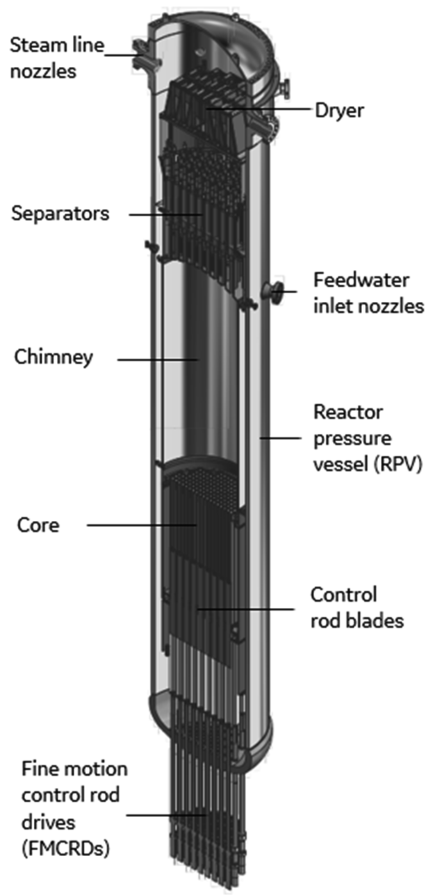

Figure 2-1: BWRX-300 Reactor Pressure Vessel and Internals

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

The major reactor internal components include:

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

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

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

2.1.2 Isolation Condenser System

The ICS passively removes heat from the reactor (i.e., heat transfer from the IC heat exchanger tubes to the surrounding IC pool water is accomplished by condensation and natural circulation, and no forced circulation equipment is required) when the normal heat removal system is unavailable following any of the following events:

- Sudden reactor isolation at power operating conditions
- During Station Blackout (SBO) (i.e., unavailability of all alternate current (AC) power)
- Anticipated Transient Without Scram (ATWS)
- LOCA

The ICS consists of three independent trains, each containing an IC heat exchanger that condenses steam on the tube side and transfers heat by heating/evaporating water in the IC pool, which is vented to the atmosphere. The arrangement of one IC heat exchanger situated in an IC pool is shown on Figure 2-2.

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

The ICS is initiated automatically on high RPV pressure indicating an overpressure event or on signals indicating a LOCA. To start an IC train, the IC condensate return valve is opened whereupon the standing condensate drains into the reactor. [[

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

2.2 Overview of Containment

The BWRX-300 containment is based upon GEH BWR experience and fleet performance:

- Containment size comparable to a small BWR drywell
- Containment peak accident pressure and temperatures within existing BWR experience base
- Containment load simplified when compared to conventional BWRs with pressure suppression containments
- Nitrogen-inerted containment same as BWR Mark I and Mark II containments
- Pressure and temperature during normal operation maintained by fan coolers, similar to existing BWRs
- Upon loss of active containment cooling, heat removal is achieved by PCCS

The BWRX-300 containment is an underground subterranean steel or reinforced concrete primary containment vessel (PCV) or a combination of steel and reinforced concrete. Figure 2-3 shows a typical steel containment. Other potential construction types are of similar size and have the same functional features. The containment does not have a suppression pool. Heat is removed by PCCS as described in Section 2.2.8. The reactor cavity pool for PCCS heat removal during design basis events is located above containment and is vented to the atmosphere.

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. Within these subcompartments there are no large bore high energy lines. Typical small piping [[ ]] located within these subcompartments include the FMCRD hydraulic lines and instrument lines. Large bore high energy lines are also located as far as practical from the outside of these subcompartment walls. Therefore, line breaks inside or outside these subcompartments do not create significant pressure differentials across the subcompartment walls.

Combustible gas control is not required 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 the containment atmosphere is initially nitrogen-inerted.

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Figure 2-3: BWRX-300 Typical Steel Containment

2.2.1 Containment Design Functions

The primary design functions of the BWRX-300 PCV 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.

The PCV design uses a nitrogen-inerted containment atmosphere during operating modes. 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 sever accident management condition. The dilution provides protection to the PCV and its internal components from hydrogen combustion or detonation. The inert atmosphere design has the additional benefit of minimizing long-term corrosion and degradation of the PCV and the contained components by limiting the exposure to oxygen during plant operating service life.

The PCV has provisions for personnel access (see Section 2.2.5) and for habitability during plant outages to perform maintenance, inspections and tests required for assuring PCV integrity and reliability, and the integrity and performance reliability of interfacing structures, systems, and components (SSCs) contained inside the PCV enclosure.

2.2.2 Containment Design Requirements

The PCV is classified as a Safety Class 1, safety-related, and seismic Category I structure.

Design Requirements:

- The PCV is designed either as a metal containment in accordance with the rules and requirements of American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NE, or as a concrete containment in accordance with the rules and requirements of ASME B&PV Code, Section III, Division 2, which is a dual standard with ACI-359.
- Piping systems that pass through PCV mechanical penetrations and CIVs, with the exception of the [[ ]] are designed in accordance with the rules and requirements of ASME B&PV Code, Section III, Division 1, Subsection NC, Class 2 Components.
- [[ ]] that function as the inboard CIVs are designed in accordance with the rules and requirements of ASME B&PV Code, Section III, Division 1, Subsection NB, Class 1 Components.
- For piping connected to the RPV isolation valve assemblies, extending to the containment wall, the BWRX-300 design requirements include identification of postulated pipe rupture locations and configurations inside containment as specified in NUREG-0800, Standard Review Plan (SRP), Branch Technical Position (BTP) 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Part B, Item 1(iii)(2), and identification of leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2).
- ASME B&PV Code, Section III, Division 1, Subarticle NE-1120, and the design criteria from BTP 3-4, Part B, Items 1(ii)(1)(d) and (e), and Items 1(ii)(2) through (7), are applied to eliminate postulating breaks and cracks in those portions of piping from the containment wall up to the outboard CIVs.
- Structural supports for piping systems and components inside the PCV are designed in accordance with the rules and requirements of ASME B&PV Code, Section III, Division 1, Subsection NF, Supports.
- Materials used for the PCV, penetration piping systems and the associated supports are designed in accordance with the rules and requirements of ASME B&PV Code, Section II, Material Specifications. Exception to the materials requirement is allowed for the nonconductive portions of electrical penetrations.
- Additional structures that are part of the PCV internals are designed in accordance ANSI/AISC N690, Specification for Safety-Related Steel Structures for Nuclear Facilities, with Supplements.

2.2.3 Containment Performance Requirements

The BWRX-300 PCV is sized and equipped to contain the mass and energy released by a large break LOCA [[ ]], and for small breaks [[ ]].

In addition, the PCV volume is sufficient to accept the additional non-condensable (NC) gas from the ICS vents, as a backup discharge volume, when the ICS is in operation during any plant operating mode or condition.

The PCV design is for a service life of 60 years.

2.2.4 Containment Boundary

The PCV physical design boundary is used to interpret design code applicability to the PCV and its component parts, including the following:

- The shell bottom head supported from the basemat and any external bottom head supports to the interfacing connection with the civil structure;
- Outside diameter of the PCV wall from the bottom head to the transition ring;
- The transition ring including the neck to the shell flange, and the flanged closure head and flange bolting;
- Any external support structures attached to or forming part of the PCV wall exterior, particularly for the transfer of load to support the RPV, to the interfacing connection with the civil structure;
- The outer surface extent of PCV hatches and airlocks;
- The PCV penetration sleeves up to the interface connecting weld joint between the sleeve closure plate or bellows and the process piping (duct), tubing penetration assembly or electrical penetration assembly;
- The outboard CIVs, including pipe support(s) and the portion of pipe beyond CIVs where the first pipe supports are affixed;
- The outer closure of electrical penetration assemblies; and,
- [[ ]] (see Section 2.2.8).

A description of containment heat removal design functions and design features can be found in Section 2.2.8, and key phenomena important to the analysis of the BWRX-300 containment response in design basis events are described in Section 3.4.

2.2.5 Access and Maintenance

The PCV has a flexible metallic seal, i.e., refueling bellows between the RPV exterior surface and PCV wall interior. The refueling bellows assembly is designed to accommodate the movement of the vessel caused by operating temperature variations and seismic activity. The refueling bellows is permanently installed by welded joints to specified attachment interface locations below the RPV and PCV head closure flanges. The refueling bellows provides a 360° structural barrier that retains the refueling cavity water above the PCV when the PCV head is removed. The design of the refueling bellows includes protection from puncture or damage from dropped items during refueling outage activities or workers performing RPV or PCV head removal or installation activities. The design also has a drain to remove water from the bellows low point and is required to be cleanable (i.e., for removal of non-soluble radioactive contamination, including fuel particles, that settle onto the bellows assembly during refueling outages).

The PCV design provides access to internal or external surfaces as required to implement a program of periodic inspection of PCV integrity. Inspection requirements are in accordance with ASME BPVC, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Subsections IWA, and IWE, Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants.

The PCV has ingress/egress through at least two personnel hatches located at different elevations; one located to facilitate under-vessel maintenance and one located to facilitate RPV integral nozzle isolation valve maintenance.

Adequate space is provided around equipment located inside the PCV for the removal, servicing, and maintenance of equipment.

Where practical, platforms and staircases are provided for access to equipment for inspection, examination, surveillances, and maintenance. Such platforms and structures should not hinder the performance functions of the PCV, and their design includes evaluation of the effects of high energy jet and impingement loads to minimize missile and debris generation. Provision for removable stairs and platforms should be used in place of permanent installations when needed to assure performance of PCV functions during operating modes other than plant outages.

The PCV has installed crane rails and cart tracks, as appropriate, and pick points to assist lifting, positioning and transport of components, equipment, maintenance tools, materials, and inspection and test machines, equipment, and tools, to service systems and components inside the PCV including the interior side of the PCV boundary.

2.2.6 Containment Penetrations

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.
- PCV electrical penetrations are sealed to the interior side of the PCV pressure boundary.
- Hydraulic lines for the 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.

- An isolation function may be shared by a group of penetrations or uniquely applied to a single pipe, tubing run or ductwork penetration based on the associated system function and the assigned instrumentation and control leakage detection and control logic.
- Penetrations for liquid process lines or process lines that can become liquid-filled following isolation are protected from excess thermal pressurization due to containment heating of the liquid volume within the penetration piping.

Sufficient space and the additional process system component facilities are provided between penetration isolation valves and the PCV boundary wall to permit:

- Inservice inspection of non-isolable welds;
- Access and facilities to perform local leak rate testing of isolation valves;
- Access to operate local manual controls;
- Access to perform isolation valve assembly maintenance; and,
- Cutout and replacement of isolation valves using standard pipe cutting equipment, pipe fitting tools and equipment, and piping component welding equipment.

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.

2.2.7 Containment Isolation Valves

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.

- Figure 2-4 shows an example of RPV isolation valves. Figure 2-5 and Figure 2-6 show the systems that are connected to the RPV boundary with [[ ]].
- Figure 2-7 shows the ICS connections to the RPV boundary and other ICS CIVs and process valves.
- Figure 2-8 shows the lines to the FMCRDs.
- Figure 2-9 and Figure 2-10 show CIVs that are connected to containment atmosphere and closed systems in order to meet GDC 56 and GDC 57, respectively.

Leak-tightness of CIVs is verified by 10 CFR 50, Appendix J, Type C tests. Leak-tightness of containment is verified by 10 CFR 50, Appendix J, Type A testing. Leak-tightness of other containment penetrations is verified by 10 CFR 50, Appendix J, Type B testing.

Design Requirements:

- Capability for isolation of pipes or ducts that penetrate the containment is performed by means or devices that provide a containment barrier to limit leakage within permissible limits.
- CIV closure timing requirements are commensurate with the timing of the potential for fission product releases.

- Isolation valves for instrument lines that penetrate containment conform to the requirements of RG 1.11, Instrument Lines Penetrating the Primary Reactor Containment.
- Isolation valves, actuators and controls are protected against the loss of their safety-related function from missiles and postulated effects of high and moderate energy line ruptures.
- Design of the CIVs, and associated piping and penetrations will meet the requirements of seismic Category I components, and designed in accordance with the rules and requirements of ASME B&PV Code, Section III, Division 1, Subsection NE, Class MC Components, and Subsection NC, Class 2 Components, in accordance with their quality group classification.
- The design of the control functions for automatic CIVs ensure that resetting the isolation signal shall not result in the automatic reopening of CIVs.
- Penetrations with trapped liquid volume between the isolation valves have adequate relief for thermally induced pressurization.
- Diversity for penetrations where RPV isolation valves are credited as one of the containment isolation valves is accomplished by actuation from separate and diverse control systems that are single failure proof. In other penetrations where two containment isolation valves are used that have automatic isolation, diverse actuation signals are applied to ensure the function is achieved.
- The CIVs for main steam line, feedwater, shutdown cooling, and reactor water cleanup shall fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means.
- [[
  ]] with valve actuators designed to maintain the valves in their as-is position by positive mechanical means.
- All other CIV penetration configurations will be designed with valve actuators with positive mechanical means to ensure that upon automatic actuation or a loss of signal or control power to both valves, the valves will be maintained in the required post-accident valve position.
- Valve qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as endorsed by NRC Regulatory Guide (RG) 1.100, will be addressed in the detailed design and the procurement process of the valves, and will be specified during future licensing activities.

The FMCRDS shown on Figure 2-8 are similar mechanically as the ESBWR with the exception that the [[
]] FMCRD system design is described in the ESBWR Design Control Document Tier 2, Chapter 4 Reactor, 26A6642AP Rev. 10, Section 4.6 [Reference 6.1].

2.2.7.1 Containment Isolation Valves Connected to RPV Boundary

The BWRX-300 RPV design, acceptance criteria, and performance is delineated in Licensing Topical Report (LTR) NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 6.2]. [[

]] The outside containment automatic CIVs closure time will be established to assure containment isolation prior to the first fission product release greater than what is contained in normal reactor coolant in source term evaluations which will be completed in future licensing activities. These closure times are expected to be in the order of minutes. Additionally, the valve closing time for all CIVs will support specific break isolation functions balanced with water hammer and valve loading considerations. LTR NEDC 33921P, Severe Accident Management, will provide the evaluation for fission product releases resulting from BDBA or SA events and provide the necessary timing information to establish the closure times for the outside containment isolation valves.

Small pipes for level instruments use Excess Flow Check Valves (EFCVs) to conform to the requirements of RG 1.11, Instrument Lines Penetrating the Primary Reactor Containment.

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Figure 2-4: RPV Isolation Valve Assembly (Example)

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Figure 2-5: Main Steam and Feedwater CIVs Connected to RPV Boundary

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Figure 2-6: CIVs Connected to RPV Boundary

For the ICS as shown on Figure 2-7, [[ ]] comply with the "other defined basis" alternative containment configuration requirements of GDC 55.

[[ ]]. An alternative arrangement is provided for the ICS where two in-series RPV isolation valves on each end of the system function as containment isolation valves. A break in an ICS line either inside or outside containment could be isolated by either of the two redundant [[ ]]. The piping in the area between the outermost [[ ]] and the containment boundaries, as well as the piping through the seismic Category I reactor building where the ICS steam supply and the ICS condensate return piping connect to the ICS heat exchanger located in the ICS pool, are designed using ASME Section III, Class 1, NB piping, which limits the probability of breaks in these segments of the piping. Additionally, a break, between the RPV isolation valves and the containment would be isolated by the RPV isolation valves to stop the leak and would be contained by the closed system outside containment that is designed to withstand full reactor pressure. It would require an additional break before a radioactive release could occur, and even with an additional break, the coolant source remains isolated. Therefore, this design can accommodate a single failure and maintain containment leak integrity.

The ICS steam supply lines for each train contain two in-series valves inside containment, combined with a closed loop outside containment, thus providing containment isolation. The ICS steam supply lines and condensate return lines pass through the seismic Category 1 reactor building in order to connect to the isolation condenser in the ICS pool. The ICS vent lines each contain two in-series containment isolation valves and are attached to the closed loop outside containment. The ICS condensate return line for each train has two valves in-series to provide RPV isolation and containment isolation functions and are located inside containment where they are protected from outside environmental conditions that may result from a failure outside containment. The ICS condensate return line along with the steam supply line is automatically isolated when leakage is detected in the specific ICS train.

Given the above rationale, the containment isolation provisions for the ICS condensate, steam, vent and purge lines constitute an appropriate application of the "other defined basis" alternative defined in 10 CFR 50, Appendix A, GDC 55. A single failure would not disable the containment isolation function, while allowing the [[ ]] to remain open to allow the ICS to function as a part of the ECCS.

[[

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Figure 2-7: Isolation Condenser CIVs Connected to the RPV Boundary

For the FMCRD hydraulic lines for the scram function shown on Figure 2-8, the containment penetrations do not have automatic isolation valves based on being closed piping system outside containment and having RCPB isolation (internal ball check valves) in the design of the FMCRDs. The hydraulic control units meet GDC 55 as an "other defined basis" alternative containment configuration by forming a closed system outside containment. The control rod drive mechanical design incorporates a brake system and ball check valve that reduces the chances of rapid rod ejection. The ball check valve functions as a safety-related component because it prevents reverse flow from the scram inlet port against the pressure and flow conditions caused by a break of the scram line, preventing the loss of pressure from the underside of the hollow piston and generation of loads on the drive that could cause a rod ejection. This normally open ball check valve isolates the RCPB as needed from the small diameter, high pressure hydraulic insertion line that penetrates containment in order to attach to the HCU assembly. The HCU assembly serves as a closed system outside containment. At the HCU assembly, the hydraulic insertion line has a normally closed scram valve which allows high pressure water to flow from the accumulator as needed for a scram and there is also a normally open check valve isolating the purge water supply. Additionally, there are manual isolation valves that can be used to further isolate the HCU from the hydraulic insertion line as needed. Adding additional isolation valves in this piping for the purpose of containment isolation is not in the direction of highest safety because it could become a new potential failure mode in a safety critical system and will not improve the containment integrity because the small diameter high pressure hydraulic lines are attached to a closed system outside containment and therefore do not cause a risk of containment leakage.

The FMCRD design provides protection against loss of leak tight integrity. High pressure purge water continually flows through the drive with the water entering the ball check valve in the middle of the housing and flows around the hollow piston into the reactor. O-rings seal the lower housing. A labyrinth seal near the top of the drive restricts the flow into the reactor. During a scram, the labyrinth seals the high-pressure scram water from the reactor vessel. Reverse flow in the unlikely event of a hydraulic supply line break causes the ball check valve to move to the closed position. This prevents loss of pressure to the underside of the hollow piston, that in turn, prevents generation of loads on the drive that could cause rod ejection and serves as an isolation of the break from the RCPB. The scram insert piping from the HCU room to the FMCRDs are designed in accordance with Articles NB-2160 and NB-3120 of the ASME Code. The only primary pressure boundary components are the lower housing of the spool piece assembly and the flange of the outer tube assembly. These components are made with 300 series stainless steel materials in accordance with the ASME Code, Section III. Some CRDs are removed each refueling outage and disassembled for routine inspection of drive parts, including the CRD bolting and hard-surfaced parts accessible for visual examination in accordance with manufacturer's CRD maintenance procedures. The inspection program is adequate to detect any defects or leaks before they become serious operating problems. Further, the design provides for detection capability such that a potential leak could be discovered by the containment leak detection system and isolated to ensure containment pressure remains within design limits.

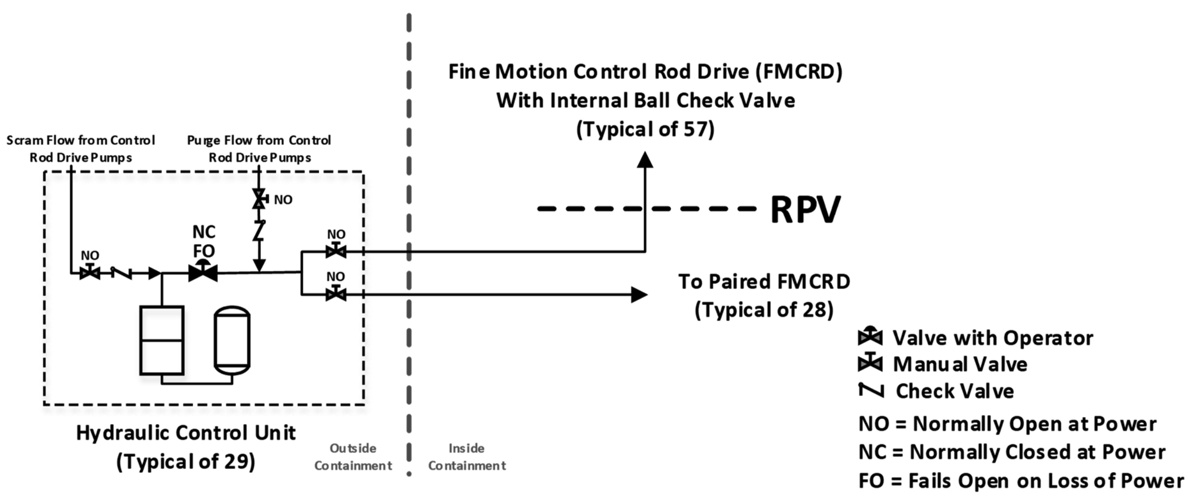

Figure 2-8: FMCRD CIVs Connected to RPV Boundary

2.2.7.2 Containment Isolation Valves Connected to Containment Atmosphere

The BWRX-300 CIVs attached directly to the containment atmosphere and shown on Figure 2-9 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 are provided with two normally closed outside containment manual CIVs. 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 is provided with normally closed inside and outside containment automatic CIVs.

The process gas and radiation monitoring system is a closed system outside containment and is provided with normally open outside containment automatic CIVs because it is an essential system following beyond design basis events and severe accident management.

The floor drain sump line is provided with two normally closed outside containment automatic CIVs, because it is not practicable to include an inside containment automatic CIV 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.

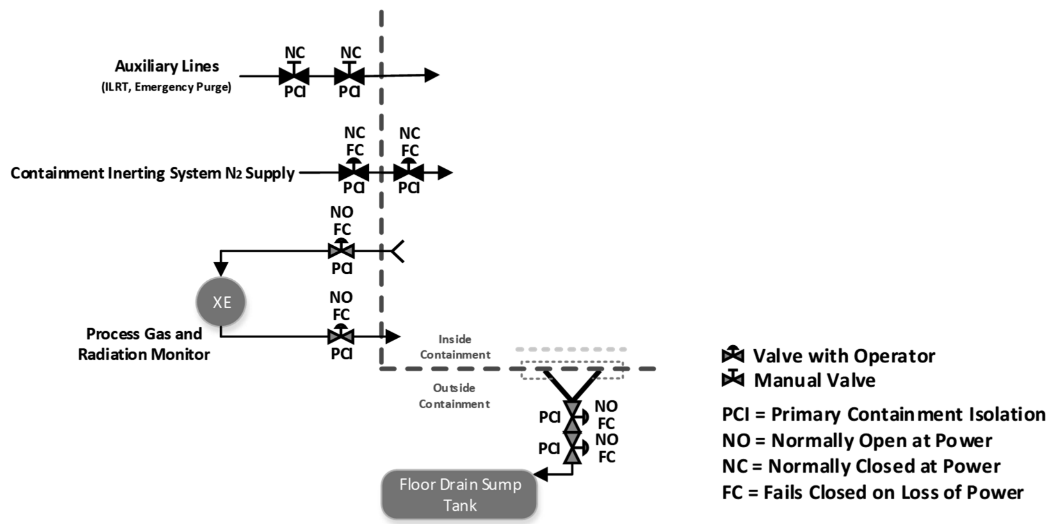

Figure 2-9: CIVs Connected to Containment Atmosphere

2.2.7.3 Containment Isolation Valves Connected to Closed Systems

The BWRX-300 closed system CIVs shown on Figure 2-10 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 pneumatic nitrogen or air system and the quench tank supply system are provided with either normally open or normally closed inside and outside containment automatic CIVs.

The service and breathing air system and demineralized water system are provided with normally closed inside and outside containment manual CIVs.

The chilled water supply and return are provided with normally open outside containment automatic CIVs.

[[

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Figure: 2-10 CIVs Connected to Closed Systems

2.2.8 Passive Containment Cooling System (PCCS)

The PCCS is based upon proven concepts and [[

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2.2.8.1 PCCS Design Functions

The PCCS transfers heat [[

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2.2.8.2 PCCS Design Requirements

The PCCS is designed in accordance with the design requirements for the containment in Section 2.2.2 above.

2.2.8.3 PCCS and Containment Boundary

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Figure 2-11: BWRX-300 PCCS (Example Configuration)

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Figure 2-12: BWRX-300 PCCS (Example Configuration)

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

3.1 Scope of the Evaluation Model

The design basis events for the containment are:

- AOO
- SBO
- ATWS
- Large break LOCA [[ ]] inside the containment
- Small breaks [[ ]] inside the containment

Because there is no discharge of steam or liquid into the containment in AOO, SBO and ATWS events, the only heat load to the containment is the heat transferred through the pipe and RPV insulation. Because 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.

Large break LOCA events inside the containment are the double-ended guillotine break of one of the following pipes:

- Main steam pipe
- Isolation condenser steam pipe
- Feedwater pipe
- Isolation condenser condensate return pipe

The pipes that are subject to a large break LOCA have two RPV isolation valves. At least one of the two valves on the broken line is closed subject to single failure criterion.

Small breaks inside the containment are assumed to remain unisolated. These small pipes include instrument lines.

The objective of the evaluation model is to demonstrate that the design pressure and temperature bound the accident peak pressure and temperature, and that the heat removal systems reduce the containment pressure rapidly. The acceptable results will demonstrate compliance with GDC 38 and GDC 50. The target for rapid depressurization is to reduce the pressure to the 50% of the peak accident pressure of the most limiting LOCA in 24 hours. The results are also used for equipment environmental qualification. Peak air/steam temperature resulting from a LOCA is not a meaningful parameter that can be compared to design limits for the structures. The figure of merit for temperature is the structure temperature, which can be compared to the design limits.

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.

The dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids will be evaluated in design of the containment and CIVs, and described during future licensing activities to comply with the design requirements of 10 CFR 50, Appendix A, GDC 4. Jet loads resulting from pipe breaks are not in the scope of the evaluation method described in this section. The jet loads and zone of influence are evaluated using a separate structural method that will be described during future licensing activities. However, the postulated break locations, type of break, and mitigating features for RPV and containment performance are within the scope of this document and LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 6.2].

3.2 Overview of the Evaluation Model

The evaluation model for the BWRX-300 containment response utilizes the applicable parts of the ESBWR evaluation methods which have been reviewed and approved for the ESBWR Design Certification [Reference 6.4].

BWRX-300 RPV is like the ESBWR RPV; however, the BWRX-300 containment is different than the ESBWR containment.

The most challenging features of the ESBWR containment for modeling are the wetwell, suppression pool, PCCS (which is much different and more complicated than the BWRX-300 PCCS), and the annulus between the RPV and the biological shield which is subject to pressurization and acoustic loads. The BWRX-300 containment does not have any of the above features. However, conservative temperature and steam / NC gas composition distributions can be calculated for the BWRX-300 containment using an appropriate model with nodalization.

The BWRX-300 containment evaluation model uses the Transient Reactor Analysis Code General Electric (TRACG) ESBWR RPV model described in Section 3.3. The containment is modeled separately using Generation of Thermal-Hydraulic Information for Containments (GOTHIC) Version 7.2a or the latest version. [[

]]

The computer codes used in the containment evaluation, TRACG and GOTHIC, are mature codes, each having an extensive qualification base, and each having been reviewed in detail. The application method developed for the purposes of the BWRX-300 containment evaluation follows the applicable sections of the Regulatory Guide 1.203 for a conservative analysis utilizing mature computer codes. Conservatism in the evaluation model is achieved by biasing the inputs and modeling parameters to bound the uncertainties, rather than performing a statistical analysis. The conservatism of the evaluation model is demonstrated by benchmarking to the available test data, which is to be established as part of the application methodology in LTR NEDC-33922P, BWRX-300 Containment Evaluation Method [Reference 6.5].

3.3 TRACG Mass and Energy Releases for Containment

Mass and energy release are calculated by TRACG and is the primary GEH tool for RPV neutronics and thermal-hydraulics calculations previously submitted in these GEH LTRs:

- NEDE-32176P, Revision 4, TRACG Model Description
- NEDE-32177P, Revision 2, TRACG Qualification
- NEDC-32725P, Revision 1, TRACG Qualification for SBWR
- NEDC-33080P, Revision 1, TRACG Qualification for ESBWR

Previous TRACG Containment/LOCA submittals for the models and qualification of TRACG are applicable to BWRX-300. The method accounts for the uncertainties and compensates for them by biases in the modeling parameters and in the plant parameters. BWRX-300 containment analysis method utilizes only those sections of the ESBWR Containment/LOCA analysis method related to the RPV and break flow, and correlations and biases. [[
]] The BWRX-300 TRACG model outputs are to be provided in LTR NEDC-33922P, BWRX-300 Containment Evaluation Method [Reference 6.5], or a separate TRACG LTR.

3.4 GOTHIC Containment Model

3.4.1 Overview of the GOTHIC Computer Code

GOTHIC is a general-purpose thermal-hydraulics software package specifically developed for nuclear power plant containments and similar confinements by the nuclear industry. GOTHIC solves the conservation equations for mass, momentum, and energy equations in multi-dimensional and/or lumped-parameter volumes. The conservation equations are solved for steam/gas mixture, continuous liquid, and liquid droplets. In addition, GOTHIC allows for the secondary fields for mist and liquid components. The NC gases may be composed of several species.

GOTHIC has been used in the industry extensively for containment pressure and temperature analyses, and equipment environment qualification outside the containment. GEH currently uses GOTHIC Version 7.2a but intends to use newer versions in the future. GOTHIC 7.2a includes several condensation models in the presence of NC gases that were lacking in earlier versions. Therefore, no code changes or additions are required to model the phenomena applicable to the BWRX-300 containment.

3.4.2 Evaluation Model Development for GOTHIC

The methodology utilizes the Code, Scaling, Applicability and Uncertainty (CSAU) in NUREG/CR-5249 and Regulatory Guide 1.203. Pressure and temperature in the air and structures are the primary parameters of merit.

3.4.2.1 Requirements of the Model

Element 1 of the RG 1.203 is to establish the requirements of the model:

- Step 1 of Element 1 is to specify the analysis purpose, transient class, and the power plant class, which are described in Section 3.1.
- Step 2 of Element 1 is to specify the Figures of Merit. The purpose of the evaluation method is also discussed in Section 3.1.
- Step 3 of Element 1 is to identify systems, components, phases, geometries, fields and processes that must be modeled.

Systems, subsystems, modules, and components that are relevant to the containment response include the following (those that are modeled by TRACG indicated within parentheses, and are not included in GOTHIC containment model development):

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 CRD systems which may add water from outside containment (TRACG)

The constituents/chemical forms of the fluids are water, nitrogen, hydrogen, and oxygen. The constituents/chemical forms of the structures/heat slabs are steel, concrete, and within the RPV TRACG model, uranium dioxide fuel and zircalloy cladding. The phases involved are solid, liquid, and vapor. 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. For GOTHIC, the geometry is like a small dry containment. The PCCS geometry is shown in Section 2.2.8. Fields include the properties that are being transported; specifically, mass, momentum, and energy. Transport Processes are mechanisms that determine the transport of, and interactions between, constituent phases throughout the system. The phenomena identified include the transport processes.

3.4.2.2 Development of the Assessment Base

Development of the assessment base follows the applicable sections of the guidance in RG 1.203. It should be noted that most of Elements 2 and 3 in RG 1.203 have been completed as part of the GOTHIC code development and documented in the GOTHIC technical and qualification reports. The remaining items of RG 1.203 Elements 2 and 3 include:

- Determining uncertainty in the correlations relating to the phenomena ranked high and medium based upon the existing experimental base for these correlations;
- Establishing suitably conservative biases in the above correlations;
- Establishing suitably conservative input parameters; and
- Benchmarking the method against the integral tests representative of the BWRX-300 containment to demonstrate the conservatism in the method.

3.5 TRACG and GOTHIC Analyses Numerical Convergence

Numerical convergence of TRACG and GOTHIC individually, and the convergence of the iteration is part of the development of the application method. Both TRACG and GOTHIC have internal convergence criteria and report the total numerical error in the output. Both codes limit the time step size automatically to maintain the error below the acceptance criteria.

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.

A BWRX-300 containment nodalization study is to be included to demonstrate that finer nodalization than used in the application method does not have a significant effect on the results.

Finally, TRACG and GOTHIC analyses iterations continue until there is no significant change in the containment pressure and temperature. The criteria for the acceptance of the sufficiency of convergence is to be established as part of the application methodology in LTR NEDC-33922P, BWRX-300 Containment Evaluation Method [Reference 6.5].

3.6 Summary of the Containment Evaluation Method

The BWRX-300 containment evaluation method for the design basis events uses the TRACG ESBWR model for the mass and energy release from the RPV, and the heat transfer from the RPV and attached piping through the insulation are used as boundary conditions for the GOTHIC containment response model. The TRACG model for the RPV has been previously reviewed in detail for the ESBWR design, which is very similar to the BWRX-300 RPV. GOTHIC code is specialized for containment analyses, particularly for dry containments. All phenomena ranked high or medium are modeled in GOTHIC. Both TRACG and GOTHIC are well qualified codes in their respective fields and have been used extensively over a few decades.

In order to establish a conservative evaluation method, the applicable steps in RG 1.203 are being followed. The steps up to and including the PIRT have been completed and presented in the sections above. Section 3.4.2.2 establishes the remaining RG 1.203 elements to be completed, while Section 3.5 discusses the numerical convergence of the TRACG and GOTHIC models. This establishes that all phenomena related to the containment evaluations for the design basis events are covered in TRACG and GOTHIC methodology codes. The other elements of the method, including the demonstration analyses, and the specifics of the application method are delineated in LTR NEDC-33922P, BWRX-300 Containment Evaluation Method [Reference 6.5].

4.0 CONTAINMENT PERFORMANCE ACCEPTANCE CRITERIA

The BWRX-300 containment performance acceptance criteria include the following:

- 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 50, Appendix A, GDC 2, GDC 4, GDC 16, GDC 38, GDC 41, GDC 50, and GDC 51.
- In accordance with 10 CFR 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% margin during the Preliminary Safety Analysis Report (PSAR) phase in order to conform to SRP 6.2.1.1.A Acceptance Criteria.
- In accordance with 10 CFR 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 maintains 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 50 Regulations

5.1.1 10 CFR 50.34(f)

10 CFR 50.34(f), Additional Three Mile Island (TMI) related requirements, requires that each applicant for a design certification, design approval, combined license, or manufacturing license under Part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section, except for paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Although it is not yet determined whether a 10 CFR 52 license application may be submitted for a BWRX-300, these requirements are evaluated herein. 10 CFR 50.34(f)(2) states that to satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the OL stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. The following requirements are evaluated as they are related to 1) [[ ]]; 2) containment purging and venting using As Low As Reasonably Achievable (ALARA) principles; 3) monitoring containment pressure, water level, and hydrogen levels during normal operations and accidents; and 4) containment structural integrity:

- Regulatory Requirement: 10 CFR 50.34(f)(2)(xiv) requires providing containment isolation systems that: (II.E.4.2) (A) Ensure all non-essential systems are isolated automatically by the containment isolation system; (B) Provide two isolation barriers in series for each non-essential penetration (except instrument lines); (C) Do not result in reopening of the CIVs on resetting of the isolation signal; (D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation; and (E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

  Statement of Compliance: All non-essential systems automatically isolate with two isolation barriers in series except for non-essential instrument lines. None of the non-essential systems reopen on containment isolation reset signals and have a set point pressure for initiating containment isolation as low as compatible with normal operation. Automatic closing on a high radiation signal is provided where required to meet the requirements of 10 CFR 100. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.34(f)(2)(xiv).

- Regulatory Requirement: 10 CFR 50.34(f)(2)(xv) requires that the design provide the capability to containment purge/vent to minimize the purging time consistent with ALARA principles for occupational exposure; and provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (Item II.E.4.4)

  Statement of Compliance: The BWRX-300 containment emergency purge system is designed to reliably isolate under accident conditions and is capable of purging and venting for consideration of ALARA occupational exposure. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.34(f)(2)(xv).

- Regulatory Requirement: 10 CFR 50.34(f)(2)(xvii) requires that the design provide instrumentation to measure, record and readout in the control room for: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1)

Statement of Compliance: 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. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.34(f)(2)(xvii).

- Regulatory Requirement: 10 CFR 50.34(f)(3)(v)(A)(1) requires that containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, subarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 subarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the NRC Staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

  Statement of Compliance: 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, the BWRX-300 design will meet the requirements of 10 CFR 50.34(f)(3)(v)(A)(1).

5.1.2 10 CFR 50.44

10 CFR 50.44, Combustible gas control for nuclear power reactors, 10 CFR 50.44(c), Requirements for future water-cooled reactor applicants and licensees, apply to all water-cooled reactor CPs or OLs under this part, and to all water-cooled reactor design approvals, design certifications, combined licenses or manufacturing licenses under part 52 of this chapter, any of which are issued after October 16, 2003.

- Regulatory Requirement: 10 CFR 50.44(c)(1), Mixed atmosphere, requires that all containments must have a capability for ensuring a mixed atmosphere during design-basis and significant beyond design-basis accidents.

  Statement of Compliance: The design features of the BWRX-300 used to comply with this requirement include a dry, nitrogen-inerted containment with no subcompartments where combustible gas mixtures may accumulate. With an inerted containment, oxygen concentrations reaching flammable mixture levels in subcompartments become a concern even if the average concentration is below the limit. The only subcompartment that may experience this phenomenon is the containment head section above the refueling bellows. However, for DBAs, natural circulation due to the presence of the passive containment cooling and the very low oxygen concentration in the main section of containment prevent significant oxygen accumulation above the refueling bellows. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.44(c)(1) for DBAs.

Compliance with this requirement for beyond design basis events and severe accident managements are to be addressed in LTR NEDC-33921P, BWRX-300 Severe Accident Management [Reference 6.9].

- Regulatory Requirement: 10 CFR 50.44(c)(2), Combustible gas control, requires that all containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features.

  Statement of Compliance: The design feature of the BWRX-300 used to comply with this requirement includes the requirement for a dry, inerted containment. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.44(c)(2).

- Regulatory Requirement: 10 CFR 50.44(c)(3), Equipment Survivability, requires that containments that do not rely upon an inerted atmosphere to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to be considered must be equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding surrounding the active fuel region.

  Statement of Compliance: The design feature of the BWRX-300 used to comply with this requirement includes the requirement for a dry, inerted containment that does not rely upon combustible gas control to maintain safe shutdown and containment structural integrity. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.44(c)(3).

- Regulatory Requirement: 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).

  Statement of Compliance: 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).

- Regulatory Requirement: 10 CFR 50.44(c)(5), Structural analysis, requires that an applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC and include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning. Systems necessary to ensure containment integrity must also be demonstrated to perform their function under these conditions.

Statement of Compliance: 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. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.44(c)(5).

5.1.3 10 CFR 50.55a

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

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

  Statement of Compliance: The BWRX-300 containment and CIV design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, including any application for a CP under 10 CFR 50 or DCA under 10 CFR 52. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.55a.

5.1.4 10 CFR 50.63

10 CFR 50.63, Loss of all alternating current powers, requires that each light-water-cooled nuclear power plant licensed to operate under this part, each light-water-cooled nuclear power plant licensed under subpart C of 10 CFR part 52 after the Commission makes the finding under § 52.103(g) of this chapter, and each design for a light-water-cooled nuclear power plant approved under a standard design approval, standard design certification, and manufacturing license under part 52 of this chapter must be able to withstand for a specified duration and recover from an SBO as defined in § 50.2. The specified SBO duration shall be based on the following factors: (i) The redundancy of the onsite emergency ac power sources; (ii) The reliability of the onsite emergency ac power sources; (iii) The expected frequency of loss of offsite power; and (iv) The probable time needed to restore offsite power.

- Regulatory Requirement: 10 CFR 50.63(a)(2) requires that the reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled, and appropriate containment integrity is maintained in the event of an SBO for the specified duration. The capability for coping with an SBO of specified duration shall be determined by an appropriate coping analysis. Licensees are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

Statement of Compliance: The BWRX-300 design includes Class 1E battery-backed 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, and 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 PCCS for containment depressurization and heat removal is passive and does not require onsite or offsite electric power system operation, including Class 1E battery-backed DC power. CIV automatic actuation isolation functions do not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed DC power and then remain isolated for at least 72 hours without any further need of onsite or offsite electric power system operation. The coping analysis to demonstrate 72 hours will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.63.

5.1.5 10 CFR 50 Appendix A, GDC 1

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

  Statement of Compliance: The BWRX-300 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.

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

5.1.6 10 CFR 50 Appendix A, GDC 2

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

Statement of Compliance: The BWRX-300 containment and CIVs 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.

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

5.1.7 10 CFR 50 Appendix A, GDC 4

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

  Statement of Compliance: The BWRX-300 containment and CIVs design features, including the ICS, RPV isolation valve assemblies, PCCS, CIVs, containment structure, containment penetrations, piping, and instrumentation lines, are to be designed to effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents, and will consider the dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids. In addition, the dynamic effects of postulated pipe breaks are to be evaluated in the BWRX-300 design. As described in this LTR, the BWRX-300 design requirements include applying the design criteria from NUREG-0800, SRP, BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Part B, Items 1(ii)(1)(d) and (e), and Items 1(ii)(2) through (7) to eliminate postulating breaks and cracks in those portions of piping from containment wall to the outboard CIVs. Breaks and cracks in those portions of piping from the RPV isolation valves that function as the inboard CIVs to the containment wall remain postulated to occur, and the dynamic effects of those postulated pipe breaks are to be evaluated in the BWRX-300 design. Each RPV isolation valve assembly is connected directly to the reactor vessel using bolted flange connections classified as break exclusion areas. For these bolted flange connections, details of the threaded fastener design, leakage detection systems design, and inservice inspection requirements, demonstrate that the probability of gross rupture is extremely low. For piping connected to the RPV isolation valve assemblies, extending to the containment wall, the BWRX-300 design requirements include identifying postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, Part B, Item 1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2). The dynamic effects of postulated breaks and cracks in those portions of the piping beyond and including the outboard CIVs will be evaluated in future licensing activities. Internal containment flooding is to be evaluated during future licensing activities.

Valve qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as endorsed by NRC Regulatory Guide (RG) 1.100, will be addressed in the detailed design and the procurement process of the valves, and will be specified during future licensing activities.

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

5.1.8 10 CFR 50 Appendix A, GDC 5

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 5, Sharing of structures, systems and components requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

  Statement of Compliance: The BWRX-300 design does not include sharing of SSCs important to safety among each unit at multi-unit sites.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 5.

5.1.9 10 CFR 50 Appendix A, GDC 13

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 13, Instrumentation and control, requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the RCPB, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

  Statement of Compliance: BWRX-300 instrumentation and controls are to be provided to monitor variables and systems important to the containment and its associated systems over their anticipated ranges for normal operation for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety. These instrumentation and control systems will be described during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 13.

5.1.10 10 CFR 50 Appendix A, GDC 16

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 16, Containment design, requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

  Statement of Compliance: A leak-tight steel or reinforced concrete PCV or a combination of steel and reinforced concrete PCV encloses the RPV, including the RCPB and other branch connections for the NBS, and includes containment penetrations with essentially leak-tight isolation design features including CIVs, blind flanges, hatches, and electrical penetrations. A steel head encloses the opening in the top of the PCV for servicing and refueling the RPV. The major piping systems (main steam, feedwater, ICS, and other miscellaneous systems) are located in the upper PCV region. The lower PCV region encloses the lower portion of the RPV and encloses the cooling system ducts, FMCRDs) and other miscellaneous systems as well as providing maintenance space below the RPV. 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 analyses to demonstrate compliance will be provided during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 16.

5.1.11 10 CFR 50 Appendix A, GDC 38

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 38, Containment heat removal, requires that a system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels. 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, assuming a single failure.

  Statement of Compliance: Containment peak pressure and temperature is limited by condensation on containment walls and RPV heat removal by the ICS and containment heat removal by the PCCS by natural convection and condensation. The PCCS is to be shown to reduce containment peak pressure rapidly for a large break LOCA, which is the limiting BWRX-300 DBA. Heat is rejected to the reactor cavity pool above 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 safety analysis assumes that the small breaks [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time actuation using onsite Class 1E battery-backed DC power.

[[ ]] For RPV isolation and SBO events, containment pressure and temperature are limited by condensation on containment walls and containment heat removal by the PCCS, and by RPV decay heat removal by the ICS. The analyses to demonstrate compliance will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 38.

5.1.12 10 CFR 50 Appendix A, GDC 39

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 39, Inspection of containment heat removal system, requires that the containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

  Statement of Compliance: The components of the PCCS within containment to remove heat during a large break LOCA, are to be designed, fabricated, erected, and tested in accordance with ASME Code Section III, Class MC and Section XI, IWE requirements for design accessibility of welds 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 and GDC 1. In addition, means are to be provided to detect and identify the location of the source of containment leakage, including the CIVs, PCCS, non-essential and closed systems, and components of the ICS and RPV isolation valves, for components of the RCPB.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 39.

5.1.13 10 CFR 50 Appendix A, GDC 40

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 40, Testing of containment heat removal system, requires that the containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Statement of Compliance: The PCCS accomplishes the containment heat removal function while the ICS performs the RPV heat removal function during a large break LOCA. [[ ]] The PCCS is designed to be periodically pressure tested as part of the overall Containment Leakage Rate Testing Program to demonstrate structural and leak-tight integrity.

[[ ]] can be individually pressure and leak tested during maintenance or in-service inspection using various non-destructive methods. Functional and operability testing of the PCCS is not needed because there are no active components of the system. Performance is established for the range of in-containment environmental conditions following a LOCA.

The components of the PCCS are to be designed with sufficient margin to assure that these requirements for periodic pressure and functional testing to ensure leak-tight integrity and operational performance under normal operations and emergency events using normal and emergency power are met. In addition, the operation of the PCCS does not require offsite electric power system operation.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 40.

5.1.14 10 CFR 50 Appendix A, GDC 41

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 41, Containment atmosphere cleanup, requires that systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

  Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities 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) its safety function can be accomplished, assuming a single failure.

  Statement of Compliance: 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. Leakage from the containment after an accident will not result in exceeding 10 CFR 50.34(a)(1)(D) dose guidelines. Containment is constructed in the subterranean of a proposed site. As a result, containment leakage is expected to be contained for a considerable time before it leaks into the reactor cavity pool above containment. Oxygen monitors are installed for monitoring during and after a DBA. Containment integrity is maintained for the most severe accident management without employing the use of any combustible gas control system and includes suitable leak detection that is powered with safety-grade backup power. The analyses to demonstrate compliance will be provided during future licensing activities. Instrumentation requirements for beyond design basis events and severe accident managements are to be addressed in LTR NEDC-33921P, BWRX-300 Severe Accident Management [Reference 6.9].

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 41.

5.1.15 10 CFR 50 Appendix A, GDC 42

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 42, Inspection of containment atmosphere cleanup systems, requires that the containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

  Statement of Compliance: The Containment Atmosphere Cleanup Systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 42.

5.1.16 10 CFR 50 Appendix A, GDC 43

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 43, Testing of containment atmosphere cleanup systems, requires that the containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

  Statement of Compliance: Containment atmosphere is provided by the containment inerting system and is designed to be periodically tested.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 43.

5.1.17 10 CFR 50 Appendix A, GDC 50

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 50, Containment design bases, requires that the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that 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 any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Statement of Compliance: Containment design is based upon consideration of a full spectrum of postulated accidents that would result in the release of reactor coolant to the containment. These accidents are evaluated using TRACG code as boundary condition to GOTHIC to calculate containment response. These accidents include liquid, steam and partial (both steam and liquid) breaks. The evaluation of the containment design 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. The GOTHIC computer methodology for measuring containment response is provided in LTR NEDC-33922P BWRX-300 Containment Evaluation Method [Reference 6.5]. The analyses to demonstrate compliance will be provided during future licensing activities. The BWRX-300 containment structural 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. The maximum expected external pressure containment structural evaluation will demonstrate compliance to 10 CFR 50, Appendix A, GDC 38 and 50 and be provided in future licensing activities.

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

5.1.18 10 CFR 50 Appendix A, GDC 51

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 51, Fracture prevention of containment pressure boundary requires that the reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

  Statement of Compliance: A leaktight containment vessel encloses the RPV, the RCPB, and other branch connections for the reactor primary coolant system, including containment penetrationand isolation devices. The containment vessel is a reinforced concrete and steel cylindrical structure with a leaktight steel liner providing the primary containment boundary. The containment vessel structure consists of the top containment slab with a reactor building pool above, cylindrical containment wall, containment floor slab, RPV pedestal, and the basement. A steel head encloses the opening in the top of the containment vessel for servicing and refueling the RPV. The containment encloses the RPV, with the major piping (main steam, feedwater, ICS, PCCS, RPVs, CIVs and other miscellaneous systems) located in the upper containment region. The lower containment encloses the lower portion of the RPV and encloses the cooling sytem ducts, FMCRDs, and other miscellaneous systems as well as providing maintenance space below the RPV.

The containment vessel is a reinforced concrete structure with ferritic parts, such a liner and a removable head that is made of materials that have a Nil-Ductility Transition Temperature (NDTT) sufficiently below the minimum service temperature to assure that under operating, maintenance, testing, and postulated accident conditions, the ferritic materials behave in a nonbrittle manner considering the uncertainties in determining the material properties, stresses and size of flaws. The containment vessel is enclosed by and integrated with the subterranean strata at a proposed site. The preoperational test program and quality assurance program ensure the integrity of the containment and its ability to meet all normal operating and accident conditions.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 51.

5.1.19 10 CFR 50 Appendix A, GDC 52

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 52, Capability for containment leakage rate testing, requires that the reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

  Statement of Compliance: 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 in order to comply with 10 CFR 50, Appendix J and the guidance of RG 1.163.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 52.

5.1.20 10 CFR 50 Appendix A, GDC 53

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 53, Provisions for containment testing and inspection, requires that the reactor containment shall be designed to permit appropriate periodic inspection of all important areas, such as penetration, an appropriate surveillance program and periodic testing at containment design pressure of the leaktightness of penetration which have resilient seals and expansion bellows.

  Statement of Compliance: 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 50, Appendix J.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 53.

5.1.21 10 CFR 50 Appendix A, GDC 54

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 54, Piping systems penetrating containment, requires that piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Statement of Compliance: Piping systems penetrating the BWRX-300 containment are designed to provide the required isolation and testing capabilities. These piping systems are provided with test connections to allow periodic leak detection as necessary to determine if valve leakage is within acceptable limits.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 54.

5.1.22 10 CFR 50 Appendix A, GDC 55

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 55, Reactor coolant pressure boundary penetrating containment requires that each line that is part of the RCPB and that penetrates primary reactor containment shall be provided with CIVs as follows, 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:

  (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

  (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

  (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

  (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

  Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

  Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

  Statement of Compliance: As discussed in LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 6.2], the BWRX-300 RPV has [[

]] comply with the requirements of GDC 55. [[

]] The FMCRD are also connected to the RPV, but do not have accompanying RPV isolation valves based on being closed-system piping outside the PCV and having RCPB isolation (internal ball check valves) in the design of the drives. All BWRX-300 CIVs are designed to withstand the effects of the most severe natural phenomena.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 55.

5.1.23 10 CFR 50 Appendix A, GDC 56

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 56, Primary containment isolation, requires that each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with CIVs as follows, 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:

  (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Statement of Compliance: The BWRX-300 CIVs 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 are provided with two normally closed outside containment manual CIVs. 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 is provided with normally closed inside and outside containment automatic CIVs. The process gas and radiation monitoring system is a closed system outside containment, and is provided with normally open outside containment automatic CIVs because it is an essential system following beyond design basis events and severe accident managements. The floor drain sump line is provided with two normally closed outside containment automatic CIVs, because it is not practicable to include an inside containment automatic CIV 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.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 56.

5.1.24 10 CFR 50 Appendix A, GDC 57

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 57, Closed system isolation valves, requires that each line that penetrates primary reactor containment and is neither part of the RCPB nor connected directly to the containment atmosphere shall have at least one CIV which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

  Statement of Compliance: The BWRX-300 closed system CIVs 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 pneumatic nitrogen or air system and the quench tank supply system are provided with either normally open or normally closed inside and outside containment automatic CIVs. The service and breathing air system and demineralized water system are provided with normally closed inside and outside containment manual CIVs. The chilled water supply and return are provided with normally open outside containment automatic CIVs.

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Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 57.

5.1.25 10 CFR 50 Appendix A, GDC 64

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 64, Monitoring radioactivity releases, requires that means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

  Statement of Compliance: The BWRX-300 is provided with a process gas and radiation monitoring system that monitors radioactivity in containment for normal operations, AOOs, Infrequent Events (IEs), and DBAs.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 64.

5.1.26 10 CFR 50 Appendix J

- Regulatory Requirement: 10 CFR 50 Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, requires that one of the conditions of all OLs under this part and combined licenses under part 52 of this chapter for water-cooled power reactors as specified in § 50.54(o) is that primary reactor containments shall meet the containment leakage test requirements set forth in this appendix. These test requirements provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment of water-cooled power reactors, and establish the acceptance criteria for these tests. The purposes of the tests are to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases; and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. These test requirements may also be used for guidance in establishing appropriate containment leakage test requirements in technical specifications or associated bases for other types of nuclear power reactors.

  Statement of Compliance: 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 in order to comply with 10 CFR 50, Appendix J, and the guidance of RG 1.163.

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

5.2 Regulatory Guides

5.2.1 Regulatory Guide 1.7

Regulatory Guide (RG) 1.7, Control of Combustible Gas Concentrations in Containment, Rev. 3, describes methods acceptable to the NRC Staff for implementing the regulatory requirements of 10 CFR 50.44 for reactors subject to the provisions of Sections 50.44(b) or 50.44(c) with regard to control of combustible gases generated by beyond-design-basis accident that could be a risk-significant threat to containment integrity. For applicants and holders of a water-cooled reactor CP or OL under 10 CFR 50, and all applicants for a light-water reactor design approval or design certification, or combined license under 10 CFR Part 52 that are docketed after October 16, 2003, containments must have an inerted atmosphere or limit combustible gas concentrations in containment during and following an accident that releases an equivalent of combustible gas as would be generated from a 100% fuel-clad coolant reaction, uniformly distributed, to less than 10% (by volume) and must maintain containment structural integrity.

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. Compliance with the requirements of 10 CFR 50.44(c)(1) and 10 CFR 50.44(c)(3) for beyond design basis events and severe accident managements are addressed in LTR NEDC-33921P, BWRX-300 Severe Accident Management [Reference 6.9].

Therefore, the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.7.

5.2.2 Regulatory Guide 1.11

RG 1.11, Instrument Lines Penetrating the Primary Reactor Containment, Rev. 1, describes methods acceptable to the NRC Staff for use in establishing that a plant’s principal design criteria GDC 55 and GDC 56 require, in part, that each line that penetrates the primary reactor containment and that is part of the RCPB or connects directly to the containment atmosphere has at least one locked, closed isolation valve or one automatic isolation valve inside containment, and at least one locked, closed isolation valve or one automatic isolation valve outside containment (a simple check valve may not be used as the automatic isolation valve outside containment) “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.”

Instrument lines that penetrates the primary reactor containment and that is part of the RCPB or that penetrates the primary reactor containment and connects 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 leak-tight integrity.

BWRX-300 instrument lines penetrating primary reactor containment that are part of the RCPB or penetrate the primary reactor containment and connects directly to the containment atmosphere comply with Regulatory Position C.3. by providing EFCVs, and also comply with the requirements of GDC 55 and GDC 56.

Each line is provided with a self-actuated EFCV 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 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.

Therefore, the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.11.

5.2.3 Regulatory Guide 1.84

RG 1.84. Design, Fabrication and Materials Code Case Acceptability, ASME Section III, Rev. 38, describes the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section III, "Rules for Construction of Nuclear Power Plant Components" Code Cases that the U.S. NRC has approved for use as voluntary alternatives to the mandatory ASME BPV Code provisions that are incorporated by reference into Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities." This RG applies to reactor licensees subject to 10 CFR Part 50, Section 50.55a, "Codes and standards".

The BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.84, Rev. 38, and using the guidance conformance to RG 1.84, Rev. 33, as described in ESBWR 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. Code Case N-782 endorses the use of the Edition and Addenda of ASME Boiler and Pressure Vessel 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 this Code case will be provided in the BWRX-300 Preliminary Safety Analysis Report (PSAR) or future licensing activities.

5.2.4 Regulatory Guide 1.141

RG 1.141, Containment Isolation Provisions for Fluid Systems, Rev. 1, describes methods acceptable to the NRC Staff for use in implementing the regulatory requirements of GDC 55, Reactor coolant pressure boundary penetrating containment, GDC 56, Primary containment isolation, and GDC 57, Closed system isolation valves, with regard to establishing piping systems that penetrate the primary reactor containment be provided with isolation capabilities that reflect the importance to safety of isolating these piping systems.

The requirements and recommendations for the containment isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors, as specified in ANSI N271-1976, are generally acceptable and provide an adequate basis for use.

Sections 2.2.8, 5.1.22, 5.1.23, and 5.1.24 of this LTR describes how the design of the BWRX-300 CIVs complies with the requirements of GDC 55, GDC 56, and GDC 57. Compliance with the requirements of 10 CFR 50.55a is described in Section 5.1.3.

Therefore, the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.141.

5.2.5 Regulatory Guide 1.147

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

Section 4.1.3 of LTR 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 B&PV Code Section XI, Division 1, specifically apply during construction and operation activities of nuclear power plants for performance of inservice inspection activities. The GEH design process and associated administrative controls considers operating plant compliance to RG 1.147 guidance in performing examinations, inspections and tests of installed systems and components, and are 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 B&PV Section XI Code Cases endorsed 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.

5.2.6 Regulatory Guide 1.155

RG 1.155, Rev. 0, Station Blackout, describes methods acceptable to the NRC for complying with 10 CFR 50.63, Loss of All Alternating Current Power, that requires nuclear power plants be capable of coping with an SBO for specified duration, so that SSCs important to safety continue to function. "Station blackout" refers to the complete loss of alternating current electric power to the essential and nonessential switchgear buses concurrent with turbine trip and failure of the onsite emergency ac power system, but not the loss of available ac power to buses fed by station batteries through inverters or loss of power from "alternate ac sources". 10 CFR 50.63 requires all licensees and applicants to assess the capability of their plants to maintain adequate core cooling and appropriate containment integrity during an SBO and to have procedures to cope with such an event. This guide further presents a method acceptable to the NRC for determining the specified duration for which a plant should be able to withstand an SBO in accordance with these requirements.

The BWRX-300 is designed to safely shut down without ac power. Safety-related CIV position indication and closure are provided by safety-grade control power, closure, and position indication in case of SBO.

Therefore, the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.155.

5.2.7 Regulatory Guide 1.163

RG 1.163, Performance-Based Containment Leak Rate Test, Rev. 0, describes acceptable cost-effective methods, including setting test intervals, for implementing the safety objectives for performing containment leak testing in order to meet the requirements of 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." This regulatory guide approves an industry guideline that describes in detail a performance-based leak-test program, leakage-rate test methods, procedures, and analyses; the NRC Staff has determined this industry guideline to be an acceptable means of demonstrating compliance with the requirements of 10 CFR 50, Appendix J.

The BWRX-300 design is to include a containment leak rate testing program that addresses containment integrated leakage rate (Type A tests), containment penetration leakage tests (Type B tests), and CIV leakage rates (Type C tests) and complies with 10 CFR 50, Appendix J, Option A or Option B as per RG 1.163 and GDC 52, GDC 53, and GDC 54. The leakage rate testing capability is consistent with the testing requirements of ANS-56.8. Type A, B, and C tests are performed prior to operations and periodically thereafter to assure that 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.

Therefore, the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.163.

5.2.8 Regulatory Guide 1.192

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

Section 4.1.3 of LTR 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 IST activities. GEH design process and associated administrative controls considers operating plant compliance to RG 1.192 guidance in performing examinations, inspections and tests of installed systems and components, and are 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.

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.

5.2.9 Regulatory Guide 1.203

RG 1.203, Transient and Accident Analysis Methods, Rev. 0, describes a process that the NRC Staff considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. An additional benefit is that evaluation models that are developed using these guidelines will provide a more reliable framework for risk-informed regulation and a basis for estimating the uncertainty in understanding transient and accident behavior.

The Regulatory Position section describes a multi-step process for developing and assessing evaluation models, and provides guidance on related subjects, such as quality assurance, documentation, general purpose codes, and a graded approach to the process. The Implementation section then specifies the target audience for whom this guide is intended, as well as the extent to which this guide applies, and the Regulatory Analysis section presents the NRC Staff related rationale and conclusion. For convenience, this guide also includes definitions of terms that are used herein. Finally, Appendix A provides additional information important to Emergency Core Cooling System (ECCS) analysis, and Appendix B presents an example of the graded application of the evaluation model development and assessment process (EMDAP) for different analysis modification scenarios.

Section 3.4 of this LTR describes how the GOTHIC methodology code utilizes the Code Scaling, Applicability and Uncertainty in NUREG/CR-5249 and RG 1.203, and the Phenomenon Identification and Ranking Table graded approach of RG 1.203 for analyzing BWRX-300 containment response to transient and accident behavior.

Therefore, the BWRX-300 design conforms to the guidance, including regulatory positions of RG 1.203.

5.3 NUREG-0800 Standard Review Plan Guidance

5.3.1 Standard Review Plan 3.6.2

Standard Review Plan (SRP) 3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Rev. 3 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 GDC 4 Environmental and Dynamic Effects Design Bases be considered in the design structures, systems and components. This SRP 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. The guidance provides specific areas for review:

1. Defining break and crack locations and configurations
2. Analytical methods to define forcing functions, including jet thrust reaction at the postulated pipe break or crack location and jet impingement loadings on adjacent safety-related SSCs
3. The dynamic analysis methods used to verify the integrity and operability of mechanical components, component supports, and piping systems, including restraints and other protective devices under postulated pipe rupture loads
4. The implementation of criteria used in defining pipe break and crack locations and configurations
5. The criteria for dealing with special features such as augmented inservice inspection programs
6. The acceptability of the analysis results, including jet thrust and impingement forcing functions and pipe-whip dynamic effects
7. The design adequacy of SSCs to ensure that the intended design functions will not be impaired to an unacceptable level of integrity or operability as a result of pipe whip or jet impingement loadings.

The BWRX-300 containment isolation system SSCs will conform to the guidance of the SRP, as well as meeting the requirements of GDC 4. The design of the containment structure, component arrangement, pipe runs, pipe whip restraints, and compartmentalization are 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 are engineered to preclude damage based upon break evaluation. A complete description of compliance to the SRP and associated branch technical positions, 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.

5.3.2 Standard Review Plan 3.9.6

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

B&PV Code Section III. The review includes other pumps, valves and dynamic restraints not categorized as ASME BPV Code Class 1. 2 or 3 that have safety-related function. Conformance with the specific guidance in Subsection II of this SRP section will provide reasonable assurance that the functional design and qualification of pumps, valves and dynamic restraints within the scope of this SRP section and their associated IST programs satisfy the applicable requirements of Section 50.55a, "Codes and Standards," of Title 10 of the Code of Federal Regulations, particularly the IST program requirements of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) 4; General Design Criterion (GDC) 1, "Quality Standards and Records," GDC 2, "Design Bases for Protection against Natural Phenomena," GDC 4, "Environmental and Dynamic Effects Design Bases," GDC 14, "Reactor Coolant Pressure Boundary," GDC 15, "Reactor Coolant System Design," GDC 37, "Testing of Emergency Core Cooling System," GDC 40, "Testing of Containment Heat Removal System," GDC 43, "Testing of Containment Atmosphere Cleanup Systems," GDC 46, "Testing of Cooling Water System," and GDC 54, "Systems Penetrating Containment," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities;" Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50; 10 CFR 52.47(b)(1), 10 CFR 52.79(a)(11), and 10 CFR 52.80(a).

The containment isolation valves are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, including any application for a construction permit under 10 CFR 50 or design certification application under 10 CFR 52. The requirements of 10 CFR 50.55a, are to be implemented during detailed design of the safety-related components of containment isolation. Therefore, the existing SRP provides adequate guidance to use during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

5.3.3 Standard Review Plan 6.2.1

SRP 6.2.1, Containment Functional Design, Rev. 3, states that the areas of review include the containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant, steam line, or feedwater line break accidents. The containment structure must also maintain functional integrity in the long term following a postulated accident, i.e., it must remain a low leakage barrier against the release of fission products for as long as postulated accident conditions require.

The design and sizing of containment systems are largely based on the pressure and temperature conditions which result from release of the reactor coolant in the event of a LOCA. The containment design basis includes the effects of stored energy in the reactor coolant system, decay energy, and energy from other sources such as the secondary system, and metal-water reactions including the recombination of hydrogen and oxygen. The containment system is not required to be a complete and independent safeguard against a LOCA by itself, but functions to contain any fission products released while the ECCS cools the reactor core. The evaluation of a containment functional design includes calculation of the various effects associated with the postulated rupture in the primary or secondary coolant system piping. The subsequent thermodynamic effects in the containment resulting from the release of the coolant mass and energy are determined from a solution of the incremental space and time-dependent energy, mass, and momentum conservation equations. The basic functional design requirements for containment are given in GDC 4, GDC 16, GDC 50, and 10 CFR 50, Appendix K.

The various containment types and aspects to be reviewed under this SRP section have been separated and assigned to a set of other SRP sections. The BWRX-300 containment design is affected by the guidance provided in SRP 6.2.1.1.A, SRP 6.2.1.1.C, SRP 6.2.1.2, SRP 6.2.1.3, SRP 6.2.1.4, and SRP 6.2.1.5. The following SRPs are not applicable to the BWRX-300 design and discussed specifically in subsequent LTR sections:

SRP 6.2.1.1.C Pressure-Suppression Type BWR Containments – the BWRX-300 does not utilize a pressure-suppression pool for maintaining containment pressure and temperature from the dynamic effects of LOCA.

SRP 6.2.1.2 Subcompartment Analysis – the BWRX-300 does not have subcompartments in the design that contain large bore high energy lines.

SRP 6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures – the BWRX-300 design does not utilize secondary system piping.

SRP 6.2.1.5 Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies – the BWRX-300 does not utilize emergency core cooling for maintaining containing pressure during design basis events. Containment pressure is maintained by the PCCS for AOOs, IEs and DBAs.

The design features of the BWRX-300 containment include:

- Underground (subterranean) steel or reinforced concrete PCV
- Dry containment with no suppression pool
- Nitrogen-inerted containment
- Passive containment heat removal for PCCS for design basis events; fan coolers for normal operations
- No subcompartments with large bore high energy lines
- ICS pools and reactor cavity pool for PCCS located above containment
- Fewer penetrations

Specific discussions under Section I, Areas of Review, are addressed in meeting the intent of the affected individual SRP sections previously delineated.

Specific discussions under Section II, Acceptance Criteria, for generic acceptance criteria that are affected are discussed in the affected individual SRP sections previously delineated.

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 Section 2.0 of this LTR. Therefore, GEH recommends that the existing SRP may be used during future review of a BWRX-300 10 CFR 52 DCA if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

5.3.4 Standard Review Plan 6.2.1.1.A

SRP 6.2.1.1.A, PWR Dry Containments, Including Subatmospheric Containments, Rev. 3, states that the areas of review include: (1) the temperature and pressure conditions in the containment due to a spectrum (including break size and location) of postulated loss-of-coolant accidents (i.e., reactor coolant system pipe breaks) and secondary system steam and feedwater line breaks; (2) the maximum expected external pressure to which the containment may be subjected; (3) the minimum containment pressure that is used in analyses of ECCS capability; (4) the effectiveness of static and active heat removal mechanisms; (5) the pressure conditions within subcompartments that act on system components and supports due to high energy line breaks; and (6) the range and accuracy of instrumentation that is provided to monitor and record containment conditions during and following an accident.

The BWRX-300 containment is nitrogen-inerted with no suppression pool to mitigate the dynamic effects of DBAs. Therefore, SRP 6.2.1.1.C no longer applies to this GEH design. As a result, SRP 6.2.1.1.A was selected to use as guidance inasmuch as the guidance and acceptance criteria described within reflect the BWRX-300 design. It should be noted that while SRP 6.2.1.1.A better reflects the design of the BWRX-300, portions of this guidance document are also 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; (3) there are no secondary systems utilized in the BWRX-300 design. The design requirements for the PCCS to reject heat to the reactor cavity pool above containment during DBAs is described in Section 2.2.8.

Section 3.0 discusses the TRACG and GOTHIC computer code methodologies utilized to analyze mass and energy release from the RPV that provide boundary conditions for the GOTHIC code to analyze the containment response for a spectrum of break sizes and locations for postulated loss of coolant accidents. The GOTHIC computer methodology for measuring containment response is provided in LTR NEDC-33922P BWRX-300 Containment Evaluation Method [Reference 6.5]. The containment performance acceptance criteria are discussed in Section 4.0. All instrumentation is to be provided with accuracy and ranges for the most severe accident management scenario and record containment conditions during and following an accident. Requirements for beyond design basis events and Severe Accident Managements are to be addressed in LTR NEDC-33921P, BWRX-300 Severe Accident Management [Reference 6.9].

Specific discussions under Section I, Areas of Review, are addressed in meeting the intent of the affected individual SRP sections previously delineated.

Specific discussions under Section II, Acceptance Criteria, for generic acceptance criteria that are affected are discussed in the affected individual SRP sections previously delineated.

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

5.3.5 Standard Review Plan 6.2.1.1.C

SRP 6.2.1.1.C, Pressure-Suppression Type BWR Containments, Rev. 7, provides guidance in evaluating the temperature and pressure condition effects in the drywell and wetwell of BWR containments incorporating a suppression pool.

The BWRX-300 design does not employ the use of a drywell and wetwell incorporating a suppression pool. Therefore, the acceptance criteria associated with these guidelines are not applicable for the BWRX-300 design.

5.3.6 Standard Review Plan 6.2.1.2

SRP 6.2.1.2, Subcompartment Analysis, Rev. 3, includes review for 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.

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.

5.3.7 Standard Review Plan 6.2.1.3

SRP 6.2.1.3, Mass and Energy Release Analysis for Postulated Loss-Of-Coolant Accidents (LOCAs), Rev. 3, includes mass and energy release data is reviewed to ensure that containment and subcompartment functional design is designed withstand the energy released to containment from all sources, and provide a mass and energy release rate calculation for the initial blowdown phase of the accident. The GDC 50 acceptance criteria is met by ensuring that the containment and subcompartments are designed with sufficient margin to accommodate the calculated peak pressure and temperature resulting from any LOCA without exceeding the design leakage rate.

Mass and energy release are calculated using GEH's TRACG code for RPV neutronics and thermal-hydraulics calculations. The method accounts for the uncertainties and compensates for them by biases in the modeling parameters and in the plant parameters. The BWRX-300 containment analysis method utilizes only those sections of the ESBWR Containment/LOCA analysis method related to the RPV and break flow, and correlations and biases. Containment back pressure and ingress of steam/gas mixture are specified as boundary conditions to the TRACG model. Previous TRACG Containment/LOCA submittals for the models and qualification of TRACG are applicable to BWRX-300. A complete discussion of TRACG for the BWRX-300 is found in Section 3.1.

The specific discussions under Section I, Areas of Review, are addressed in meeting the intent of the affected individual SRP sections previously delineated.

Specific discussions under Section II, Acceptance Criteria, for generic acceptance criteria that are affected are discussed in the affected individual SRP sections previously delineated.

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

5.3.8 Standard Review Plan 6.2.1.4

SRP 6.2.1.4, Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures, Rev. 2, provides guidance for the review of the mass and energy release for secondary system pipe ruptures to evaluate the containment and subcompartment functional design in order to comply with GDC 50 for postulated pressurized-water reactor PWR secondary system pipe ruptures to ensure the reactor containment structure, including access openings, penetrations, and the containment heat removal system can withstand the calculated pressure and temperature conditions resulting from any LOCA.

The BWRX-300 design does not employ the use of any secondary systems for feedwater or steam production. Containment temperature and pressure are removed by the PCCS for all postulated DBAs. See Section 2.2.8 for complete discussion of PCCS heat removal capability. Therefore, the acceptance criteria associated with these guidelines are not applicable for the BWRX-300 design.

5.3.9 Standard Review Plan 6.2.1.5

SRP 6.2.1.5, Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies, Rev. 3, provides guidance for compliance to 10 CFR 50.46 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 during the time until the core is reflooded.

The BWRX-300 design includes the use of RPV isolation valves and the ICS to perform the ECCS design functions as described in LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 6.2]. For large break LOCAs, containment pressure does not affect the performance of the ECCS design functions as the [[

]] Therefore, the acceptance criteria associated with these guidelines are not applicable for the BWRX-300 design.

5.3.10 Standard Review Plan 6.2.2

SRP 6.2.2, Containment Heat Removal Systems, Rev. 5, provides guidance for the review of containment heat removal under post-accident conditions to ensure conformance with the requirements of GDC 38, GDC 39, GDC 40, and 10 CFR 50.46(b)(5).

Specific Areas of Review under Section I include: 1. the consequences of single component malfunctions; 2. analyses of Net Positive Suction Head (NPSH) to the ECCS and containment heat removal pumps; 3. the analyses of the heat removal capability of the spray water system; 4. the analyses of the heat removal of the Residual Heat Removal (RHR) and fan cooler heat exchangers; 5. the potential for surface fouling and flow blockage of the fan cooler, recirculation , and RHR heat exchangers and the effect on heat exchanger performance; 6. the design provisions and proposed program for periodic inservice inspection and operability testing of each system or component; 7. the design of sumps and water sources for ECCS and containment spray system performance; and 8. the effects of accident-generated debris, including loss of long-term cooling capability resulting from LOCA-generated and latent debris.

The BWRX-300 does not employ the use of a spray water system, ECCS, or a sump in the design to actively remove heat or pressure within containment, [[

]]. As a result, Section 1, Areas of Review, for: 2. analyses of NPSH to the ECCS and containment heat removal pumps; 3. heat removal capability of the spray water system; 7. the design of sumps and water sources and ECCS and containment spray performance; and 8. the effects of accident-generated debris are not applicable to the BWRX-300 design. Section II. Acceptance Criteria, GDC 38, GDC 39, and GDC 40 for the ability to rapidly reduce containment pressure and temperature following a LOCA and maintain these indicators at acceptably low levels, including inspection and testing of containment heat removal systems are met for the BWRX-300 design by the PCCS rapidly reducing containment pressure and temperature following the most severe LOCA with Loss of Offsite Power (LOOP), assuming a single active failure and maintaining pressure and temperature at acceptably low levels.

BWRX-300 conformance to the requirements of 10 CFR 50.46(b)(5) for long-term core cooling is addressed in LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 6.2].

With exception to the Areas of Review identified as not being applicable, GEH recommends that the existing SRP may be used during future review of a BWRX-300 10 CFR 52 DCA if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

5.3.11 Standard Review Plan 6.2.3

SRP 6.2.3, Secondary Containment Functional Design, Rev. 3, provides guidance for analyzing pressure and temperature response of a secondary containment, including the outer containment structure of dual containment plants, and systems that mitigate the radiological consequences of postulated accidents in order to meet the acceptance criteria of GDC 4, GDC 16, GDC 43, and 10 CFR 50, Appendix J, as it relates to secondary containment leakage rate testing.

The BWRX-300 design does not employ the use of a secondary containment or dual containment. Therefore, the acceptance criteria associated with these guidelines are not applicable for the BWRX-300 design.

5.3.12 Standard Review Plan 6.2.4

SRP 6.2.4, Containment Isolation System, Rev. 3, provides guidance for containment isolation to prevent or limit the escape of fission products from postulated accidents. Section I. Areas of Review include: the number and location of isolation valves, the position of these valves under normal operation, post-accident conditions, valve operator power failures, associated actuation signals, valve closure time basis, redundancy, and the acceptability of closed piping systems inside containment as isolation barriers. Additionally, the areas of review include the protection of SSCs from missiles, pipe whip and earthquakes as well as environmental conditions inside and outside containment. Further, review areas include detection for need to isolate, associated technical specifications, containment atmosphere prior to isolation valve closure, containment purging/venting while keeping ALARA for occupational exposure, isolation under accident conditions and containment isolation and valve indication for SBO.

Section II. Acceptance Criteria include: GDC 1 for designing, fabricating, erecting and testing SSCs to quality standards, GDC 4 for designing SSCs to accommodate the effects of environmental conditions associated with normal operations, maintenance, and postulated accidents and consideration of the effects of missiles, pipe whipping and discharging fluids, GDC 16 as it relates to maintaining a leak-tight barrier against the uncontrolled release of radioactivity to the environment; GDC 54 as it relates to piping systems penetrating containment having leak detection, isolation and containment capabilities which reflect the importance of safety; GDC 55 and GDC 56 as it relates to isolation valves penetrating (GDC 55) the containment boundary as part of the RCPB or as direct connections to the containment atmosphere (GDC 56); and GDC 57 as it relates to lines penetrating the primary containment and are neither part of the Reactor Coolant System (RCS) boundary nor connected directly to containment atmosphere.

CIVs provide the necessary isolation of the containment in the event of accidents or other condition and prevent the unfiltered release of containment contents that would exceed 10 CFR 50.34(a)(1)(D) limits. Leak-tightness of the valves shall be verified by Type C tests. Capability for rapid closure or isolation of pipes or ducts that penetrate the containment is performed by means or devices that provide a containment barrier to limit leakage within permissible limits. The design of isolation valves for lines penetrating containment follow the requirements of GDC 55, GDC 56, and GDC 57. Compliance to GDC 55, GDC 56, and GDC 57 is discussed in Section 2.2.7 and Sections 5.1.22, 5.1.23, and 5.1.24. The use of [[

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Isolation valves for instrument lines that penetrate containment conform to the requirements of RG 1.11. Isolation valves, actuators and controls are protected against the loss of their safety-related function from missiles and postulated effects of high and moderate energy line ruptures. Design of the CIVs, and associated piping and penetrations will meet the requirements of seismic Category I components, and the ASME Boiler and Pressure Vessel Code, Section III, Class 1 or 2, in accordance with their quality group classification. The design of the control functions for automatic CIVs ensure that resetting the isolation signal shall not result in the automatic reopening of CIVs. Penetrations with trapped liquid volume between the isolation valves have adequate relief for thermally-induced pressurization. Piping penetrations through the containment are designed to the requirements of Subsection NE, (MC component) of Section III of the ASME Code, and comply with the requirements of 10 CFR 50, Appendix A, GDC 54.

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

5.3.13 Standard Review Plan 6.2.5

SRP 6.2.5, Combustible Gas Control in Containment, Rev. 3, provides guidance for complying with 10 CFR 50.44 "Combustible Gas Control for Nuclear Power Reactors" with RG 1.7, Rev. 3, "Control of Combustible Gas Concentration in Containment", describing methods acceptable to the NRC for implementing 10 CFR 50.44. The review includes the control of combustible gases in the containment following a beyond-design-basis accident involving 100 percent fuel clad-coolant reaction or postulated accident to ensure conformance with the requirements of GDC 5, GDC 41, GDC 42, GDC 43 and 10 CFR 50.44. As described in Section I, Areas of Review, the review includes the following general areas:

1. Production and accumulation of combustible gases within the containment following a BDBA.
2. The capability to monitor combustible gas concentration within containment, and, for inerted containments, oxygen concentrations within containment.
3. The capability to monitor combustible gas concentration within containment, and for inerted containments, oxygen concentrations within containment.
4. The capability to reduce combustible gas concentration within containment by suitable means, such as igniters.

Specific areas of review include:

1. Analysis of combustible gas (e.g., hydrogen, carbon monoxide, oxygen) production and accumulation within the containment following a beyond-design-basis accident.
2. Analysis of the functional capability of the systems or passive design features provided to mix the combustible gas within the containment.
3. Analysis of the functional capability of the systems provided to reduce combustible gas concentrations within the containment.
4. Analyses of the capability of systems or system components to withstand dynamic effects, such as transient differential pressures that would occur early in the blowdown phase of an accident.
5. Analyses of the consequences of single active component malfunctions, to meet GDC 41.
6. The quality classification of each system.
7. The seismic design classification of each system.
8. The results of qualification tests performed on system components to demonstrate functional capability.
9. The design provisions and proposed program (including Technical Specifications at the OL or COL stage of review) for periodic inservice inspection, operability testing, and leakage rate testing of each system or component.
10. The functional aspects of instrumentation provided to monitor system or system component performance.

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. Therefore, the BWRX-300 conforms to the acceptance criteria associated with these guidelines for DBAs. Compliance with the requirements of 10 CFR 50.44(c)(1) and 10 CFR 50.44(c)(3) for beyond design basis events and severe accident managements are addressed in LTR NEDC-33921P, BWRX-300 Severe Accident Management [Reference 6.9].

5.3.14 Standard Review Plan 6.2.6

SRP 6.2.6, Containment Leakage Testing, Rev. 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.

The BWRX-300 design conforms to the guidance of SRP 6.2.6 in the same manner as described in the ESBWR Design Control Document, Tier 2, 26A6642AT, Rev. 10, April 2014, Section 6.2.6 [Reference 6.4], and the related safety evaluation from NUREG-1966, Volume 2, Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design, Section 6.2.6.

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

5.3.15 Standard Review Plan 6.2.7

SRP 6.2.7, Fracture Prevention of Containment Pressure Boundary, Rev. 1, provides guidance on ensuring that the reactor containment pressure boundary that consists of ferritic steel parts that sustain loading and provide a pressure boundary in the performance of the containment function under the operating, maintenance, testing and postulated accident conditions cited by GDC 51 are met. Typically, the Section I. Areas of Review, provides guidance for the review of ferritic materials of components such as freestanding containment vessels, equipment hatches, personnel airlocks, heads of primary containment drywells, tori, containment penetration sleeves, process pipes, end closure caps and flued heads, and penetrating-piping systems connecting to penetration process pipes and extending to and including the system isolation valves.

Specific area of review includes: the containment vessel and all penetration assemblies or appurtenances attached to the containment vessel; all piping, pumps and valves attached to the containment vessel, or to penetration assemblies out to and including the pressure boundary materials of any valves required to isolate the system and provide a pressure boundary for the containment function.

The BWRX-300 design conforms to the guidance of SRP 6.2.7 in the same manner as described in the ESBWR Design Control Document, Tier 2, 26A6642AT, Rev. 10, April 2014, Section 6.2.7 [Reference 6.4], and the related safety evaluation from NUREG-1966, Volume 2, Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design, Section 6.2.7.

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

5.4 Generic Issues

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

5.4.1 NUREG-0737

NUREG-0737, Clarification of TMI Action Plan Requirements, November 1980, contains requirements approved for implementation by the NRC Commissioners as a result of the accident at TMI Unit 2. The NRC Commission subsequently recommended that certain of these requirements be added to the 10 CFR 50 regulations, which were subsequently implemented in 10 CFR 50.34(f). Compliance with the items that are related to containment performance are discussed in Section 5.1.1. Compliance with the items that are related to RPV isolation and the mitigation of the effects of a LOCA are discussed in Section 4.1.1 of LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 6.2].

5.5 Operational Experience and Generic Communications

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

5.5.1 Generic Letter 83-02

Generic Letter 83-02, NUREG-0737 Technical Specifications, dated January 10, 1983, contains a request for information for the current BWR licensees regarding NUREG-0737 items for which technical specifications are required, including guidance on the scope of a specification which the 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.

5.5.2 Generic Letter 95-07

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

6.0 REFERENCES

6.1 26A6642AP Revision 10, "ESBWR Design Control Document, Tier 2, Chapter 4 Reactor," GE Hitachi Nuclear Energy, April 2014

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

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

6.4 26A6642AT Revision 10, "ESBWR Design Control Document, Tier 2, Chapter 6 Engineered Safety Features," GE Hitachi Nuclear Energy, April 2014

6.5 NEDC-33922P, "BWRX-300 Containment Evaluation Method"

6.6 NEDC-33083P-A Revision 1, "TRACG Application for ESBWR," September 2010

6.7 NEA/CSNI/R3(2014), "Containment Code Validation Matrix," May 2014

6.8 SMSAB-02-02, "An Assessment of CONTAIN 2.0: A Focus on Containment Thermal Hydraulics (Including Hydrogen Distributions)," July 2002

6.9 NEDC-33921P "BWRX-300 Severe Accident Management"

6.10 GEH Letter M200090, "Response to Request for Additional Information (eRAI) 9745 for Licensing Topical Report NEDC-33911P, Revision 0, BWRX-300 Containment Performance," dated June 26, 2020

6.11 GEH Letter M200096, "Responses to Requests for Additional Information (eRAIs) 9746, 9760, 9764, 9765, 9766, and 9767 for Licensing Topical Report NEDC-33911P, Revision 0, BWRX-300 Containment Performance," dated July 24, 2020

6.12 GEH Letter M200100, "Responses to Request for Additional Information (eRAI) 9758 for Licensing Topical Report NEDC-33911P, Revision 0, BWRX-300 Containment Performance," dated July 31, 2020

6.13 GEH Letter M200109, "Supplemental Response to Request for Additional Information (eRAI) 9746 for Licensing Topical Report NEDC-33911P, Revision 0, BWRX-300 Containment Performance," dated August 7, 2020

6.14 Letter from (NRC) to (GEH), Subject: Final Safety Evaluation for GE-Hitachi Licensing Topical Report NEDC-33911P, Revision 0, Supplement 1, "BWRX-300 Containment Performance," ADAMS Accession Number ML21008A369, March 12, 2021

Appendix A
GEH Responses to NRC RAIs on NEDC-33911P, Revision 0

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eRAI No.: 9745
Date of eRAI Issue: 06/22/20

NRC Question 03.09.06-15

SER Section 3.9.6: NRC Standard Review Plan (SRP) Section 3.9.6 specifies that the NRC staff will review the functional design, qualification, and inservice testing (IST) program for pumps, valves, and dynamic restraints to perform their design-basis safety functions. The NRC regulations in 10 CFR Part 50, Appendices A and B, and 10 CFR 50.55a include requirements for the capability of pumps, valves, and dynamic restraints to perform their safety functions. Section 2.2.7, "Containment Isolation Valves," in NEDC-33911 describes the various containment isolation valves (CIVs) used in the BWRX-300 nuclear power plant. The capability of CIVs to perform their design-basis functions is safety significant to provide assurance that the containment of the BWRX-300 reactor can be safely isolated and prevent radioactive release to the environment that exceeds regulatory requirements. In accordance with 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 54, 55, 56, and 57 for the various CIVs to be used in the BWRX-300, the NRC staff requests the following:

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

(b) Valve and actuator types,

(c) Valve size,

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

(e) Plans for valve and actuator diversity,

(f) Incorporation of lessons learned from CIV performance,

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

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

(i) OM Code leakage classification.

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

GEH Response to NRC Question 03.09.06-15

Although detailed design of the BWRX-300 containment isolation valves (CIVs) has not yet been completed, the design functions and features of the CIVs are anticipated to be similar to what is described in Section 6.2.4 of the Economically Simplified Boiling Water Reactor (ESBWR) CIVs described in Design Control Document (DCD) Tier 2, 26A6642AT. Compliance to the requirements of 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 54, 55, 56, and 57 for the various CIVs to be used in the BWRX-300 is anticipated to be the same as what is described in Section 6.2.4 of the ESBWR DCD.

(a) GEH does not anticipate any FOAK features for the BWRX-300 CIVs. If any FOAK features are identified during the CIV detailed design, they will be specified during future licensing activities.

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

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

(d) Qualification, such as compliance with ASME Standard QME-1 as endorsed by NRC Regulatory Guide (RG) 1.100, will be addressed in the detailed design and the procurement process of the valves, and will be specified during future licensing activities.

(e) As described in Licensing Topical Report (LTR) NEDC-33911P, Section 5.1.22, for penetrations where RPV isolation valves are credited as one of the containment isolation valves, diversity for the RPV automatic actuation signals is accomplished by actuation from separate and diverse control systems that are single failure proof. In other penetrations where two containment isolation valves are used that have automatic isolation, diverse actuation signals are applied to ensure the function is achieved. NEDC-33911P, Section 2.2.7, Design Requirements, will be revised to address diversity of CIVs. Detailed design of the CIVs will be specified during future licensing activities.

(f) Incorporation of lessons learned from international operating experience for CIVs will be addressed in the detailed design of the valves and will be specified during future licensing activities.

(g) Accessibility for IST activities in accordance with 10 CFR 50.55a will be addressed in the detailed system design layout of the valves and will consider IST requirements like those described in ESBWR DCD, Tier 2, 26A6642AK, Rev. 10, April 2014, Table 3.9-8. The CIV design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, either by GEH in support of a 10 CFR 52 DCA or by a license applicant for requesting a CP and OL under 10 CFR 50 or COL under 10 CFR 52. The specific IST requirements for the BWRX-300 design will be specified during future licensing activities.

(h) Design features to avoid thermal binding or pressure locking of the valves are not necessary for the BWRX-300 CIVs. The safety function of the CIVs, with exception of the isolation condenser valves that fail as is, is to close as opposed to having an opening safety function. Automatic actuation of the CIVs performs the function of mitigating the effects of a LOCA, AOOs, and IEs. Detailed valve design will consider the potential for excess internal pressurization due to accident conditions heating of the valves and their actuator mechanisms, and any effected CIV design will be provided the necessary and appropriate overpressure protection.

    NEDC-33911P will be revised to include a discussion for Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, as this operating experience may be applicable to the detailed design of the CIVs. However, the specific IST requirements for the BWRX-300 CIV design will be specified during future licensing activities.

(i) ASME OM code leakage requirements will be ASME OM paragraph ISTC-1300. The BWRX-300 containment isolation valves are expected to have specified leakage criteria, and are expected to fall under Category A requirements of the OM Code. However, there may be some valves designated at Category C, such as relief valves or check valves, when the individual system designs are performed. The OM Category assignments will be confirmed during the detailed system design and the valve accessories will be designed and selected accordingly with the appropriate leakage requirements applied for their CIV isolation function. The specific OM code leakage requirements for the BWRX-300 design will be specified during future licensing activities.

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 0, will be revised to reflect the addition of new bullet to subsection 2.2.7 for valve diversity and subsection 5.5.2 that addresses design features to avoid thermal binding discussed in Generic Letter 95-07:

…

2.2.7 Containment Isolation Valves

. . .

Design Requirements:

. . .

- <u>Diversity for penetrations where RPV isolation valves are credited as one of the containment isolation valves is accomplished by actuation from separate and diverse control systems that are single failure proof. In other penetrations where two containment isolation valves are used that have automatic isolation, diverse actuation signals are applied to ensure the function is achieved.</u>

…

<u>**5.5.2 Generic Letter 95-07**</u>

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

eRAI No.: 9745
Date of eRAI Issue: 06/22/20

NRC Question 03.09.06-16

SER Section 3.9.6: NRC Standard Review Plan (SRP) Section 3.9.6 specifies that the NRC staff will review the functional design, qualification, and inservice testing (IST) program for pumps, valves, and dynamic restraints to perform their design-basis safety functions. The NRC regulations in 10 CFR Part 50, Appendices A and B, and 10 CFR 50.55a include requirements for the capability of pumps, valves, and dynamic restraints to perform their safety functions. Section 2.2.8, "Passive Containment Cooling System (PCCS)," in NEDC-33911 provides an overview description of the PCCS for the BWRX-300 that [[ ]]. The PCCS provides a [[ ]]. However, the topical report is not clear regarding the use of valves or pumps in the PCCS. In accordance with 10 CFR Part 50, Appendix A, GDC 1, 2, 4, 54, 55, 56, and 57, the NRC staff requests that GEH describe any valves or pumps that are part of the PCCS for the BWRX-300, and their design and qualification.

GEH Response to NRC Question 03.09.06-16

The BWRX-300 PCCS does not employ the use of any valves or pumps to perform their design-basis safety function as described in LTR NEDC-33911P, Section 2.2.8.

Proposed Changes to NEDC-33911P, Revision 0

None

eRAI No.: 9745
Date of eRAI Issue: 06/22/20

NRC Question 03.09.06-17

SER Section 3.9.6: Section 5.1.1, "10 CFR 50.34(f)," in NEDC-33911 discusses compliance with specific requirements in 10 CFR 50.34(f) related to the TMI-2 accident lessons learned. In describing compliance with 10 CFR 50.34(f)(2)(xiv) with respect to containment isolation systems, NEDC-33911 indicates that the CIVs do not reopen on resetting of the signals. NRC Standard Review Plan (SRP) Section 3.9.6 specifies that the NRC staff will review the functional design, qualification, and inservice testing (IST) program for pumps, valves, and dynamic restraints to perform their design-basis safety functions. The NRC regulations in 10 CFR Part 50, Appendices A and B, and 10 CFR 50.55a include requirements for the capability of pumps, valves, and dynamic restraints to perform their safety functions. The capability of CIVs to perform their design-basis functions is safety significant to provide assurance that the containment of the BWRX-300 reactor can be safely isolated and prevent radioactive release to the environment that exceeds regulatory requirements. The NRC staff requests that GEH clarify how the CIV designs will prevent valve disc movement during their long-term isolation function.

GEH Response to NRC Question 03.09.06-17

The detailed design of the containment isolation valves, and valve actuators has not been completed. Positive mechanical means shall be required in the design of the valve actuators to ensure that upon automatic actuation or a loss of signal or control power to both valves, the valves will be maintained in the required post-accident valve positions. These requirements are to be implemented during detailed design of the valves and actuators. Therefore, additional information requiring the use of positive mechanical means in the design of the valve actuators to maintain these valves in their required post-accident valve positions will be added to NEDC-33911P.

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 0, Section 2.2.7 will be revised to add the design requirements for the use of positive mechanical means in the design of the valve actuators to maintain these valves in their required post-accident valve positions:

…

2.2.7 Containment Isolation Valves

. . .

Design Requirements:

. . .

- <u>The CIV isolation valves for main steam line, feedwater, shutdown cooling, and reactor water cleanup shall fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means.</u>

- [[ ]] with valve actuators designed to maintain the valves in their as-is position by positive mechanical means.
- All other CIV penetration configurations will be designed with valve actuators with positive mechanical means to ensure that upon automatic actuation or a loss of signal or control power to both valves, the valves will be maintained in the required post-accident valve position.

eRAI No.: 9745
Date of eRAI Issue: 06/22/20

NRC Question 03.09.06-18

SER Section 3.9.6: NRC Standard Review Plan (SRP) Section 3.9.6 specifies that the NRC staff will review the functional design, qualification, and inservice testing (IST) program for pumps, valves, and dynamic restraints to perform their design-basis safety functions. The NRC regulations in 10 CFR Part 50, Appendices A and B, and 10 CFR 50.55a include requirements for the capability of pumps, valves, and dynamic restraints to perform their safety functions. The NRC staff review under SRP Section 3.9.6 includes planned alternatives and their safety significance to the editions and addenda of the ASME *Boiler and Pressure Vessel Code* (BPV Code) and ASME *Operation and Maintenance of Nuclear Power Plants* (OM Code). Section 5.2, "Regulatory Guides," in NEDC-33911 does not discuss NRC Regulatory Guide (RG) 1.84, RG 1.147, or RG 1.192, as they relate to the acceptability of Code Cases for the ASME BPV Code and ASME OM Code for design, inservice inspection, and IST activities in satisfying 10 CFR 50.55a, respectively. The NRC staff requests GEH to clarify the intent of the topical report regarding these RGs.

GEH Response to NRC Question 03.09.06-18

NEDC-33911P will be revised to describe conformance to the regulatory guidance of RG 1.84, RG 1.147 and RG 1.192. The BWRX-300 will conform to the guidance and regulatory positions of RG 1.84, Revision 38, and using the guidance conformance described in the ESBWR DCD Tier 2, 26A6642AD, Table 1.9-21 to RG 1.84, Rev. 33. ASME BPV Code Case N-782 is also applied to the BWRX-300. Code Case N-782 endorses the use of the Edition and Addenda of ASME Boiler and Pressure Vessel 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 this Code case will be provided in the BWRX-300 Preliminary Safety Analysis Report (PSAR) or future licensing activities. Regulatory Guide 1.147, Rev. 19, the requirements of ASME B&PV Code Section XI, Division 1, specifically apply during construction and operations activities of nuclear power plants for performance of inservice inspection activities, and RG 1.192, Rev. 3, the requirements of ASME OM Code, specifically apply during operations and maintenance activities of nuclear power plants for performance of inservice testing activities. However, GEH design process and associated administrative controls considers operating plant compliance to RG 1.147 and RG 1.192 guidance in performing examinations, inspections and tests of installed systems and components and are incorporated in the design review process to support plant operation and maintenance best practices. Therefore, the guidance of RG 1.184, RG 1.147 and RG 1.192 will be considered in the BWRX-300 design phase in meeting the requirements of 10 CFR 50.55a.

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 0, will be revised to address the addition of Regulatory Guides 1.84, 1.147 and 1.192 as they relate to the acceptability of ASME Code Cases:

5.2.3 Regulatory Guide 1.84

RG 1.84. Design, Fabrication and Materials Code Case Acceptability, ASME Section III, Rev. 38, describes the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section III, "Rules for Construction of Nuclear Power Plant Components" Code Cases that the U.S. NRC has approved for use as voluntary alternatives to the mandatory ASME BPV Code provisions that are incorporated by reference into Title 10 of the Code of Federal Regulations (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities." This RG applies to reactor licensees subject to 10 CFR Part 50, Section 50.55a, "Codes and standards".

The BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.84, Rev. 38, and using the guidance conformance to RG 1.84, Rev. 33 described in ESBWR 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. Code Case N-782 endorses the use of the Edition and Addenda of ASME Boiler and Pressure Vessel 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 this Code case will be provided in the BWRX-300 Preliminary Safety Analysis Report (PSAR) or future licensing activities.

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7.0 5.2.~~3~~4 Regulatory Guide 1.141

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8.0 5.2.5 Regulatory Guide 1.147

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

Section 4.1.3 of LTR NEDC-33910P, Revision 0, Supplement 1, 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 B&PV Code Section XI, Division 1, specifically apply during construction and operation activities of nuclear power plants for performance of inservice inspection activities. The GEH design process and associated administrative controls considers operating plant compliance to RG 1.147 guidance in performing examinations, inspections and tests of installed systems and components, and are 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 B&PV Section XI Code Cases endorsed in RG 1.147 where necessary, will 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.

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9.0 5.2.~~4~~6 Regulatory Guide 1.155

. . .

10.0 5.2.~~5~~7 Regulatory Guide 1.163

. . .

11.0 5.2.8 Regulatory Guide 1.192

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

Section 4.1.3 of LTR NEDC-33910P, Revision 1, 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 IST activities. GEH design process and associated administrative controls considers operating plant compliance to RG 1.192 guidance in performing examinations, inspections and tests of installed systems and components, and are 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, will be demonstrated during future licensing activities.

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.

eRAI No.: 9745
Date of eRAI Issue: 06/22/20

NRC Question 03.09.06-19

SER Section 3.9.6: NRC Standard Review Plan (SRP) Section 3.9.6 specifies that the NRC staff will review the functional design, qualification, and inservice testing (IST) program for pumps, valves, and dynamic restraints to perform their design-basis safety functions. The NRC regulations in 10 CFR Part 50, Appendices A and B, and 10 CFR 50.55a include requirements for the capability of pumps, valves, and dynamic restraints to perform their safety functions. Section 5.3, "NUREG-0800 Standard Review Plan Guidance," in NEDC-33911 does not address the plans to satisfy SRP Section 3.9.6 for the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints, to provide assurance of their design-basis capability. The NRC staff requests GEH to clarify the intent of the topical report regarding SRP Section 3.9.6.

GEH Response to NRC Question 03.09.06-19

NEDC-33911P did not describe conformance to the regulatory guidance of SRP 3.9.6 and will be revised to include this information. The BWRX-300 containment isolation valves, which includes the RPV isolation valves, are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, including any application for a construction permit under 10 CFR 50 or design certification application under 10 CFR 52. These requirements are to be implemented during detailed design of the safety-related components. Therefore, the conclusion of this additional information to be provided is that the existing SRP provides adequate guidance to use during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 0, will be revised to add the following new Subsection 5.3.1:

. . .

5.3.1 Standard Review Plan 3.9.6

Standard Review Plan (SRP) 3.9.6, Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints, Rev. 4, states that the areas of review include the functional design and qualification provisions and IST programs for safety-related pumps, valves, and dynamic restraints (snubbers) designated as Class 1, 2, or 3 under ASME B&PV Code Section III. The review includes other pumps, valves and dynamic restraints not categorized as ASME BPV Code Class 1. 2 or 3 that have safety-related function. Conformance with the specific guidance in Subsection II of this SRP section will provide reasonable assurance that the functional design and qualification of pumps, valves and dynamic restraints within the scope of this SRP section and their associated IST programs satisfy the applicable requirements of Section 50.55a, “Codes and Standards,” of Title 10 of the *Code of Federal Regulations*, particularly the IST program requirements of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) 4; General Design Criterion (GDC) 1, "Quality Standards and Records," GDC 2, "Design Bases for Protection against Natural Phenomena," GDC 4, "Environmental and Dynamic Effects Design Bases," GDC 14, "Reactor Coolant Pressure Boundary," GDC 15, "Reactor Coolant System Design," GDC 37, "Testing of Emergency Core Cooling System," GDC 40, "Testing of Containment Heat Removal System," GDC 43, "Testing of Containment Atmosphere Cleanup Systems," GDC 46, "Testing of Cooling Water System," and GDC 54, "Systems Penetrating Containment," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities;" Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50; 10 CFR 52.47(b)(1), 10 CFR 52.79(a)(11), and 10 CFR 52.80(a).

The containment isolation valves are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, including any application for a construction permit under 10 CFR 50 or design certification application under 10 CFR 52. The requirements of 10 CFR 50.55a, are to be implemented during detailed design of the safety-related components of containment isolation. Therefore, the existing SRP provides adequate guidance to use during future review of a BWRX-300 10 CFR 52 design certification application if pursued
(as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

eRAI No.: 9745
Date of eRAI Issue: 06/22/20

NRC Question 03.09.06-20

SER Section 3.9.6: NRC Standard Review Plan (SRP) Section 3.9.6 specifies that the NRC staff will review the functional design, qualification, and inservice testing (IST) program for pumps, valves, and dynamic restraints to perform their design-basis safety functions. The NRC regulations in 10 CFR Part 50, Appendices A and B, and 10 CFR 50.55a include requirements for the capability of pumps, valves, and dynamic restraints to perform their safety functions. Section 5.4, "Generic Issues," and Section 5.5, "Operational Experience and Generic Communications," in NEDC-33911 only discusses two items with respect to these topics. As part of its SRP Section 3.9.6 review, the NRC staff evaluates the consideration of safety significant aspects of generic issues, operational experience, and generic communications for the functional design, qualification, and IST programs for pumps, valves, and dynamic restraints to provide assurance that those components are capable of performing their design-basis functions. The NRC staff requests that GEH clarify that an up-to-date evaluation of generic issues, and operational experience and generic communications, will be provided during future licensing activities under 10 CFR Part 50 or Part 52.

GEH Response to NRC Question 03.09.06-20

Sections 5.4 and 5.5 of NEDC-33911P do not represent the total listing required to support a 10 CFR 52 design certification application if pursued or for future 10 CFR 50 license applications and are provided based upon their relevance to the scope of this LTR. Therefore, NEDC-33911P will be revised to include a discussion in each section identifying the limited scope of this evaluation, and committing to the up-to-date evaluation of these issues to be provided during future licensing activities in support of a BWRX-300 10 CFR 52 design certification application or for future 10 CFR 50 license applications.

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 0, will be revised to include the following changes to Section 5.4 and Section 5.5:

. . .

5.4 Generic Issues

The following generic issues ~~do not represent the total listing required to support a 10 CFR 52 DCA if pursued or for future 10 CFR 50 license applications but~~ are provided based on their relevance to the scope of this LTR., and an up-to-date evaluation of generic issues is to be provided during future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant requesting a CP and OL under 10 CFR 50 or COL under 10 CFR 52.

5.5

The following operational experience and generic communications ~~do not represent the total listing~~ ~~required to support a 10 CFR 52 DCA if pursued or for future 10 CFR 50 license applications but~~ ~~are~~ provided based on their relevance to the scope of this LTR~~.~~, and an up-to-date evaluation of operational experience and generic communications is to be provided during future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant for requesting a CP and OL under 10 CFR 50 or COL under 10 CFR 52.

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eRAI No.: 9746
Date of eRAI Issue: 07/10/20

NRC Question 03.06.02-4

SER Section 3.6.2: Section 2.2.2, "Containment Design Requirements," in NEDC-33911 states that ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE-1120 and the design criteria from NRC Branch Technical Position (BTP) 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Part B, Items 1(ii)(1)(d) and (e), are applied to eliminate postulating breaks and cracks in those portions of piping from the containment wall to including the outboard containment isolation valves (CIVs). Similar statements are also included in Section 5.1.7, "10 CFR Part 50, Appendix A, GDC 4," in NEDC-33911. Eliminating postulated breaks and cracks in those portions of piping from the containment wall is safety significant to provide assurance that the containment of the BWRX-300 reactor will not be breached and cause a radioactive release to the environment that exceeds regulatory requirements.

a. The NRC staff requests that GEH clarify that BTP 3-4, Part B, Items 1(ii)(2) through (7), if applicable, are also applied to eliminate postulating breaks and cracks in those portions of the piping.

b. The NRC staff requests GEH describe how the BWRX-300 design requirements will provide assurance that the functionality of those outboard CIVs will not be affected by the dynamic effects resulting from postulated pipe breaks beyond those portions of piping from the containment wall to including the outboard CIVs.

GEH Response to NRC Question 03.06.02-4

a. The BWRX-300 will meet BTP 3-4 guidance similarly to the Economically Simplified Boiling Water Reactor (ESBWR) described in Design Control Document (DCD) 26A6642A5, Revision 10, Sections 3.6.1.1 and 3.6.2. In addition to ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-1120 and the design criteria from Branch Technical Position (BTP) 3-4, Items 1(ii)(1)(d) and (e), BTP 3-4, 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 including the outboard containment isolation valves (CIVs).

b. BTP 3-4, B.1.(ii) Fluid System Piping in Containment Penetration Areas states that breaks and cracks need not be postulated in those portions of piping from containment wall to and including the inboard or outboard isolation valves provided that they meet the design criteria of ASME Code, Section III, Subarticle NE-1120 and additional design criteria (1) through (7). As stated in LTR NEDC-33911P, Section 2.2.2, Design Requirements, the BWRX-300 design applies the ASME Code Section III, Subarticle NE-1120 design criteria as well as BTP 3-4 Items 1(ii)(1)(d) and (e) for ASME Code, Section III, Class 2 piping. In response to item a above, BTP 3-4, Items 1(ii)(2) through (7) design criteria will also be applied to those portions of piping from containment wall to including the outboard CIV. As a result of applying the design criteria of ASME Code, Section III, Subarticle NE-1120 and BTP 3-4, Part B, Items 1(ii)(1)(d) and (e), and Items 1(ii)(2) through (7), the dynamic effects of postulated breaks and cracks in those portions of the piping from the containment wall to the outboard CIV are not considered in the BWRX-300 design. No other additional changes to LTR NEDC-33911P are warranted as a result of adding BTP 3-4, Part B Items 1(ii)(2) through (7).

Proposed Changes to NEDC-33911P, Revision 0

a. NEDC-33911P, Revision 0, will be revised to reflect the revision of bullet five to Subsection 2.2.2, Design Requirements, and Section 5.1.7, 10 CFR 50 Appendix A, GDC 4, Statement of Compliance, to add compliance to the guidance of BTP 3-4, Part B, Items 1(ii)(2) through (7) that eliminates postulated breaks and cracks in piping:

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2.2.2 Containment Design Requirements

. . .

Design Requirements:

. . .

- <u>ASME B&PV Code, Section III, Division 1, Subarticle NE-1120, and the design criteria from BTP 3-4, Part B, Items 1(ii)(1)(d) and (e), and Items 1(ii)(2) through (7), are applied to eliminate postulated breaks and cracks in those portions of piping from the containment wall to the outboard CIVs.</u>

5.1.7 10 CFR 50 Appendix A, GDC 4

. . .

Statement of Compliance: . . . As described in this LTR, the BWRX-300 design requirements include applying the design criteria from NUREG-0800, SRP, BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Part B, Items 1(ii)(1)(d) and (e), <u>and Items 1(ii)(2) through (7)</u> to eliminate postulating breaks and cracks in those portions of piping from containment wall to and including the outboard CIVs.

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eRAI No.: 9746
Date of eRAI Issue: 07/10/20

NRC Question 03.06.02-5

SER Section 3.6.2: Section 3.1, "Scope of the Evaluation Model," in NEDC-33911 states that 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 to provide assurance that a breach in the containment of the BWRX-300 reactor will not occur and cause a radioactive release to the environment that exceeds regulatory requirements. The NRC staff requests that GEH clarify that the BWRX-300 containment response to all of the dynamic effects (not only the jet loads) 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 10 CFR Part 50, Appendix A, GDC 4 requirements.

GEH Response to NRC Question 03.06.02-5

The BWRX-300 design will consider the dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids in the containment design and associated piping, valves, penetrations, and instrument lines in future licensing activities. NEDC-33911P, Revision 0, Section 3.1 and Section 5.1.7 will be revised to include the dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids in the BWRX-300 design of containment and CIV design features.

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 0, Sections 3.1 and 5.1.7 will be revised to reflect the dynamic effects of pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids to comply with the design requirements of 10 CFR 50. Appendix A, GDC 4:

3.1 Scope of the Evaluation Model

. . .

<u>The dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids will be evaluated in design of the containment and CIVs, and described during future licensing activities to comply with the design requirements of 10 CFR 50, Appendix A, GDC 4.</u>

5.1.7 10 CFR 50 Appendix A, GDC 4

. . .

Statement of Compliance: The BWRX-300 containment and CIVs design features . . . are to be designed to effects of, and to-be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents, and will consider the dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids. . . .

eRAI No.: 9746
Date of eRAI Issue: 07/10/2020

NRC Question 03.06.02-6

SER Section 3.6.2: Section 5.1.7 in NEDC-33911 states that breaks and cracks in those portions of piping from the RPV isolation valves that function as the inboard CIVs to the containment wall remain postulated to occur, and the dynamic effects of those postulated pipe breaks are to be evaluated in the BWRX-300 design. The NRC staff notes that during its review of NEDC-33910, in RAI Question 03.06.02-2, the NRC staff requested that GEH describe how the BWRX-300 design requirements will provide assurance that the functionality of those dual function safety-related valves will not be affected by the dynamic effects resulting from the postulated pipe breaks. In a letter dated May 4, 2020, GEH responded to the staff's request by referencing Section 5.1.7 in NEDC-33911, and stated that the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 4. In addition, GEH stated that qualification, such as compliance with ASME Standard QME-1-2007 (or a later edition) as accepted in NRC Regulatory Guide 1.100, will be addressed in the detailed design of the valves and will be specified during future licensing activities. The capability of CIVs to perform their design-basis functions is safety significant to provide assurance that the containment of the BWRX-300 reactor can be safely isolated and prevent radioactive release to the environment that exceeds regulatory requirements. The NRC staff requests that GEH update Section 5.1.7 in NEDC-33911 to describe the valve qualification in compliance with ASME Standard QME-1-2007 as described above.

GEH Response to NRC Question 03.06.02-6

BWRX-300 containment isolation valve qualification, using ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100, will be addressed in the detailed design of the valves and will be specified during future licensing activities. NEDC-33911P, Revision 0, Sections 2.2.7 and 5.1.7 will be revised to reflect the use of the ASME Standard QME-1-2007 (or later edition).

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 0, Sections 2.2.7 and 5.1.7 will be revised to add valve qualification compliance to ASME Standard QME-1-2007 or later edition:

2.2.7 Containment Isolation Valves

Design Requirements:

. . .

- <u>Valve qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as endorsed by NRC Regulatory Guide (RG) 1.100, will be addressed in the detailed design and the procurement process of the valves, and will be specified during future licensing activities.</u>

5.1.7 10 CFR 50 Appendix A, GDC 4

. . .

Statement of Compliance: . . . Internal containment flooding is to be evaluated during future licensing activities. Valve qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as endorsed by NRC Regulatory Guide (RG) 1.100, will be addressed in the detailed design and the procurement process of the valves, and will be specified during future licensing activities.

eRAI No.: 9760
Date of eRAI Issue: 07/21/20

NRC Question 06.02.05-1

10 CFR 50.44 provides requirements for the mitigation of combustible gas generated by a beyond-design-basis accident. In light-water reactors, the principal combustible gas is hydrogen. Hydrogen monitors are used, in conjunction with oxygen monitors in inerted containments, to guide response by reactor operators through emergency operating procedures. 10 CFR Section 50.44 requires that equipment be provided for monitoring hydrogen within the containment structure or vessel. The equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a beyond-design-basis accident. Mitigation of combustible gas is safety significant to prevent detonation and provide assurance that the containment of the BWRX-300 reactor will not be breached and cause a radioactive release to the environment that exceeds regulatory limits.
Licensing Topical Report NED-33911P, "BWRX-300 Containment Performance," Section 5.1.2, indicates in the statement of compliance for regulation 10 CFR 50.44 (c)(4), that the BWRX-300 design will include the use of oxygen analyzers for monitoring oxygen concentrations. The requirement for hydrogen monitoring is also identified in 10 CFR 50.44 (c)(4)(ii) but is not addressed in the LTR.
The NRC staff requests that GEH explain how the BWRX-300 design will comply with 10 CFR 50.44 (c)(4)(ii).

GEH Response to NRC Question 06.02.05-1

During severe accident conditions with a significant amount of fission product gases and hydrogen release to the containment, the containment remains inerted without any additional action because radiolytic oxygen production remains below the concentration that could pose a risk of hydrogen burning for a significant period of time following the event. Accumulation of combustible gases that may develop in the period after 72 hours can be managed by implementation of the severe accident management guidelines. Compliance to 10 CFR 50.44 for the beyond-design-basis accident (BDBA) will be addressed in the forthcoming LTR NEDC-33921P BWRX-300 Severe Accident Management.

The BWRX-300 inerted containment design will include reliable oxygen and hydrogen analyzers for monitoring oxygen and hydrogen concentrations in containment to comply with the requirements of 10 CFR 50.44(c)(4). NEDC-33911P, Revision 0, Section 5.1.2 will be revised to reflect both oxygen and hydrogen analyzers used to monitor these gas concentrations in containment.

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 1, Section 5.1.2 will be revised to reflect that both oxygen and hydrogen concentrations in containment will be monitored using reliable equipment:

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5.1.2 10 CFR 50.44

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Regulatory Requirement: 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).

Statement of Compliance: 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).

eRAI No.: 9760
Date of eRAI Issue: 07/21/20

NRC Question 06.02.05-2

10 CFR 50.44, "Combustible gas control for nuclear power reactors" subpart (c), requires in part that all containments must have the capability to ensure a mixed atmosphere during design-basis and significant beyond design basis accidents. A mixed atmosphere means that the concentration of combustible gases in any part of the containment is below a level that supports combustion or detonation that could cause loss of containment integrity. An analysis of the effectiveness of the method used for providing a mixed atmosphere should demonstrate that combustible gases, generated by a beyond-design-basis accident, will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause loss of containment integrity. Ensuring that a mixed atmosphere is established and maintained in the containment is safety significant since it provides protection against a hydrogen detonation that could compromise the containment and thus cause a large radioactive release to the environment that exceeds regulatory limits.

LTR Section 5.1.2 indicates in the statement of compliance with 10 CFR 50.44 (c)(4), that with an inerted containment, oxygen concentrations reaching flammable mixture levels in sub-compartments become a concern even if the average concentration is below the limit. The only sub-compartment that may experience this phenomenon is the containment head section above the refueling bellows. However, natural circulation from the presence of the passive containment cooling and the very low oxygen concentrations in the main section of the containment prevent significant oxygen accumulation above the refueling bellows.

The NRC staff requests that GEH explain how the BWRX-300 design precludes significant oxygen accumulation above the refueling bellows through natural circulation from passive containment cooling. The NRC staff also requests that GEH to identify the location where monitoring of combustible gases will occur and describe how natural circulation flow provides sufficient mixing.

GEH Response to NRC Question 06.02.05-2

The oxygen generated from radiolysis is much less in the BWRX-300 than conventional boiling water reactors (BWRs) and the Economically Simplified Boiling Water Reactor (ESBWR) due to the lower power output of the BWRX-300. The dome region in the BWRX-300 is similar to the ESBWR. Since the dome region in the BWRX-300 is much smaller than the containment volume, the majority of the oxygen that is mixed in steam remains in containment and never migrates into the dome region following a design basis accident. In the long term, over several days, there is sufficient air exchange between the containment and the dome region through two manholes on the refueling bellows that results in limiting the oxygen concentration in the dome region. The amount of oxygen that may accumulate in the dome region will be determined for the most limiting design basis accidents using bounding assumptions that will be described in NEDC-33922P, BWRX-300 Containment Evaluation Method. If the oxygen concentration cannot be demonstrated to remain below the deflagration limits conclusively, the dome region will be equipped with a hydrogen removal device such as a passive autocatalytic recombiner device or surface coating similarly used on the ESBWR.

Once the most limiting design basis accident with bounding assumptions is evaluated, the location for reliable combustible gas monitoring equipment will be identified during future licensing activities or in the Preliminary Safety Analysis Report (PSAR).

Proposed Changes to NEDC-33911P, Revision 0

None

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eRAI No.: 9764
Date of eRAI Issue: 07/17/20

NRC Question 06.02.01-1

Maximum Expected External Pressure

The applicant elected to use SRP 6.2.1.1.A, “PWR Dry Containments, Including Subatmospheric Containments, Rev. 3,” as the guidance and acceptance criteria 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 General Design Criteria (GDCs) 38 and 50 of Appendix A to 10 CFR Part 50 with respect to the functional capability of the containment heat removal systems and containment structure under loss-of-coolant accident conditions, provisions would be needed to protect the containment structure against possible damage from external pressure conditions. The provisions should include conservative structural design to assure that the containment structure is capable of withstanding the maximum expected external pressure; or interlocks in the plant protection system and administrative controls preclude inadvertent operation of the containment heat removal systems. If the containment is designed to withstand the maximum expected external pressure, the external design pressure of the containment should provide an adequate margin above the maximum expected external pressure to account for uncertainties in the analysis of the postulated event. However, the applicant did not provide any information regarding the demonstration methodology/evaluation model and safety margin to satisfy the external pressure acceptance criterion in its Topical Report (TR), “BWRX-300 Containment Performance” (NEDC-33911P, Revision 0). Therefore, the staff requests the applicant to discuss how its design meets GDCs 38 and 50 to assure containment functional capability of the BWRX-300 containment, and to update its TR with the necessary information demonstrating that the design meets the external pressure criterion.

GEH Response to NRC Question 06.02.01-1

The BWRX-300 containment structural 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. A negative design pressure limit is established as part of the containment structural design. However, a significant negative pressure does not develop during design basis accidents. The non-condensable gases that exist initially in the containment remain in the containment following a design basis accident except for a small amount of leakage. It is not possible to condense enough steam to create a significant negative pressure during design basis accidents in the BWRX-300.

In beyond design basis accidents, the potential for negative pressure is mitigated by using the nitrogen supply system to limit negative pressure. Beyond design basis accidents will be discussed in the future submittal of licensing topical report NEDC-33921P, BWRX-300 Severe Accident Management.

The BWRX-300 containment structural design evaluation to withstand the maximum expected external pressure to demonstrate compliance to 10 CFR 50, Appendix A, GDC 38 and GDC 50 will be provided during future licensing activities. NEDC-33911P, Revision 0, Section 5.1.18 10 CFR 50 Appendix A, GDC 50 will be revised to reflect the future submittal of the maximum external containment pressure structural design evaluation with sufficient margin to account for uncertainties from a full spectrum of postulated accidents.

Proposed Changes to NEDC-33911P, Revision 0

5.1.17 10 CFR 50 Appendix A, GDC 50

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Statement of Compliance: . . .The GOTHIC computer methodology for measuring containment response is provided in LTR NEDC-33922P BWRX-300 Containment Evaluation Method [Reference 6.5]. The analyses to demonstrate compliance will be provided during future licensing activities. The BWRX-300 containment structural 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. The maximum expected external pressure containment structural evaluation will demonstrate compliance to 10 CFR 50, Appendix A, GDC 38 and 50 and be provided in future licensing activities.

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eRAI No.: 9765
Date of eRAI Issue: 07/17/20

NRC Question 06.02.01-2

Containment Subcompartments

In GEH Licensing Topical Report (TR), "BWRX-300 Containment Performance (NEDC-33911P, Revision 0)," the applicant 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." The report states that 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. The applicant concluded that subcompartments are used in the model only to the extent to calculate containment atmosphere mixing, and that the acceptance criteria associated with SRP Section 6.2.1.2, "Subcompartment Analysis," are met without the need for specific analyses for the BWRX-300 design.

The staff notes that the BWRX-300 containment design is at a conceptual stage and considers the volume below the RPV, the space between RPV and the biological shield, and the containment head area above the refueling bellows, as sub-compartments. For the staff to conclude that subcompartment pressurization does not apply to the BWRX-300 containment, the applicant would need to justify that requirements in General Design Criteria (GDC) 4 and GDC 50 are met for the actual subcompartments functional design by showing that no significant pressure differentials are created across the subcompartment walls under the postulated DBAs caused by line breaks both inside or outside the subcompartments. Therefore, the staff requests the applicant to discuss how the BWRX-300 containment subcompartments meet the regulations and provide a statement of compliance and the related design methodology development in the TR.

GEH Response to NRC Question 06.02.01-2

The containment shell, as well as internal structures and components are evaluated for the dynamic effects of jet impingement, missiles, postulated high energy breaks, discharging fluids, and pipe whip as part of the detailed design in accordance with 10 CFR 50 Appendix A, GDC 4 and GDC 50. In response to NRC Question 9746, Question 03.03.02-2, dated June 30, 2020, GEH committed to revising Section 3.1 and Section 5.1.17 of NEDC-33911P to reflect that all of these dynamic effects are evaluated for compliance to GDC 4 and GDC 50, accordingly.

Since the BWRX-300 subcompartments do not contain large bore high energy line breaks, they are not subject to subcompartment pressurization, and significant acoustic waves do not develop in the subcompartments creating dynamic loads. As a result, the boundaries of the subcompartments do not experience pressure differentials that are higher than the pressure differentials exerted on the containment boundaries. The subcompartment boundaries are designed with sufficiently large openings preventing a significant pressure differential to develop between the subcompartments and the rest of the containment as a result of a pipe break outside the subcompartment.

Forthcoming Licensing Topical Report NEDC-33922P BWRX-300 Containment Evaluation Method will provide the detailed demonstration case results that show no significant pressure differential across subcompartment walls due to large breaks outside the subcompartments. The requirements of GDC 4 and GDC 50 are met.

Proposed Changes to NEDC-33911P, Revision 0

None

eRAI No.: 9766
Date of eRAI Issue: 07/17/2020

NRC Question 06.02.01-3

Passive Containment Cooling System (PCCS) Design

To meet the General Design Criteria (GDCs) 16, 38, and 50 of Appendix A to 10 CFR Part 50 relevant to the containment design basis and guided by SRP Sections 6.2.1.1.A and 6.2.1.3, the staff reviewed the BWRX-300 analytical models and assumptions used in the containment thermal-hydraulics response analysis methodology to determine if the licensing-basis safety analyses are acceptably conservative. Specifically, the staff reviewed the containment design pressure to determine whether it bounds the peak accident containment pressure resulting from the most limiting DBE with sufficient margin to conform to SRP 6.2.1.1.A acceptance criteria.

In GEH Licensing Topical Report, "BWRX-300 Containment Performance (NEDC-33911P, Revision 0)," Table 3-2, "Phenomena Identification and Ranking Table for Containment (Excluding RPV)," the applicant provided a summary of the GOTHIC Phenomenon Identification and Ranking Table (PIRT) for Containment (Excluding RPV). For [[

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GEH Response to NRC Question 06.02.01-3

These phenomena and the remainder of the PIRT Table 3-2 phenomena will be discussed in detail in LTR NEDC-33922P, along with the sensitivity and demonstration cases, providing additional support for the rankings identified in PIRT Table 3-2.

Boiling heat transfer is included in PIRT [[ ]] mainly for the potential of long-term boiling. Immediately following a large break LOCA, when the PCCS is exposed to a mixture of mostly steam with very little non-condensables, the condensation heat transfer coefficient on the outer surface of the PCCS may become large. However, this period is not long enough for heat up to occur on the PCCS metal and the PCCS working fluid to bring the fluid near the boiling temperature. Boiling may occur only in the long-term resulting in the high ranking. In the short-term, there is no boiling. Therefore, it is ranked low.

Similarly, in PIRT [[ ]], convection inside the pipe is not important in the short-term for large breaks for the same reasons discussed above for boiling. Convection may have some importance for small breaks in the short-term, although it is not the dominant resistance because overall PCCS heat removal rate is limited by the heat transfer on the outer surface of the PCCS due to the high concentration of non-condensables. This results in PIRT Item 3 being ranked as medium importance. Sensitivity studies performed by changing the PCCS loss coefficient will be addressed in the forthcoming Licensing Topical Report (LTR) NEDC-33922P Containment Evaluation Method that will support this ranking.

LTR NEDC-33911P, Sections 3.4.2.2, 3.4.2.3, 3.4.2.4, and Tables 3-1 Phenomena Ranking Criteria and 3-2 Phenomena Identification and Ranking Table for Containment (Excluding RPV) are moved to LTR NEDC-33922P Containment Evaluation Method for review with the TRACG and GOTHIC methods where detailed evaluation and analysis of the BWRX-300 containment response are provided.

Proposed Changes to NEDC-33911P, Revision 0

LTR NEDC-33911P, Sections 3.4.2.2 GOTHIC Phenomenon Identification and Ranking Table (PIRT), 3.4.2.3 PIRT Survey, 3.4.2.4 Development of the Assessment Base and Tables 3-1 Phenomena Ranking Criteria and 3-2 Phenomena Identification and Ranking Table for Containment (Excluding RPV) are moved to LTR NEDC-33922P Containment Evaluation Method for review with the TRACG and GOTHIC methods. There is no content change for this move.

eRAI No.: 9767
Date of eRAI Issue: 07/17/20

NRC Question 06.02.01-4

Addressing 10 CFR Part 50, Appendix K

In GEH Licensing Topical Report (TR), “BWRX-300 Containment Performance (NEDC-33911P, Revision 0),” the applicant mentioned the regulations in 10 CFR Part 50, Appendix K, “ECCS Evaluation Models,” once in Section 5.3.1, “Standard Review Plan 6.2.1,” by 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 staff noted that the TR did not include a compliance statement for 10 CFR Part 50, Appendix K, as it did for GDC 4, GDC 16, and GDC 50. Therefore, the staff requests the applicant to discuss how it plans to address 10 CFR Part 50, Appendix K, as it relates to sources of energy during the loss-of-coolant accident, and to assure that all the energy sources have been considered in its design.

GEH Response to NRC Question 06.02.01-4

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 50, Appendix K, using the assumptions and correlations similar to those that were used in Licensing Topical Report (LTR) NEDC-33083P-A, “TRACG Application for ESBWR.” These applicable energy sources, the correlations and the conservative biases will be described in detail in forthcoming LTR NEDC-33922P, Containment Evaluation Method.

10 CFR 50, Appendix K also specifies analysis requirements for 10 CFR 50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors compliance. Compliance to 10 CFR 50.46 was provided in LTR NEDC-33910P, Revision 0, Supplement 1, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection, Section 4.1.2.

The BWRX 300 evaluation model will not use the evaluation models and correlations provided in 10 CFR 50 Appendix K; however, the BWRX-300 ECCS evaluation model will address the required applicable features of 10 CFR 50, Appendix K in response to a LOCA.

Proposed Changes to NEDC-33911P, Revision 0

None

eRAI No.: 9758
Date of eRAI Issue: 07/21/20

NRC Question 06.02.04-1

GDC 55, which governs containment isolation valves (CIVs) for lines penetrating the primary containment boundary as parts of the reactor coolant pressure boundary, requires two containment isolation valves: one inside containment and one outside containment. SRP Section 6.2.4 provides guidance that containment isolation valve closure times should be selected for rapid isolation of the containment following postulated accidents to minimize the release of containment atmosphere to the environs.

NEDC-33911 Section 2.2.7.1 indicates that the BWRX-300 lines penetrating the primary containment boundary are part of the reactor coolant pressure boundary, and that RPV isolation valves are used for inboard containment isolation and automatic CIVs outside containment are designed for GDC 55 compliance. However, GEH 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. The basis for this statement is not clear. Additionally, it is not clear what scenarios and assumptions have been considered in the GEH analysis, and what analysis results can be used to demonstrate that the outboard CIV valves are not required for fast closing.

The NRC staff requests GEH to explain the basis of outboard CIV valve closure time and provide the time scale for the outboard CIV closure for each scenario being considered.

GEH Response to NRC Question 06.02.04-1

As discussed in NEDE-33910P, Revision 0, Supplement 1, the BWRX-300 reactor incorporates the use of [[

]]. The RPV isolation valves for main steam, feedwater, shutdown cooling, and reactor water cleanup fail in the closed position and automatically close on high containment pressure indicating a loss-of-coolant accident (LOCA). [[

]]. NEDC-33910P, Section 2.7 discusses the BWRX-300 steam and liquid pipe breaks evaluated and divides the safety analysis into two size categories. The largest steam line break is a main steam line break. The largest liquid line break is the feedwater line break. [[

]]. These small lines meet the isolation requirements delineated in RG 1.11 for instrument lines penetrating containment that are connected to the primary coolant system.

As a result of the fast closing RPV isolation valves in conjunction with the large capacity of the isolation condenser system, there are no credible design basis accidents (DBAs) where postulated fission product releases greater than what is contained in normal reactor coolant could occur for small, medium or large break LOCAs. As stated in Section 2.2.7 of NEDC-33911P, Design Requirements, CIV closure timing requirements are commensurate with the timing of the potential for fission product releases. This supports the statement in Section 2.2.7.1 of NEDC-33911P. The outside containment automatic CIVs closure time will be established to assure containment isolation prior to the first fission product release greater than what is contained in normal reactor coolant established by source term evaluations. However, Section 2.2.7.1 will be revised to clarify that there is no credible fission product release scenario greater than what is contained in normal reactor coolant for a DBA, and that the outside containment isolation valves closure timing will be based upon the fission product release timing.

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 0, Section 2.2.7.1 Containment Isolation Valves Connected to RPV Boundary will be revised to reflect that containment isolation valves outside containment closure times will be based upon the most limiting Beyond Design Basis Accident (BDBA) or severe accident (SA).

2.2.7.1 Containment Isolation Valves Connected to RPV Boundary

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[[

]]. The outside containment automatic CIVs closure time will be established to assure containment isolation prior to the first fission product release greater than what is contained in normal reactor coolant in source term evaluations which will be completed in future licensing activities. These closure times are expected to be in the order of minutes. Additionally, the valve closing time for all CIVs will support specific break isolation functions balanced with water hammer and valve loading considerations. LTR NEDC-33921P, Severe Accident, will provide the evaluation for fission product releases resulting from BDBA or SA events and provide the necessary timing information to establish the closure times for the outside containment isolation valves.

eRAI No.: 9758
Date of eRAI Issue: 07/21/20

NRC Question 06.02.04-2

GDC 55, as to isolation valves for lines penetrating the primary containment boundary as part of the reactor coolant pressure boundary, requires two containment isolation valves, one inside containment and one outside containment.

NEDC-33911 Section 2.2.7.1 states that the isolation condenser system (ICS) reactor pressure vessel (RPV) isolation valves for the ICS steam supply and condensate return piping are inside containment automatic CIVs. The closed loop isolation condenser (IC) located outside the containment is used as a "passive" substitute for an open "active" outside containment automatic CIV.

Substituting the IC located outside the containment with an open "active" outside containment automatic CIV does not comply with the requirements of GDC 55. The NRC staff requests GEH to explain how the IC loop located outside the containment can be used as a substitute for an outboard containment automatic CIV to satisfy GDC 55 and provide the explanation in a supplement to the LTR.

GEH Response to NRC Question 06.02.04-2

<u>Additional Isolation Configuration for Other Defined Basis Provision – GDC 55</u>

The combined design features of the [[ ]] meet the definition of an ECCS as described in 10 CFR 50.46(a)(1)(i) and has a calculated cooling performance following postulated LOCAs in compliance with the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria set forth in 10 CFR 50.46(b). This system function necessitates treatment as "other defined basis" under 10 CFR 50, Appendix A, GDC 55, that justifies this containment isolation configuration. The ICS RPV isolation valves, which also function as containment isolation valves, will only close if a pipe break is detected on the specific ICS train where the leakage is detected. For all other LOCAs, [[ ]]. Containment atmosphere leakage to the environment is precluded because the ICS is a closed system and also operates at a higher pressure than containment. The ICS CIV location and actuation is in the direction of highest safety because the ECCS function is ensured for all LOCAs, and for an ICS pipe break, the isolation point is as close as possible to the energy source. This limits the loss of coolant from the reactor and due to the closed system design, eliminates this system as a containment leakage path. If an additional isolation valve were added to the ICS train outside of containment, it would not improve the leakage potential from containment, and it would introduce an unnecessary new potential failure mode of an ECCS.

Standard Review Plan 6.2.4, Containment Isolation System, allows two other configurations rather than the explicit configuration delineated in 10 CFR 50, Appendix A, General Design Criteria (GDC) 55 or 56: (1) one CIV and a closed system, both outside containment; or (2) two CIVs outside containment. The BWRX-300 ICS does not conform to either of these configurations. However, GDC 55 does allow the approval of additional isolation configurations under the "other defined basis" provision of GDC 55. The proposed changes to LTR NEDC-33911P provide justification to the proposed alternative in order to apply the "other defined basis" to ensure sufficient safety consistent with the overall containment isolation design philosophy expressed in GDC 55 and the guidance documents. For example, SRP Section 6.2.4 states: "If it is not practical to locate a valve inside containment (for example, the valve may be under water as a result of an accident), both valves may be located outside containment." As discussed above, these CIVs [[ ]]. Post-accident, the ICS operating pressure is equal to or greater than the containment pressure and the associated [[ ]]. The ICS is designed to provide high assurance of leak integrity during post-accident operation.

The guidance of Regulatory Guide (RG) 1.141, Containment Isolation Provisions for Fluid Systems, July 2020, Revision 1, C.1, allows for system integrity inspections applied to closed systems in lieu of leak testing, as stated in ANSI N271-1976 Section 3.6.4. Additionally, RG 1.141, C.8 states that compliance to Section 4.4.8 of ANSI N271-1976 allows for a branch line in a closed system to constitute one of the containment isolation barriers by meeting the design criteria of Section 3.5 or Section 3.6.7 of ANSI N2710-1976. Further RG 1.141, C.10, in accordance with Section 4.1.4 of ANSI N271-1976 states: "The piping between isolation barriers or piping which forms part of isolation barriers, shall meet the requirements of 3.7 and applicable requirements for isolation barriers." Piping between isolation barriers should meet the applicable requirements of Section 3.5 or Section 3.7 of ANSI N271-1976.

System Design Considerations

Given the other defined basis alternative containment configuration of GDC 55 and the guidance of RG 1.141 for closed system containment isolation provisions for fluid systems, an alternative arrangement is provided for the ICS where two in-series RPV isolation valves on each end of the system also function as containment isolation valves. A break in an ICS line either inside or outside containment could be isolated by either of the two redundant [[ ]]. The piping in the area between the outermost [[ ]] and containment boundaries, as well as the piping through the seismic Category I reactor building where the ICS steam supply and the ICS condensate return piping connect to the ICS heat exchanger inside the IC pool, are designed using ASME Section III, Class 1, NB piping, which limits the probability of breaks in these segments of the piping. Additionally, a break, between the isolation valves and the containment would be isolated by the RPV isolation valves to stop the leak and would still be contained by the closed system outside containment that is designed to withstand full reactor pressure. It would require an additional break before a radioactive release could occur, and even with an additional break, the coolant source remains isolated. Therefore, this design can accommodate a single failure.

The ICS steam supply lines for each train contain two valves in series inside containment, combined with a closed loop outside containment, providing sufficient containment isolation. The ICS steam supply lines, and condensate return lines passes through the seismic Category 1 reactor building in order to connect to the isolation condenser in the ICS pool. The ICS vent lines each contain two in-series containment isolation valves and are attached to the closed loop outside containment. The ICS condensate return line for each train has two valves in series to provide RPV isolation and containment isolation functions located in containment where they are protected from outside environmental conditions that may be caused by a failure outside containment. The condensate line along with the steam supply line is automatically isolated when leakage is detected in the specific ICS train.

Given the above rationale, the containment isolation provisions for the ICS condensate, steam, vent and purge lines constitute an appropriate application of the “other defined basis” alternative defined in 10 CFR 50, Appendix A, GDC 55 because a single failure would not disable the containment isolation function, while allowing the [[ ]] to remain open to allow the ICS to function as part of the ECCS.

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 0, Section 2.2.7.1 Containment Isolation Valves Connected to RPV Boundary, will be revised to reflect that the design requirement of a closed loop outside containment plus two in-series automatic isolation valves inside containment constitute meeting the other defined basis provision of GDC 55.

2.2.7.1 Containment Isolation Valves Connected to RPV Boundary

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For the ICS as shown on Figure 2-7, [[ ]] comply with the “other defined basis” alternative containment configuration requirements of GDC 55.

[[ ]]. An alternative arrangement is provided for the ICS where two in-series RPV isolation valves on each end of the system function as containment isolation valves. A break in an ICS line either inside or outside containment could be isolated by either of the two redundant [[ ]]. The piping in the area between the outermost [[ ]] and the containment boundaries, as well as the piping through the seismic Category I reactor building where the ICS steam supply and the ICS condensate return piping connect to the ICS heat exchanger located in the ICS pool, are designed using ASME Section III, Class 1, NB piping, which limits the probability of breaks in these segments of the piping. Additionally, a break, between the RPV isolation valves and the containment would be isolated by the RPV isolation valves to stop the leak and would be contained by the closed system outside containment that is designed to withstand full reactor pressure. It would require an additional break before a radioactive release could occur, and even with an additional break, the coolant source remains isolated. Therefore, this design can accommodate a single failure and maintain containment leak integrity.

The ICS steam supply lines for each train contain two in-series valves inside containment, combined with a closed loop outside containment, thus providing containment isolation. The ICS steam supply lines and condensate return lines pass through the seismic Category 1 reactor building in order to connect to the isolation condenser in the ICS pool. The ICS vent lines each contain two in-series containment isolation valves and are attached to the closed loop outside containment. The ICS condensate return line for each train has two valves in-series to provide RPV isolation and containment isolation functions and are located inside containment where they are protected from outside environmental conditions that may result from a failure outside containment. The ICS condensate return line along with the steam supply line is automatically isolated when leakage is detected in the specific ICS train.

Given the above rationale, the containment isolation provisions for the ICS condensate, steam, vent and purge lines constitute an appropriate application of the “other defined basis” alternative defined in 10 CFR 50, Appendix A, GDC 55. A single failure would not disable the containment isolation function, while allowing the [[ ]] to remain open to allow the ICS to function as a part of the ECCS.

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eRAI No.: 9758
Date of eRAI Issue: 07/21/20

NRC Question 06.02.04-3

NEDC-33911, Section 2.2.7.1 states that the containment penetrations for fine motion control rod drive (FMCRD) hydraulic lines (shown on Figure 2-8 of NEDC-33911) do not have isolation valves based on closed-system piping outside the primary containment vessel (PCV) and reactor coolant pressure vessel (RCPB) isolation valves (internal ball check valves) in the design of the drives.

The NRC staff requests GEH to explain how the design of the hydraulic lines penetrating containment with only one internal ball check valve and a closed system outside containment complies with the requirements of GDC 54 for containment isolation and GDC 55 for one inboard and one outboard containment isolation valve.

GEH Response to NRC Question 06.02.04-3

Additional Isolation Configuration for Other Defined Basis Provision – GDC 55

As stated in NEDC-33911P, Containment Performance, Section 2.2.7, the BWRX-300 CRD system is designed similarly to the Economically Simplified Boiling Water Reactor (ESBWR) (see ESBWR Design Control Document (DCD) Tier 2, Chapter 4, Reactor, 26A6642AP Rev. 10, Section 4.6). The control rod drive mechanism design incorporates a brake system and ball check valve that reduces the likelihood of rapid rod ejection. The ball check valve is classified as safety-related as it actuates closure of the scram inlet port by reverse flow under system pressure, and fluid flow conditions caused by a potential break in the scram line. This prevents the loss of pressure to the underside of the hollow piston and generation of loads on the drive that could cause a rapid control rod withdrawal. The ball check valve is only open when the system is pressurized. The design pressure for the CRD hydraulic system and associated ball check valve accommodates reactor pressure and is therefore well above containment post-accident conditions.

These FMCRDs and hydraulic control units (HCUs) have considerable commercial service in the Japanese Advanced Boiling Water Reactors (ABWRs) with over 22 reactor-years of service without design issues or anomalies identified. These units have been manufactured and tested for the Japanese ABWRs and Taiwanese projects as well and successfully passed performance testing requirements. The predecessor design to the FMCRD is the hydraulic locking piston Control Rod Drive. The locking piston hydraulic CRDs have years of proven experience in the operating BWR fleet. These hydraulic CRDs use the same internal ball check valve at the scram inlet line hydraulic port from the HCU.

The FMCRDs and HCUs shown on Figure 2-8 of NEDC-33911P are similar mechanically to the ESBWR with the exception that the [[

]].

The design provides for isolation capability that terminates high pressure CRD reverse flow in the event of depressurization due to a line break to ensure containment pressure remains within design limits.

As discussed in the Section 4.6.3 Staff Evaluation, NUREG-1966, Volume 2, Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design Volume 2 (Chapters 4-8), the staff's review of the functional design of the ESBWR CRD system confirmed that it satisfies the safety design bases and regulatory criteria in Section 4.6.1 of NUREG-1966. Further, the staff indicated that the single-failure analysis of the FMCRD and HCU components indicates that the system design is satisfactory.

The FMCRDs are mounted to the reactor vessel bottom head inside the primary containment. The HCUs are housed in four dedicated rooms directly in the reactor building adjacent to the primary containment. Each HCU room serves a set of FMCRDs associated with designated control rods of the reactor core. The HCUs are connected to the FMCRDs by the scram insert piping that penetrates primary containment. All materials for use in this system are selected for their compatibility with the reactor coolant as described in Articles NB-2160 and NB-3120 of the ASME Code. The only primary pressure boundary components are the lower housing of the spool piece assembly and the flange of the outer tube assembly. These components are made with 300 series stainless steel materials in accordance with the ASME Code, Section III. Some CRDs are removed each refueling outage and disassembled for routine inspection with drive parts, including the CRD bolting and hard-surfaced parts accessible for visual examination in accordance with manufacturer's CRD maintenance procedures. The inspection program is adequate to detect any defects or leaks before they become serious operating problems.

Standard Review Plan 6.2.4, Containment Isolation System, allows two other configurations rather than the explicit configuration delineated in 10 CFR 50, Appendix A, General Design Criteria (GDC) 55 or 56: (1) one CIV and a closed system, both outside containment; or (2) two CIVs outside containment. The BWRX-300 CRD does not explicitly conform to either of these configurations. However, GDC 55 does allow the approval of additional isolation configurations under the "other defined basis" provision of GDC 55. The proposed changes to LTR NEDC-33911P provide justification for the proposed alternative to ensure sufficient safety consistent with the overall containment isolation design philosophy expressed in GDC 55 and the guidance documents. This system function necessitates an "other defined basis" under 10 CFR 50, Appendix A, GDC 55, that justifies this containment isolation configuration for a closed system arrangement outside containment.

Additionally, SRP 6.2.4, SRP Acceptance Criteria, Item 5, states that containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems normally consist of two isolation valves in series. A single isolation valve is acceptable if the system reliability can be shown to be greater, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. Further, the closed system outside containment should be protected from missiles, designed to seismic Category I and Group B quality standards, have a design temperature and pressure rating at least equal to that for the containment. The closed system outside containment should be leak-tested unless system integrity can be shown to be maintained during normal operations. For this type of isolation valve arrangement, the valve is located outside containment, and the piping between the containment and the valve should be enclosed in leak-tight or controlled-leakage housing. In lieu of housing, the piping and valve are designed conservatively to preclude a breach of piping integrity, the design should comply with SRP Section 3.6.2 requirements. The design of the valve or the piping compartment should provide the capability to detect and terminate leakage from the valve shaft or bonnet seals.

As discussed previously, the HCU unit rooms are seismic Category 1and protected from missiles, and the ball check valve is classified as safety-related as it actuates closure of the scram inlet port by reverse flow under system pressure, and fluid flow conditions caused by a potential break in the scram line. The scram insert piping from the HCU room to the FMCRDs are designed in accordance with the ASME Code. Some CRDs are removed at each refueling outage such that defects could be discovered. Further, the design provides for detection capability such that a potential leak could be discovered by the containment leak detection system and isolated to ensure containment pressure remains within design limits. Therefore, the design complies with the provisions of SRP 3.6.2, as discussed in Item 5 of SRP 6.2.4 SRP Acceptance Criteria, as well, and compliance to SRP 3.6.2 will be added to LTR NEDE-33911P.

The guidance of Regulatory Guide (RG) 1.141, Containment Isolation Provisions for Fluid Systems, July 2020, Revision 1, C.1, allows for system integrity inspections applied to closed systems in lieu of leak testing, as stated in ANSI N271-1976 Section 3.6.4. Further RG 1.141, C.10, in accordance with Section 4.1.4 of ANSI N271-1976 states: “The piping between isolation barriers or piping which forms part of isolation barriers, shall meet the requirements of 3.7 and applicable requirements for isolation barriers.” Piping between isolation barriers should meet the applicable requirements of Section 3.5 or Section 3.7 of ANSI N271-1976. As described in NEDC-33911P, Section 5.2.4 (renumbered from 5.2.3), the CRD system CIV configuration meets and conforms to the guidance of RG 1.141.

The BWRX-300 CRD system meets the closed system containment isolation provisions for fluid systems as an alternative arrangement criteria.

Control Rod Drive System Design Basis Function

Given the other defined basis alternative containment configuration of GDC 55 and the guidance of RG 1.141 for closed system containment isolation provisions for fluid systems, an alternative arrangement is provided for the CRD system where the HCU outside containment forms a closed system outside containment. Each HCU has a normally closed scram valve that isolates the scram accumulator and CRD charging water header from two parallel scram hydraulic lines supplying their respective FMCRD. These same hydraulic lines serve as the flow path for normal purge flow to the FMCRDs. This purge flow is isolated from the CRD hydraulic supply by use of a check valve located on the HCU assembly. The hydraulic supply line to each FMCRD directs its flow through an internal ball check valve. Each hydraulic line from the HCUs has a manual isolation valve that can be used to remove it from service. 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. Adding additional isolation valves in this piping for the purpose of containment isolation is not in the direction of highest safety because it could become a new potential failure mode in a safety critical system and will not improve the containment integrity because the small diameter high pressure hydraulic lines are attached to a closed system outside containment and therefore do not cause a risk of containment leakage.

The control rod mechanical design incorporates a brake system and ball check valve, which reduces the chances of rapid rod ejection. The ball check valve functions as a safety-related component because it prevents reverse flow from the scram inlet port against the pressure and flow conditions caused by a break of the scram line, thereby stopping the loss of reactor coolant to the containment.

Hydraulic Control Unit

Each HCU furnishes pressurized water for hydraulic scram on a signal from the reactor protection system to drive two FMCRD mechanisms. The nitrogen gas bottle provides a source of readily available high-pressure, high-discharge flow rate of nitrogen to the accumulator.

Control Rod Drive Hydraulic System

The CRD system provides electric motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid insertion of control rods during abnormal operating conditions. High-pressure water stored in the individual HCUs provides the hydraulic power required for scram. Each HCU contains nitrogen-water accumulator charged to high pressure and the necessary valves and components to scram two FMCRDs. During normal operation, the HCUs provide a flow path for purge water to the associated FMCRDs. The CRD hydraulic system (CRDHS) supplies clean, demineralized water that is regulated and distributed to provide charging of the HCU scram accumulators and purge water flow to the FMCRDs during normal operation.

Fine Motion Control Rod Drive and FMCRD Pressure Boundary

The FMCRD design provides protection against loss of leak tight integrity. High pressure purge water continually flows through the drive with the water entering the ball check valve in the middle of the housing and flows around the hollow piston into the reactor. O-rings seal the lower housing. A labyrinth seal near the top of the drive restricts the flow into the reactor. During a scram, the labyrinth seals the high-pressure scram water from the reactor vessel. The ball check valve in the middle of the flange of the drive at the scram port serves as an isolation boundary in the unlikely event of a high pressure hydraulic insert line break. Reverse flow during a line break causes the ball to move to the closed position to prevent further loss of out of the break.

Summary

The containment isolation provisions for the CRD system FMCRD and HCU units constitute an application of the "other defined basis" alternative design basis to what is required by 10 CFR 50, Appendix A, GDC 55 because the small hydraulic line is pressurized to well above containment pressure, isolated from the RCPB by a ball check valve, and connected to a closed system outside containment. This design configuration allows the CRD system to remain in service for normal operational maneuvering and to perform its safety critical function to scram the reactor should an accident occur.

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 0, will be revised as follows:

1. Section 2.2.7.1, Containment Isolation Valves Connected to the RPV Boundary, will be revised to reflect that the hydraulic control units meet GDC 55 as an "other defined alternative

containment configuration design" by forming a closed system outside containment and is connected to an FMCRD inside containment with a normally open internal ball check valve to provide isolation from the RCPB.

2. Section 2.2.7.1, Containment Isolation Valves Connected to the RPV Boundary, will be revised to reflect that the hydraulic control units meet GDC 55 as an "other defined alternative containment configuration design" by forming a closed system outside containment and is connected to an FMCRD inside containment with a normally open internal ball check valve to provide isolation from the RCPB.

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2.2.7.1 Containment Isolation Valves Connected to RPV Boundary

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For the FMCRD hydraulic lines for the scram function shown on Figure 2-8, the containment penetrations do not have automatic isolation valves based on being a closed piping system outside containment and having RCPB isolation (internal ball check valve) in the design of the FMCRDs. The hydraulic control units meet GDC 55 as an "other defined basis" alternative containment configuration by forming a closed system outside containment. The control rod drive mechanical design incorporates a brake system and ball check valve that reduces the chances of rapid rod ejection. The ball check valve functions as a safety-related component because it prevents reverse flow from the scram inlet port against the pressure and flow conditions caused by a break of the scram line, preventing the loss of pressure from the underside of the hollow piston and generation of loads on the drive that could cause a rod ejection. This normally open ball check valve isolates the RCPB as needed from the small diameter, high pressure hydraulic insertion line that penetrates containment in order to attach to the HCU assembly. The HCU assembly serves as a closed system outside containment. At the HCU assembly, the hydraulic insertion line has a normally closed scram valve which allows high pressure water to flow from the accumulator as needed for a scram and there is also a normally open check valve isolating the purge water supply. Additionally, there are manual isolation valves that can be used to further isolate the HCU from the hydraulic insertion line as needed. Adding additional isolation valves in this piping for the purpose of containment isolation is not in the direction of highest safety because it could become a new potential failure mode in a safety critical system and will not improve the containment integrity because the small diameter high pressure hydraulic lines are attached to a closed system outside containment and therefore do not cause a risk of containment leakage.

The FMCRD design provides protection against loss of leak tight integrity. High pressure purge water continually flows through the drive with the water entering the ball check valve in the middle of the housing and flows around the hollow piston into the reactor. O-rings seal the lower housing. A labyrinth seal near the top of the drive restricts the flow into the reactor. During a scram, the labyrinth seals the high-pressure scram water from the reactor vessel. Reverse flow in the unlikely event of a hydraulic supply line break causes the ball check valve to move to the closed position. This prevents loss of pressure to the underside of the hollow piston, that in turn, prevents generation of loads on the drive that could cause rod ejection and serves as an isolation of the break from the RCPB. The scram insert piping from the HCU room to the FMCRDs are designed in accordance with Articles NB-2160 and NB-3120 of the ASME Code. The only primary pressure boundary components are the lower housing of the spool piece assembly and the flange of the outer tube assembly. These components are made with 300 series stainless steel materials in accordance with the ASME Code, Section III. Some CRDs are removed each refueling outage and disassembled for routine inspection of drive parts, including the CRD bolting and hard-surfaced parts accessible for visual examination in accordance with manufacturer's CRD maintenance procedures. The inspection program is adequate to detect any defects or leaks before they become serious operating problems. Further, the design provides for detection capability such that a potential leak could be discovered by the containment leak detection system and isolated to ensure containment pressure remains within design limits.

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5.3 NUREG-0800 Standard Review Plan Guidance

5.3.1 Standard Review Plan 3.6.2

Standard Review Plan (SRP) 3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Rev. 3, 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 GDC 4 Environmental and Dynamic Effects Design Bases be considered in the design structures, systems and components. This SRP 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. The guidance provides specific areas for review:

1. Defining break and crack locations and configurations,
2. Analytical methods to define forcing functions, including jet thrust reaction at the postulated pipe break or crack location and jet impingement loadings on adjacent safety-related SSCs,
3. The dynamic analysis methods used to verify the integrity and operability of mechanical components, component supports, and piping systems, including restraints and other protective devices under postulated pipe rupture loads,
4. The implementation of criteria used in defining pipe break and crack locations and configurations,
5. The criteria for dealing with special features such as augmented inservice inspection programs,
6. The acceptability of the analysis results, including jet thrust and impingement forcing functions and pipe-whip dynamic effects, and
7. The design adequacy of SSCs to ensure that the intended design functions will not be impaired to an unacceptable level of integrity or operability as a result of pipe whip or jet impingement loadings.

The BWRX-300 containment isolation system SSCs will conform to the guidance of the SRP, as well as meeting the requirements of GDC 4. The design of the containment structure, component arrangement, pipe runs, pipe whip restraints, and compartmentalization are 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 are engineered to preclude damage based upon break evaluations. A complete description of compliance to the SRP and associated branch technical positions, 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.

eRAI No.: 9746
Date of eRAI Issue: 07/10/20

NRC Question 03.06.02-4

SER Section 3.6.2: Section 2.2.2, "Containment Design Requirements," in NEDC-33911 states that ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE-1120 and the design criteria from NRC Branch Technical Position (BTP) 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Part B, Items 1(ii)(1)(d) and (e), are applied to eliminate postulating breaks and cracks in those portions of piping from the containment wall to including the outboard containment isolation valves (CIVs). Similar statements are also included in Section 5.1.7, "10 CFR Part 50, Appendix A, GDC 4," in NEDC-33911. Eliminating postulated breaks and cracks in those portions of piping from the containment wall is safety significant to provide assurance that the containment of the BWRX-300 reactor will not be breached and cause a radioactive release to the environment that exceeds regulatory requirements.

a. The NRC staff requests that GEH clarify that BTP 3-4, Part B, Items 1(ii)(2) through (7), if applicable, are also applied to eliminate postulating breaks and cracks in those portions of the piping.

b. The NRC staff requests GEH describe how the BWRX-300 design requirements will provide assurance that the functionality of those outboard CIVs will not be affected by the dynamic effects resulting from postulated pipe breaks beyond those portions of piping from the containment wall to including the outboard CIVs.

GEH Supplemental Response to NRC Question 03.06.02-4

c. The BWRX-300 will meet BTP 3-4 guidance similarly to the Economically Simplified Boiling Water Reactor (ESBWR) described in Design Control Document (DCD) 26A6642A5, Revision 10, Sections 3.6.1.1 and 3.6.2. In addition to ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-1120 and the design criteria from Branch Technical Position (BTP) 3-4, Items 1(ii)(1)(d) and (e), BTP 3-4, 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 ~~including~~ the outboard containment isolation valves (CIVs).

d. ~~BTP 3-4, B.1.(ii) Fluid System Piping in Containment Penetration Areas states that breaks and cracks need not be postulated in those portions of piping from containment wall to and including the inboard or outboard isolation valves provided that they meet the design criteria of ASME Code, Section III, Subarticle NE-1120 and additional design criteria (1) through (7). As stated in LTR NEDC-33911P, Section 2.2.2, Design Requirements, the BWRX-300 design applies the ASME Code Section III, Subarticle NE-1120 design criteria as well as BTP 3-4 Items 1(ii)(1)(d) and (e) for ASME Code, Section III, Class 2 piping. In response to item a above, BTP 3-4, Items 1(ii)(2) through (7) design criteria will also be applied to those portions of piping from containment wall to including the outboard CIV. As a result of applying the design criteria of ASME Code, Section III, Subarticle NE-1120 and BTP 3-4, Part B, Items 1(ii)(1)(d) and (e), and Items 1(ii)(2) through (7), the dynamic effects of postulated breaks and~~

~~cracks in those portions of the piping from the containment wall to the outboard CIV are not considered in the BWRX-300 design. No other additional changes to LTR NEDC-33911P are warranted as a result of adding BTP 3-4, Part B Items 1(ii)(2) through (7).~~ As discussed in response to NRC Questions 03.06.02-5 and 03.06.02-6, the BWRX-300 design will consider the dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids in the containment design and associated piping, valves, penetrations, and instrument lines in future licensing activities. These dynamic effects on containment piping for those portions of piping beyond and including the outboard CIVs will be evaluated for postulated breaks and cracks. LTR NEDC-33911P, Section 5.1.7 will be revised to address evaluating the dynamic effects of postulated breaks and cracks in those portions of the piping beyond and including the outboard CIVs in future licensing activities.

Proposed Changes to NEDC-33911P, Revision 0

a. NEDC-33911P, Revision 0, will be revised to reflect the revision of bullet five to Subsection 2.2.2, Design Requirements, and Section 5.1.7, 10 CFR 50 Appendix A, GDC 4, Statement of Compliance, to add compliance to the guidance of BTP 3-4, Part B, Items 1(ii)(2) through (7) that eliminates postulated breaks and cracks in piping in those portions of the piping from the containment wall up to the outboard CIVs:

. . .

2.2.2 Containment Design Requirements

. . .

Design Requirements:

. . .

- ASME B&PV Code, Section III, Division 1, Subarticle NE-1120, and the design criteria from BTP 3-4, Part B, Items 1(ii)(1)(d) and (e), and Items 1(ii)(2) through (7), are applied to eliminate postulated breaks and cracks in those portions of piping from the containment wall to the outboard CIVs.

5.1.7 10 CFR 50 Appendix A, GDC 4

. . .

Statement of Compliance: . . . As described in this LTR, the BWRX-300 design requirements include applying the design criteria from NUREG-0800, SRP, BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Part B, Items 1(ii)(1)(d) and (e), and Items 1(ii)(2) through (7) to eliminate postulating breaks and cracks in those portions of piping from containment wall to ~~and including~~ the outboard CIVs.

b. NEDC-33911P, Revision 0, Section 5.1.7, 10 CFR 50 Appendix A, GDC 4, Statement of Compliance, will be revised to address evaluating the dynamic effects of postulated breaks and cracks in those portions of the piping beyond and including the outboard CIVs in future licensing activities:

. . .

5.1.7 10 CFR 50 Appendix A, GDC 4

. . .

Statement of Compliance:…The dynamic effects of postulated breaks and cracks in those portions of the piping beyond and including the outboard CIVs will be evaluated in future licensing activities.

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eRAI No.: 9746
Date of eRAI Issue: 07/10/20

NRC Question 03.06.02-5

SER Section 3.6.2: Section 3.1, "Scope of the Evaluation Model," in NEDC-33911 states that 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 to provide assurance that a breach in the containment of the BWRX-300 reactor will not occur and cause a radioactive release to the environment that exceeds regulatory requirements. The NRC staff requests that GEH clarify that the BWRX-300 containment response to all of the dynamic effects (not only the jet loads) 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 10 CFR Part 50, Appendix A, GDC 4 requirements.

GEH Response to NRC Question 03.06.02-5

The BWRX-300 design will consider the dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids in the containment design and associated piping, valves, penetrations, and instrument lines in future licensing activities. NEDC-33911P, Revision 0, Section 3.1 and Section 5.1.7 will be revised to include the dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids in the BWRX-300 design of containment and CIV design features.

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 0, Sections 3.1 and 5.1.7 will be revised to reflect the dynamic effects of pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids to comply with the design requirements of 10 CFR 50. Appendix A, GDC 4:

3.1 Scope of the Evaluation Model

. . .

<u>The dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids will be evaluated in design of the containment and CIVs, and described during future licensing activities to comply with the design requirements of 10 CFR 50, Appendix A, GDC 4.</u>

5.1.7 10 CFR 50 Appendix A, GDC 4

. . .

Statement of Compliance: The BWRX-300 containment and CIVs design features . . . are to be designed to effects of, and to-be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents, and will consider the dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids. . . .

eRAI No.: 9746
Date of eRAI Issue: 07/10/2020

NRC Question 03.06.02-6

SER Section 3.6.2: Section 5.1.7 in NEDC-33911 states that breaks and cracks in those portions of piping from the RPV isolation valves that function as the inboard CIVs to the containment wall remain postulated to occur, and the dynamic effects of those postulated pipe breaks are to be evaluated in the BWRX-300 design. The NRC staff notes that during its review of NEDC-33910, in RAI Question 03.06.02-2, the NRC staff requested that GEH describe how the BWRX-300 design requirements will provide assurance that the functionality of those dual function safety-related valves will not be affected by the dynamic effects resulting from the postulated pipe breaks. In a letter dated May 4, 2020, GEH responded to the staff's request by referencing Section 5.1.7 in NEDC-33911, and stated that the BWRX-300 design will meet the requirements of 10 CFR Part 50, Appendix A, GDC 4. In addition, GEH stated that qualification, such as compliance with ASME Standard QME-1-2007 (or a later edition) as accepted in NRC Regulatory Guide 1.100, will be addressed in the detailed design of the valves and will be specified during future licensing activities. The capability of CIVs to perform their design-basis functions is safety significant to provide assurance that the containment of the BWRX-300 reactor can be safely isolated and prevent radioactive release to the environment that exceeds regulatory requirements. The NRC staff requests that GEH update Section 5.1.7 in NEDC-33911 to describe the valve qualification in compliance with ASME Standard QME-1-2007 as described above.

GEH Response to NRC Question 03.06.02-6

BWRX-300 containment isolation valve qualification, using ASME Standard QME-1-2007 (or later edition) as accepted in NRC Regulatory Guide 1.100, will be addressed in the detailed design of the valves and will be specified during future licensing activities. NEDC-33911P, Revision 0, Sections 2.2.7 and 5.1.7 will be revised to reflect the use of the ASME Standard QME-1-2007 (or later edition).

Proposed Changes to NEDC-33911P, Revision 0

NEDC-33911P, Revision 0, Sections 2.2.7 and 5.1.7 will be revised to add valve qualification compliance to ASME Standard QME-1-2007 or later edition:

2.2.7 Containment Isolation Valves

Design Requirements:

. . .

- <u>Valve qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as endorsed by NRC Regulatory Guide (RG) 1.100, will be addressed in the detailed design and the procurement process of the valves, and will be specified during future licensing activities.</u>

5.1.7 10 CFR 50 Appendix A, GDC 4

. . .

Statement of Compliance: . . . Internal containment flooding is to be evaluated during future licensing activities. Valve qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as endorsed by NRC Regulatory Guide (RG) 1.100, will be addressed in the detailed design and the procurement process of the valves, and will be specified during future licensing activities.

Appendix B
Replaced Pages from NEDC-33911P Revision 0, Supplement 1

REVISION SUMMARY

Revision Number	Description of Change
0	Initial Issue
Supplement 1	Revised to incorporate the following: • [[ ]] from NEDC-33910P for Sections 2.1, 2.2, 2.2.3, 2.2.7.1, 3.1, 5.1.11, 5.3.8. • Sections 3.2, 3.3, 3.5, 3.6, 5.1.17, 5.3.3 and 6.5 are revised to reflect a title change for NEDC-33922P from GOTHIC Application for BWRX-300 Containment to BWRX-300 Containment Evaluation Method. • Changed Section Numbers as a result of response to eRAIs 9745: 5.2.3 to 5.2.4, 5.2.4 to 5.2.6 and 5.2.5 to 5.2.7. • Changed "Since" to "Because" in Sections 3.1 and 3.2. • Information that has been reclassified as non-proprietary is identified with change bars for Section 2.1.2, last paragraph; 2.2, 2nd paragraph; 2.2.2, 4th bullet; 2.2.7.3; 3.1, 4th, 5th and 6th paragraphs; 5.1.4; 5.1.5; 5.1.6; 5.1.7; 5.1.24; 5.3.3; 5.3.4, 2nd paragraph; 5.3.9, 2nd paragraph; and 5.4.1. • NRC Requests for Additional Information (eRAIs): – NRC eRAI 9745, Question 03.09.06-15, item (h) where Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves was added as Section 5.5.2 as this operating experience may be applicable to the detailed design of the CIVs to avoid thermal binding. – NRC eRAI 9745, Question 03.09.06-17, where Section 2.2.7 Containment Isolation Valves Design Requirements was revised to add two additional bullets that address CIV design to prevent valve movement from normal operating position will be accomplished by positive mechanical means. – NRC eRAI 9745, Question 03.09.06-18, where new Sections 5.2.3, 5.2.5, and 5.2.8 for Regulatory Guides 1.84, 1.147 and 1.192, respectively, were added to reflect ASME Code Cases for design, inservice inspection and IST activities in satisfying 10 CFR 50.55a. Renumbering of Section 5.2 occurred due to the insertion of these Regulatory Guides. – NRC eRAI 9745, Question 03.09.06-19, where new Section 5.3.1 Standard Review Plan 3.9.6 was added to specify compliance with

	this guidance for functional design, qualification and IST programs for pumps, valves, and dynamic restraints for containment isolation valves. Renumbering of subsequent 5.3 sections follows. – NRC eRAI 9745, Question 03.09.06-20, where Sections 5.4 and 5.5 were revised to reflect that generic issues and operational experience would be provided in future licensing activities either by 10 CFR 50 or 10 CFR 52 licensing activities, and that the operational experience and generic communications provided in the LTR are based upon their relevance to the scope of the LTR only. – NRC eRAI 9745, Question 03.09.06-15, where Section 2.2.7 Containment Isolation Valve Design Requirements added a new bullet that addresses diversity for RPV isolation valve penetrations and where two containment isolation valves that have automatic isolation with RPV isolation valves ensure diversity; additionally, added new Section 5.5.2 Generic Letter 95-07 Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves that addresses applicability of pressure locking and thermal binding of valves during future licensing activities. – NRC eRAI 9745, Question 03.09.06-17, where Section 2.2.7 Containment Isolation Valve Design Requirements added three new bullets that addresses positive mechanical means in valve actuators to maintain these valves in their required post-accident valve positions. – NRC eRAI 9745, Question 03.09.06-18, where new Sections 5.2.3, 5.2.5 and 5.2.8 were added to reflect BWRX-300 compliance to the guidance of Regulatory Guides 1.84, 1.147, and 1.192, respectively for the acceptability of ASME Code Cases. Renumbering of previous and later sections in Section 5.2 occurred to reflect these new sections. – NRC eRAI 9745, Question 03.09.06-19, where new Section 5.3.2 Standard Review Plan 3.9.6 was added to indicate BWRX-300 compliance to functional design, qualification and inservice testing for pumps, valves and dynamic restraints guidance. Renumbered subsequent sections in Section 5.3 to reflect new Section 5.3.1 (see eRAI 9758) and 5.3.2. – NRC eRAI 9745, Question 03.09.06-20, where Section 5.4 Generic Issues and Section 5.5 Operational Experience and Generic Communications was modified to reflect that an up-to-date evaluation of relevant generic communications and experiences would be evaluated during future licensing activities in support of a 10 CFR 52 DCA or a 10 CFR 50 CP and OP application.

	– NRC eRAI 9746, Question 03.06.02-4, where bullet five was added to Subsection 2.2.2 Design Requirements that specified that BTP 3-4, Part B, Items 1(ii)(1)(d) and (e), and Items 1(ii)(2) through (7), are applied and Section 5.1.7 10 CFR 50 Appendix A, GDC 4 was modified to add compliance to BTP 3-4 Part B, Items 1(ii)(2) through (7) and the dynamic effects of those portions of the piping from the containment to the outboard CIVs. Section 5.1.7 was revised to reflect that the dynamic effects of postulated breaks and cracks in those portions of the piping beyond and including the outboard CIVs will be evaluated in future licensing activities. – NRC eRAI 9746, Question 03.06.02-5, where Section 3.1 and Section 5.1.7 was revised to show compliance to GDC 4 by evaluating the dynamic effects of jet loads, pipe whipping, postulated high-energy breaks, missiles and discharging fluids in the design of containment and the CIVs and described during future licensing activities to show compliance with GDC 4. Section 5.1.7 10 CFR 50, Appendix A, GDC 4 compliance was revised to indicate that the dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids will be included. – NRC eRAI 9746, Question 03.06.02-6, where Section 2.2.7 Design Requirements was revised to add a new bullet that shows ASME Standard QME-1-20007 (or later edition) compliance for the design and procurement of the valves specified in future licensing activities. Section 5.1.7, 10 CFR 50 Appendix A, GDC 4 was also revised to comply with this same ASME Standard. – NRC eRAI 9760, Question 06.02.05-1, where Section 5.1.2 10 CFR 50.44 was revised to show that reliable equipment will be provided to monitor both oxygen and hydrogen concentrations in the BWRX-300 inerted containment during and following a BDBA. – NRC eRAI 9764, Question 06.02.01-1, where Section 5.1.17 10 CFR 50, Appendix A, GDC 50 compliance was revised to indicate that the containment structural design will be evaluated against the maximum expected external pressure with sufficient margin from a full spectrum of postulated accidents that would release reactor coolant to containment. – NRC eRAI 9766, Question 06.02.01-3, where Sections 3.4.2.2 GOTHIC Phenomenon Identification and Ranking Table (PIRT), 3.4.2.3 PIRT Survey, 3.4.2.4 Development of the Assessment Base and Tables 3-1 Phenomenon Ranking Criteria and 3-2 Phenomena Identification and Ranking Table for Containment (Excluding

	RPV) are moved to LTR NEDC-33922P BWRX-300 Containment Evaluation Method. – NRC eRAI 9758, Question 06.02.04-1, where Section 2.2.7.1 was revised to state that the outside containment automatic CIVs closure time will be based upon the first fission product release greater than what is contained in the normal reactor coolant in source term evaluations and will be completed in future licensing activities. – NRC eRAI 9758, Question 06.02.04-2, where Section 2.2.7.1 was revised to reflect the design requirement of a closed loop outside containment plus two in-series automatic isolation valves inside containment meet the other defined basis provision of GDC 55. – NRC eRAI 9758, Question 06.02.04-3, where Section 2.2.7.1 was revised to reflect that the HCUs meet GDC 55 as an “other defined alternative containment configuration design” by forming a closed system outside containment and is connected to an FMCRD inside containment with a normally open internal ball check valve to provide isolation from the RCPB and new Section 5.3.1, Standard Review Plan 3.6.2 was added to reflect compliance to the provisions of SRP 6.2.4, Item 5, that requires compliance to SRP 3.6.2 for the BWRX-300 CRD system HCU piping and ball check valve alternative containment isolation valve arrangement.

Term	Definition
IE	Infrequent Event
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LTR	Licensing Topical Report
LWR	Light-Water-Reactor
NBS	Nuclear Boiler System
NC	Non-condensable
NDTT	Nil-Ductility Transition Temperature
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OL	Operating License
PCCS	Passive Containment Cooling System
PCV	Primary Containment Vessel
PIRT	Phenomenon Identification and Ranking Table
PSAR	Preliminary Safety Analysis Report
PWR	Pressurized Water Reactor
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SA	Severe Accident Management
SBO	Station Blackout
SMR	Small Modular Reactor
SRP	Standard Review Plan
SSC	Structure, System, and Component
TAF	Top of Active Fuel
TMI	Three Mile Island
TRACG	Transient Reactor Analysis Code General Electric

2.0 TECHNICAL EVALUATION OF CONTAINMENT PERFORMANCE

2.1 General Introduction

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

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

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

2.1.1 Reactor Pressure Vessel

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

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

Figure 2-1 shows a representation of BWRX-300 RPV and internals.

The ICS is initiated automatically on high RPV pressure indicating an overpressure event or on signals indicating a LOCA. To start an IC train, the IC condensate return valve is opened whereupon the standing condensate drains into the reactor. [[

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

2.2 Overview of Containment

The BWRX-300 containment is based upon GEH BWR experience and fleet performance:

- Containment size comparable to a small BWR drywell
- Containment peak accident pressure and temperatures within existing BWR experience base
- Containment load simplified when compared to conventional BWRs with pressure suppression containments
- Nitrogen-inerted containment same as BWR Mark I and Mark II containments
- Pressure and temperature during normal operation maintained by fan coolers, similar to existing BWRs
- Upon loss of active containment cooling, heat removal is achieved by PCCS

The BWRX-300 containment is an underground subterranean steel or reinforced concrete primary containment vessel (PCV) or a combination of steel and reinforced concrete. Figure 2-3 shows a typical steel containment. Other potential construction types are of similar size and have the same functional features. The containment does not have a suppression pool. Heat is removed by PCCS as described in Section 2.2.8. The reactor cavity pool for PCCS heat removal during design basis events is located above containment and is vented to the atmosphere.

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. Within these subcompartments there are no large bore high energy lines. Typical small piping [[ ]] located within these subcompartments include the FMCRD hydraulic lines and instrument lines. Large bore high energy lines are also located as far as practical from the outside of these subcompartment walls. Therefore, line breaks inside or outside these subcompartments do not create significant pressure differentials across the subcompartment walls.

Combustible gas control is not required 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 the containment atmosphere is initially nitrogen-inerted.

The PCV has provisions for personnel access (see Section 2.2.5) and for habitability during plant outages to perform maintenance, inspections and tests required for assuring PCV integrity and reliability, and the integrity and performance reliability of interfacing structures, systems, and components (SSCs) contained inside the PCV enclosure.

2.2.2 Containment Design Requirements

The PCV is classified as a Safety Class 1, safety-related, and seismic Category I structure.

Design Requirements:

- The PCV is designed either as a metal containment in accordance with the rules and requirements of American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (B&PV) Code, Section III, Division 1, Subsection NE, or as a concrete containment in accordance with the rules and requirements of ASME B&PV Code, Section III, Division 2, which is a dual standard with ACI-359.
- Piping systems that pass through PCV mechanical penetrations and CIVs, with the exception of the [[ ]] are designed in accordance with the rules and requirements of ASME B&PV Code, Section III, Division 1, Subsection NC, Class 2 Components.
- [[ ]] that function as the inboard CIVs are designed in accordance with the rules and requirements of ASME B&PV Code, Section III, Division 1, Subsection NB, Class 1 Components.
- For piping connected to the RPV isolation valve assemblies, extending to the containment wall, the BWRX-300 design requirements include identification of postulated pipe rupture locations and configurations inside containment as specified in NUREG-0800, Standard Review Plan (SRP), Branch Technical Position (BTP) 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment,” Part B, Item 1(iii)(2), and identification of leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2).
- ASME B&PV Code, Section III, Division 1, Subarticle NE-1120, and the design criteria from BTP 3-4, Part B, Items 1(ii)(1)(d) and (e), and Items 1(ii)(2) through (7), are applied to eliminate postulating breaks and cracks in those portions of piping from containment wall to the outboard CIVs.
- Structural supports for piping systems and components inside the PCV are designed in accordance with the rules and requirements of ASME B&PV Code, Section III, Division 1, Subsection NF, Supports.
- Materials used for the PCV, penetration piping systems and the associated supports are designed in accordance with the rules and requirements of ASME B&PV Code, Section II, Material Specifications. Exception to the materials requirement is allowed for the nonconductive portions of electrical penetrations.
- Additional structures that are part of the PCV internals are designed in accordance ANSI/AISC N690, Specification for Safety-Related Steel Structures for Nuclear Facilities, with Supplements.

2.2.3 Containment Performance Requirements

The BWRX-300 PCV is sized and equipped to contain the mass and energy released by a large break LOCA [[ ]], and for small breaks [[ ]].

In addition, the PCV volume is sufficient to accept the additional non-condensable (NC) gas from the ICS vents, as a backup discharge volume, when the ICS is in operation during any plant operating mode or condition.

The PCV design is for a service life of 60 years.

2.2.4 Containment Boundary

The PCV physical design boundary is used to interpret design code applicability to the PCV and its component parts, including the following:

- The shell bottom head supported from the basemat and any external bottom head supports to the interfacing connection with the civil structure;
- Outside diameter of the PCV wall from the bottom head to the transition ring;
- The transition ring including the neck to the shell flange, and the flanged closure head and flange bolting;
- Any external support structures attached to or forming part of the PCV wall exterior, particularly for the transfer of load to support the RPV, to the interfacing connection with the civil structure;
- The outer surface extent of PCV hatches and airlocks;
- The PCV penetration sleeves up to the interface connecting weld joint between the sleeve closure plate or bellows and the process piping (duct), tubing penetration assembly or electrical penetration assembly;
- The outboard CIVs, including pipe support(s) and the portion of pipe beyond CIVs where the first pipe supports are affixed;
- The outer closure of electrical penetration assemblies; and,
- [[ ]] (see Section 2.2.8).

A description of containment heat removal design functions and design features can be found in Section 2.2.8, and key phenomena important to the analysis of the BWRX-300 containment response in design basis events are described in Section 3.4.

2.2.5 Access and Maintenance

The PCV has a flexible metallic seal, i.e., refueling bellows between the RPV exterior surface and PCV wall interior. The refueling bellows assembly is designed to accommodate the movement of the vessel caused by operating temperature variations and seismic activity. The refueling bellows is permanently installed by welded joints to specified attachment interface locations below the RPV and PCV head closure flanges. The refueling bellows provides a 360° structural barrier that retains the refueling cavity water above the PCV when the PCV head is removed. The design of

- Isolation valves for instrument lines that penetrate containment conform to the requirements of RG 1.11, Instrument Lines Penetrating the Primary Reactor Containment.
- Isolation valves, actuators and controls are protected against the loss of their safety-related function from missiles and postulated effects of high and moderate energy line ruptures.
- Design of the CIVs, and associated piping and penetrations will meet the requirements of seismic Category I components, and designed in accordance with the rules and requirements of ASME B&PV Code, Section III, Division 1, Subsection NE, Class MC Components, and Subsection NC, Class 2 Components, in accordance with their quality group classification.
- The design of the control functions for automatic CIVs ensure that resetting the isolation signal shall not result in the automatic reopening of CIVs.
- Penetrations with trapped liquid volume between the isolation valves have adequate relief for thermally induced pressurization.
- Diversity for penetrations where RPV isolation valves are credited as one of the containment isolation valves is accomplished by actuation from separate and diverse control systems that are single failure proof. In other penetrations where two containment isolation valves are used that have automatic isolation, diverse actuation signals are applied to ensure the function is achieved.
- The CIVs for main steam line, feedwater, shutdown cooling, and reactor water cleanup shall fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means.
- [[ ]] with valve actuators designed to maintain the valves in their as is position by positive mechanical means.
- All other CIV penetration configurations will be designed with valve actuators with positive mechanical means to ensure that upon automatic actuation or a loss of signal or control power to both valves, the valves will be maintained in the required post accident valve position.
- Valve qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as endorsed by NRC Regulatory Guide (RG) 1.100, will be addressed in the detailed design and the procurement process of the valves, and will be specified during future licensing activities.

The FMCRDS shown in Figure 2-8 are similar mechanically as the ESBWR with the exception that the [[ ]] FMCRD system design is described in the ESBWR Design Control Document Tier 2, Chapter 4 Reactor, 26A6642AP Rev. 10, Section 4.6 [Reference 6.1].

2.2.7.1 Containment Isolation Valves Connected to RPV Boundary

The BWRX-300 RPV design, acceptance criteria, and performance is delineated in Licensing Topical Report (LTR) NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 6.2]. [[

]]

The outside containment automatic CIVs closure time will be established to assure containment isolation prior to the first fission product release greater than what is contained in normal reactor coolant in source term evaluations which will be completed in future licensing activities. These closure times are expected to be in the order of minutes. Additionally, the valve closing time for all CIVs will support specific break isolation functions balanced with water hammer and valve loading considerations. LTR NEDC 33921P, Severe Accident Management, will provide the evaluation for fission product releases resulting from BDBA or SA events and provide the necessary timing information to establish the closure times for the outside containment isolation valves.

Small pipes for level instruments use Excess Flow Check Valves (EFCVs) to conform to the requirements of RG 1.11, Instrument Lines Penetrating the Primary Reactor Containment.

For the ICS as shown in Figure 2-7, [[

]] comply with the "other defined basis" alternative containment configuration requirements of GDC 55.

[[

]]. An alternative arrangement is provided for the ICS where two in-series RPV isolation valves on each end of the system function as containment isolation valves. A break in an ICS line either inside or outside containment could be isolated by either of the two redundant [[
]]. The piping in the area between the outermost [[
]] and the containment boundaries, as well as the piping through the seismic Category I reactor building where the ICS steam supply and the ICS condensate return piping connect to the ICS heat exchanger located in the ICS pool, are designed using ASME Section III, Class 1, NB piping, which limits the probability of breaks in these segments of the piping. Additionally, a break, between the RPV isolation valves and the containment would be isolated by the RPV isolation valves to stop the leak and would be contained by the closed system outside containment that is designed to withstand full reactor pressure. It would require an additional break before a radioactive release could occur, and even with an additional break, the coolant source remains isolated. Therefore, this design can accommodate a single failure and maintain containment leak integrity.

The ICS steam supply lines for each train contain two in-series valves inside containment, combined with a closed loop outside containment, thus providing containment isolation. The ICS steam supply lines and condensate return lines pass through the seismic Category 1 reactor building in order to connect to the isolation condenser in the ICS pool. The ICS vent lines each contain two in-series containment isolation valves and are attached to the closed loop outside containment. The ICS condensate return line for each train has two valves in-series to provide RPV isolation and containment isolation functions and are located inside containment where they are protected from outside environmental conditions that may result from a failure outside containment. The ICS condensate return line along with the steam supply line is automatically isolated when leakage is detected in the specific ICS train.

Given the above rationale, the containment isolation provisions for the ICS condensate, steam, vent and purge lines constitute an appropriate application of the "other defined basis" alternative defined in 10 CFR 50, Appendix A, GDC 55. A single failure would not disable the containment isolation function, while allowing the [[
]] to remain open to allow the ICS to function as a part of the ECCS.

[[

]]

Figure 2-7: Isolation Condenser CIVs Connected to the RPV Boundary

For the FMCRD hydraulic lines for the scram function shown on Figure 2-8, the containment penetrations do not have automatic isolation valves based on being closed piping system outside containment and having RCPB isolation (internal ball check valves) in the design of the FMCRDs. The hydraulic control units meet GDC 55 as an "other defined basis" alternative containment configuration by forming a closed system outside containment. The control rod drive mechanical design incorporates a brake system and ball check valve that reduces the chances of rapid rod ejection. The ball check valve functions as a safety-related component because it prevents reverse flow from the scram inlet port against the pressure and flow conditions caused by a break of the scram line, preventing the loss of pressure from the underside of the hollow piston and generation of loads on the drive that could cause a rod ejection. This normally open ball check valve isolates the RCPB as needed from the small diameter, high pressure hydraulic insertion line that penetrates containment in order to attach to the HCU assembly. The HCU assembly serves as a closed system outside containment. At the HCU assembly, the hydraulic insertion line has a normally closed scram valve which allows high pressure water to flow from the accumulator as needed for a scram and there is also a normally open check valve isolating the purge water supply. Additionally, there are manual isolation valves that can be used to further isolate the HCU from the hydraulic insertion line as needed. Adding additional isolation valves in this piping for the purpose of containment isolation is not in the direction of highest safety because it could become a new potential failure mode in a safety critical system and will not improve the containment integrity because the small diameter high pressure hydraulic lines are attached to a closed system outside containment and therefore do not cause a risk of containment leakage.

The FMCRD design provides protection against loss of leak tight integrity. High pressure purge water continually flows through the drive with the water entering the ball check valve in the middle of the housing and flows around the hollow piston into the reactor. O-rings seal the lower housing. A labyrinth seal near the top of the drive restricts the flow into the reactor. During a scram, the labyrinth seals the high-pressure scram water from the reactor vessel. Reverse flow in the unlikely event of a hydraulic supply line break causes the ball check valve to move to the closed position. This prevents loss of pressure to the underside of the hollow piston, that in turn, prevents generation of loads on the drive that could cause rod ejection and serves as an isolation of the break from the RCPB. The scram insert piping from the HCU room to the FMCRDs are designed in accordance with Articles NB-2160 and NB-3120 of the ASME Code. The only primary pressure boundary components are the lower housing of the spool piece assembly and the flange of the outer tube assembly. These components are made with 300 series stainless steel materials in accordance with the ASME Code, Section III. Some CRDs are removed each refueling outage and disassembled for routine inspection of drive parts, including the CRD bolting and hard-surfaced parts accessible for visual examination in accordance with manufacturer's CRD maintenance procedures. The inspection program is adequate to detect any defects or leaks before they become serious operating problems. Further, the design provides for detection capability such that a potential leak could be discovered by the containment leak detection system and isolated to ensure containment pressure remains within design limits.

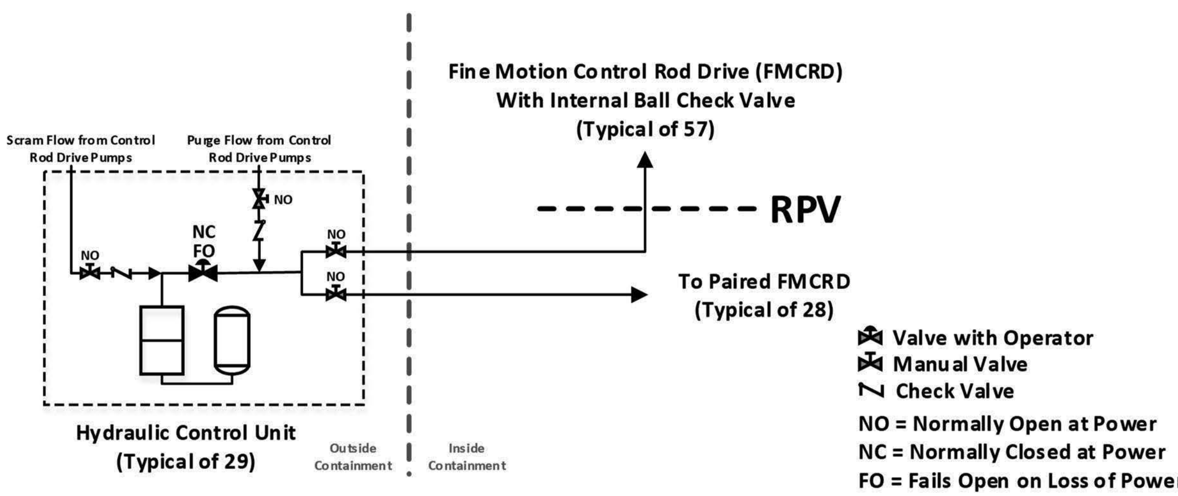

Figure 2-8: FMCRD CIVs Connected to RPV Boundary

2.2.7.2 Containment Isolation Valves Connected to Containment Atmosphere

The BWRX-300 CIVs attached directly to the containment atmosphere and shown on Figure 2-9 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 are provided with two normally closed outside containment manual CIVs. 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 is provided with normally closed inside and outside containment automatic CIVs.

The process gas and radiation monitoring system is a closed system outside containment and is provided with normally open outside containment automatic CIVs because it is an essential system following beyond design basis events and severe accidents.

The floor drain sump line is provided with two normally closed outside containment automatic CIVs, because it is not practicable to include an inside containment automatic CIV 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.

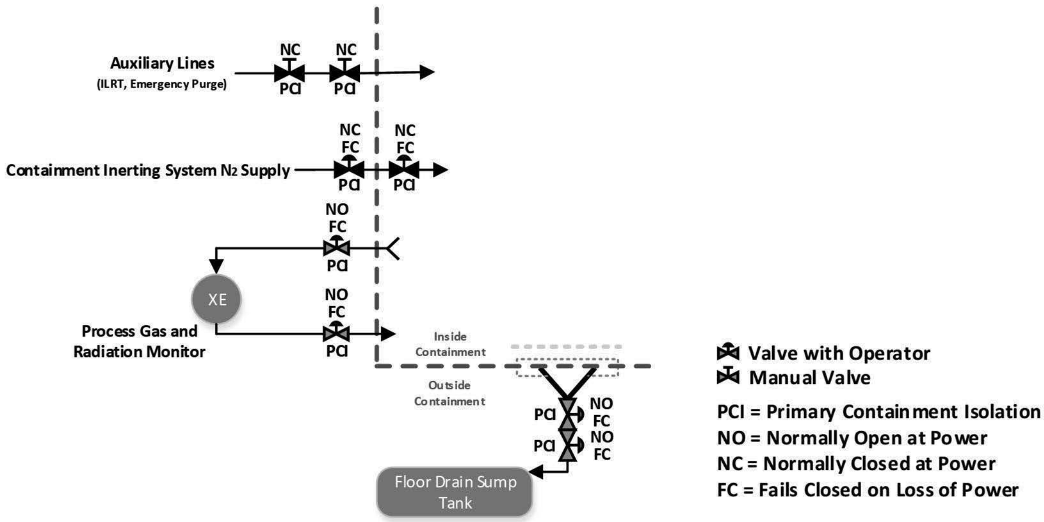

Figure 2-9: CIVs Connected to Containment Atmosphere

2.2.7.3 Containment Isolation Valves Connected to Closed Systems

The BWRX-300 closed system CIVs shown on Figure 2-10 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 pneumatic nitrogen or air system and the quench tank supply system are provided with either normally open or normally closed inside and outside containment automatic CIVs.

The service and breathing air system and demineralized water system are provided with normally closed inside and outside containment manual CIVs.

The chilled water supply and return are provided with normally open outside containment automatic CIVs.

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

3.1 Scope of the Evaluation Model

The design basis events for the containment are:

- AOO
- SBO
- ATWS
- Large break LOCA [[ ]] inside the containment
- Small breaks [[ ]] inside the containment

Because there is no discharge of steam or liquid into the containment in AOO, SBO and ATWS events, the only heat load to the containment is the heat transferred through the pipe and RPV insulation. Because 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.

Large break LOCA events inside the containment are the double-ended guillotine break of one of the following pipes:

- Main steam pipe
- Isolation condenser steam pipe
- Feedwater pipe
- Isolation condenser condensate return pipe

The pipes that are subject to a large break LOCA have two RPV isolation valves. At least one of the two valves on the broken line is closed subject to single failure criterion.

Small breaks inside the containment are assumed to remain unisolated. These small pipes include instrument lines.

The objective of the evaluation model is to demonstrate that the design pressure and temperature bound the accident peak pressure and temperature, and that the heat removal systems reduce the containment pressure rapidly. The acceptable results will demonstrate compliance with GDC 38 and GDC 50. The target for rapid depressurization is to reduce the pressure to the 50% of the peak accident pressure of the most limiting LOCA in 24 hours. The results are also used for equipment environmental qualification. Peak air/steam temperature resulting from a LOCA is not a meaningful parameter that can be compared to design limits for the structures. The figure of merit for temperature is the structure temperature, which can be compared to the design limits.

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.

The dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids will be evaluated in design of the containment and CIVs, and described during future licensing activities to comply with the design requirements of 10 CFR 50, Appendix A, GDC 4. Jet loads resulting from pipe breaks are not in the scope of the evaluation method described in this section. The jet loads and zone of influence are evaluated using a separate structural method that will be described during future licensing activities. However, the postulated break locations, type of break, and mitigating features for RPV and containment performance are within the scope of this document and LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 6.2].

3.2 Overview of the Evaluation Model

The evaluation model for the BWRX-300 containment response utilizes the applicable parts of the ESBWR evaluation methods which have been reviewed and approved for the ESBWR Design Certification [Reference 6.4].

BWRX-300 RPV is like the ESBWR RPV; however, the BWRX-300 containment is different than the ESBWR containment.

The most challenging features of the ESBWR containment for modeling are the wetwell, suppression pool, PCCS (which is much different and more complicated than the BWRX-300 PCCS), and the annulus between the RPV and the biological shield which is subject to pressurization and acoustic loads. The BWRX-300 containment does not have any of the above features. However, conservative temperature and steam / NC gas composition distributions can be calculated for the BWRX-300 containment using an appropriate model with nodalization.

The BWRX-300 containment evaluation model uses the Transient Reactor Analysis Code General Electric (TRACG) ESBWR RPV model described in Section 3.3. The containment is modeled separately using Generation of Thermal-Hydraulic Information for Containments (GOTHIC) Version 7.2a or the latest version. [[

]]

The computer codes used in the containment evaluation, TRACG and GOTHIC, are mature codes, each having an extensive qualification base, and each having been reviewed in detail. The application method developed for the purposes of the BWRX-300 containment evaluation follows the applicable sections of the Regulatory Guide 1.203 for a conservative analysis utilizing mature computer codes. Conservatism in the evaluation model is achieved by biasing the inputs and modeling parameters to bound the uncertainties, rather than performing a statistical analysis. The conservatism of the evaluation model is demonstrated by benchmarking to the available test data, which is to be established as part of the application methodology in LTR NEDC-33922P, BWRX-300 Containment Evaluation Method [Reference 6.5].

3.3 TRACG Mass and Energy Releases for Containment

Mass and energy release are calculated by TRACG and is the primary GEH tool for RPV neutronics and thermal-hydraulics calculations previously submitted in these GEH LTRs:

- NEDE-32176P, Revision 4, TRACG Model Description
- NEDE-32177P, Revision 2, TRACG Qualification
- NEDC-32725P, Revision 1, TRACG Qualification for SBWR
- NEDC-33080P, Revision 1, TRACG Qualification for ESBWR

Previous TRACG Containment/LOCA submittals for the models and qualification of TRACG are applicable to BWRX-300. The method accounts for the uncertainties and compensates for them by biases in the modeling parameters and in the plant parameters. BWRX-300 containment analysis method utilizes only those sections of the ESBWR Containment/LOCA analysis method related to the RPV and break flow, and correlations and biases. [[ ]] The BWRX-300 TRACG model outputs are to be provided in LTR NEDC-33922P, BWRX-300 Containment Evaluation Method [Reference 6.5], or a separate TRACG LTR.

3.4 GOTHIC Containment Model

3.4.1 Overview of the GOTHIC Computer Code

GOTHIC is a general-purpose thermal-hydraulics software package specifically developed for nuclear power plant containments and similar confinements by the nuclear industry. GOTHIC solves the conservation equations for mass, momentum and energy equations in multi-dimensional and/or lumped-parameter volumes. The conservation equations are solved for steam/gas mixture, continuous liquid, and liquid droplets. In addition, GOTHIC allows for the secondary fields for mist and liquid components. The NC gases may be composed of several species.

GOTHIC has been used in the industry extensively for containment pressure and temperature analyses, and equipment environment qualification outside the containment. GEH currently uses GOTHIC Version 7.2a but intends to use newer versions in the future. GOTHIC 7.2a includes several condensation models in the presence of NC gases that were lacking in earlier versions. Therefore, no code changes or additions are required to model the phenomena applicable to the BWRX-300 containment.

3.5 TRACG and GOTHIC Analyses Numerical Convergence

Numerical convergence of TRACG and GOTHIC individually, and the convergence of the iteration is part of the development of the application method. Both TRACG and GOTHIC have internal convergence criteria and report the total numerical error in the output. Both codes limit the time step size automatically to maintain the error below the acceptance criteria.

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.

A BWRX-300 containment nodalization study is to be included to demonstrate that finer nodalization than used in the application method does not have a significant effect on the results.

Finally, TRACG and GOTHIC analyses iterations continue until there is no significant change in the containment pressure and temperature. The criteria for the acceptance of the sufficiency of convergence is to be established as part of the application methodology in LTR NEDC-33922P, BWRX-300 Containment Evaluation Method [Reference 6.5].

3.6 Summary of the Containment Evaluation Method

The BWRX-300 containment evaluation method for the design basis events uses the TRACG ESBWR model for the mass and energy release from the RPV, and the heat transfer from the RPV and attached piping through the insulation are used as boundary conditions for the GOTHIC containment response model. The TRACG model for the RPV has been previously reviewed in detail for the ESBWR design, which is very similar to the BWRX-300 RPV. GOTHIC code is specialized for containment analyses, particularly for dry containments. All phenomena ranked high or medium are modeled in GOTHIC. Both TRACG and GOTHIC are well qualified codes in their respective fields and have been used extensively over a few decades.

In order to establish a conservative evaluation method, the applicable steps in RG 1.203 are being followed. The steps up to and including the PIRT have been completed and presented in the sections above. Section 3.4.2.4 establishes the remaining RG 1.203 elements to be completed, while Section 3.5 discusses the numerical convergence of the TRACG and GOTHIC models. This establishes that all phenomena related to the containment evaluations for the design basis events are covered in TRACG and GOTHIC methodology codes. The other elements of the method, including the demonstration analyses, and the specifics of the application method are planned to be delineated in LTR NEDC-33922P, BWRX-300 Containment Evaluation Method [Reference 6.5].

combustible gas mixtures may accumulate. With an inerted containment, oxygen concentrations reaching flammable mixture levels in subcompartments become a concern even if the average concentration is below the limit. The only subcompartment that may experience this phenomenon is the containment head section above the refueling bellows. However, for DBAs, natural circulation due to the presence of the passive containment cooling and the very low oxygen concentration in the main section of containment prevent significant oxygen accumulation above the refueling bellows. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.44(c)(1) for DBAs.

Compliance with this requirement for beyond design basis events and severe accidents are to be addressed in LTR NEDC-33921P, BWRX-300 Severe Accident Management [Reference 6.9].

- Regulatory Requirement: 10 CFR 50.44(c)(2), Combustible gas control, requires that all containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features.

  Statement of Compliance: The design feature of the BWRX-300 used to comply with this requirement includes the requirement for a dry, inerted containment. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.44(c)(2).

- Regulatory Requirement: 10 CFR 50.44(c)(3), Equipment Survivability, requires that containments that do not rely upon an inerted atmosphere to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to be considered must be equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding surrounding the active fuel region.

  Statement of Compliance: The design feature of the BWRX-300 used to comply with this requirement includes the requirement for a dry, inerted containment that does not rely upon combustible gas control to maintain safe shutdown and containment structural integrity. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.44(c)(3).

- Regulatory Requirement: 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).

  Statement of Compliance: 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).

- Regulatory Requirement: 10 CFR 50.44(c)(5), Structural analysis, requires that an applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC and include assumptions, analyses, and related information used in their coping evaluations available for NRC review.

Statement of Compliance: The BWRX-300 design includes Class 1E battery-backed 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, and 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 PCCS for containment depressurization and heat removal is passive and does not require onsite or offsite electric power system operation, including Class 1E battery-backed DC power. CIV automatic actuation isolation functions do not require offsite electric power system operation, and only requires one-time automatic actuation using onsite Class 1E battery-backed DC power and then remain isolated for at least 72 hours without any further need of onsite or offsite electric power system operation. The coping analysis to demonstrate 72 hours will be completed during future licensing activities. Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50.63.

5.1.5 10 CFR 50 Appendix A, GDC 1

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

  Statement of Compliance: The BWRX-300 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.

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

5.1.6 10 CFR 50 Appendix A, GDC 2

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

Statement of Compliance: The BWRX-300 containment and CIVs 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.

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

5.1.7 10 CFR 50 Appendix A, GDC 4

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

  Statement of Compliance: The BWRX-300 containment and CIVs design features, including the ICS, RPV isolation valve assemblies, PCCS, CIVs, containment structure, containment penetrations, piping, and instrumentation lines, are to be designed to effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents, and will consider the dynamic effects of jet loads, pipe whipping, postulated high-energy pipe breaks, missiles and discharging fluids. In addition, the dynamic effects of postulated pipe breaks are to be evaluated in the BWRX-300 design. As described in this LTR, the BWRX-300 design requirements include applying the design criteria from NUREG-0800, SRP, BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Part B, Items 1(ii)(1)(d) and (e), and Items 1(ii)(2) through (7) to eliminate postulating breaks and cracks in those portions of piping from containment wall to the outboard CIVs. Breaks and cracks in those portions of piping from the RPV isolation valves that function as the inboard CIVs to the containment wall remain postulated to occur, and the dynamic effects of those postulated pipe breaks are to be evaluated in the BWRX-300 design. Each RPV isolation valve assembly is connected directly to the reactor vessel using bolted flange connections classified as break exclusion areas. For these bolted flange connections, details of the threaded fastener design, leakage detection systems design, and inservice inspection requirements, demonstrate that the probability of gross rupture is extremely low. For piping connected to the RPV isolation valve assemblies, extending to the containment wall, the BWRX-300 design requirements include identifying postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, Part B, Item 1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2). The dynamic effects of postulated breaks and cracks in those portions of the piping beyond and including the outboard CIVs will be evaluated in future licensing activities. Internal containment flooding is to be evaluated during future licensing activities.

Valve qualification, such as compliance with ASME Standard QME-1-2007 (or later edition) as endorsed by NRC Regulatory Guide (RG) 1.100, will be addressed in the detailed design and the procurement process of the valves, and will be specified during future licensing activities.

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

5.1.8 10 CFR 50 Appendix A, GDC 5

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 5, Sharing of structures, systems and components requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

  Statement of Compliance: The BWRX-300 design does not include sharing of SSCs important to safety among each unit at multi-unit sites.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 5.

5.1.9 10 CFR 50 Appendix A, GDC 13

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 13, Instrumentation and control, requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the RCPB, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

  Statement of Compliance: BWRX-300 instrumentation and controls are to be provided to monitor variables and systems important to the containment and its associated systems over their anticipated ranges for normal operation for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety. These instrumentation and control systems will be described during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 13.

5.1.10 10 CFR 50 Appendix A, GDC 16

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 16, Containment design, requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

  Statement of Compliance: A leak-tight steel or reinforced concrete PCV or a combination of steel and reinforced concrete PCV encloses the RPV, including the RCPB and other branch connections for the NBS, and includes containment penetrations with essentially leak-tight isolation design features including CIVs, blind flanges, hatches, and electrical penetrations. A steel head encloses the opening in the top of the PCV for servicing and refueling the RPV. The major piping systems (main steam, feedwater, ICS, and other miscellaneous systems) are located in the upper PCV region. The lower PCV region encloses the lower portion of the RPV and encloses the cooling system ducts, FMCRDs) and other miscellaneous systems as well as providing maintenance space below the RPV. 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 analyses to demonstrate compliance will be provided during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 16.

5.1.11 10 CFR 50 Appendix A, GDC 38

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 38, Containment heat removal, requires that a system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels. 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, assuming a single failure.

  Statement of Compliance: Containment peak pressure and temperature is limited by condensation on containment walls and RPV heat removal by the ICS and containment heat removal by the PCCS by natural convection and condensation. The PCCS is to be shown to reduce containment peak pressure rapidly for a large break LOCA, which is the limiting BWRX-300 DBA. Heat is rejected to the reactor cavity pool above 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 safety analysis assumes that the small breaks [[

]] In addition, the operation of the ICS does not require offsite electric power system operation, and only requires one-time actuation using onsite Class 1E battery-backed DC power.

[[

]] For RPV isolation and SBO events, containment pressure and temperature are limited by condensation on containment walls and containment heat removal by the PCCS, and by RPV decay heat removal by the ICS. The analyses to demonstrate compliance will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 38.

5.1.12 10 CFR 50 Appendix A, GDC 39

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 39, Inspection of containment heat removal system, requires that the containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

  Statement of Compliance: The components of the PCCS within containment to remove heat during a large break LOCA, are to be designed, fabricated, erected, and tested in accordance with ASME Code Section III, Class MC and Section XI, IWE requirements for design accessibility of welds 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 and GDC 1. In addition, means are to be provided to detect and identify the location of the source of containment leakage, including the CIVs, PCCS, non-essential and closed systems, and components of the ICS and RPV isolation valves, for components of the RCPB.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 39.

5.1.13 10 CFR 50 Appendix A, GDC 40

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 40, Testing of containment heat removal system, requires that the containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Statement of Compliance: Containment design is based upon consideration of a full spectrum of postulated accidents that would result in the release of reactor coolant to the containment. These accidents are evaluated using TRACG code as boundary condition to GOTHIC to calculate containment response. These accidents include liquid, steam and partial (both steam and liquid) breaks. The evaluation of the containment design 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. The GOTHIC computer methodology for measuring containment response is provided in LTR NEDC-33922P BWRX-300 Containment Evaluation Method [Reference 6.5]. The analyses to demonstrate compliance will be provided during future licensing activities. The BWRX-300 containment structural 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. The maximum expected external pressure containment structural evaluation will demonstrate compliance to 10 CFR 50, Appendix A, GDC 38 and 50 and be provided in future licensing activities.

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

5.1.18 10 CFR 50 Appendix A, GDC 51

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 51, Fracture prevention of containment pressure boundary requires that the reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

  Statement of Compliance: A leaktight containment vessel encloses the RPV, the RCPB, and other branch connections for the reactor primary coolant system, including containment penetrationand isolation devices. The containment vessel is a reinforced concrete and steel cylindrical structure with a leaktight steel liner providing the primary containment boundary. The containment vessel structure consists of the top containment slab with a reactor building pool above, cylindrical containment wall, containment floor slab, RPV pedestal, and the basement. A steel head encloses the opening in the top of the containment vessel for servicing and refueling the RPV. The containment encloses the RPV, with the major piping (main steam, feedwater, ICS, PCCS, RPVs, CIVs and other miscellaneous systems) located in the upper containment region. The lower containment encloses the lower portion of the RPV and encloses the cooling sytem ducts, FMCRDs, and other miscellaneous systems as well as providing maintenance space below the RPV.

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Statement of Compliance: The BWRX-300 CIVs 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 are provided with two normally closed outside containment manual CIVs. 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 is provided with normally closed inside and outside containment automatic CIVs. The process gas and radiation monitoring system is a closed system outside containment, and is provided with normally open outside containment automatic CIVs because it is an essential system following beyond design basis events and severe accidents. The floor drain sump line is provided with two normally closed outside containment automatic CIVs, because it is not practicable to include an inside containment automatic CIV 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.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 56.

5.1.24 10 CFR 50 Appendix A, GDC 57

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 57, Closed system isolation valves, requires that each line that penetrates primary reactor containment and is neither part of the RCPB nor connected directly to the containment atmosphere shall have at least one CIV which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

  Statement of Compliance: The BWRX-300 closed system CIVs 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 pneumatic nitrogen or air system and the quench tank supply system are provided with either normally open or normally closed inside and outside containment automatic CIVs. The service and breathing

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

5.2 Regulatory Guides

5.2.1 Regulatory Guide 1.7

Regulatory Guide (RG) 1.7, Control of Combustible Gas Concentrations in Containment, Rev. 3, describes methods acceptable to the NRC Staff for implementing the regulatory requirements of 10 CFR 50.44 for reactors subject to the provisions of Sections 50.44(b) or 50.44(c) with regard to control of combustible gases generated by beyond-design-basis accident that could be a risk-significant threat to containment integrity. For applicants and holders of a water-cooled reactor CP or OL under 10 CFR 50, and all applicants for a light-water reactor design approval or design certification, or combined license under 10 CFR Part 52 that are docketed after October 16, 2003, containments must have an inerted atmosphere or limit combustible gas concentrations in containment during and following an accident that releases an equivalent of combustible gas as would be generated from a 100% fuel-clad coolant reaction, uniformly distributed, to less than 10% (by volume) and must maintain containment structural integrity.

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. Compliance with the requirements of 10 CFR 50.44(c)(1) and 10 CFR 50.44(c)(3) for beyond design basis events and severe accidents are addressed in LTR NEDC-33921P, BWRX-300 Severe Accident Management [Reference 6.9].

Therefore, the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.7.

5.2.2 Regulatory Guide 1.11

RG 1.11, Instrument Lines Penetrating the Primary Reactor Containment, Rev. 1, describes methods acceptable to the NRC Staff for use in establishing that a plant's principal design criteria GDC 55 and GDC 56 require, in part, that each line that penetrates the primary reactor containment and that is part of the RCPB or connects directly to the containment atmosphere has at least one locked, closed isolation valve or one automatic isolation valve inside containment, and at least one locked, closed isolation valve or one automatic isolation valve outside containment (a simple check valve may not be used as the automatic isolation valve outside containment) "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."

Instrument lines that penetrates the primary reactor containment and that is part of the RCPB or that penetrates the primary reactor containment and connects 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 leak-tight integrity.

BWRX-300 instrument lines penetrating primary reactor containment that are part of the RCPB or penetrate the primary reactor containment and connects directly to the containment atmosphere comply with Regulatory Position C.3. by providing EFCVs, and also comply with the requirements of GDC 55 and GDC 56.

Each line is provided with a self-actuated EFCV 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 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.

Therefore, the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.11.

5.2.3 Regulatory Guide 1.84

RG 1.84. Design, Fabrication and Materials Code Case Acceptability, ASME Section III, Rev. 38, describes the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section III, “Rules for Construction of Nuclear Power Plant Components” Code Cases that the U.S. NRC has approved for use as voluntary alternatives to the mandatory ASME BPV Code provisions that are incorporated by reference into Title 10 of the Code of Federal Regulations (10 CFR), Part 50, “Domestic Licensing of Production and Utilization Facilities.” This RG applies to reactor licensees subject to 10 CFR Part 50, Section 50.55a, “Codes and standards”.

The BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.84, Rev. 38, and using the guidance conformance to RG 1.84, Rev. 33, as described in ESBWR 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. Code Case N-782 endorses the use of the Edition and Addenda of ASME Boiler and Pressure Vessel 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 this Code case will be provided in the BWRX-300 Preliminary Safety Analysis Report (PSAR) or future licensing activities.

5.2.4 Regulatory Guide 1.141

RG 1.141, Containment Isolation Provisions for Fluid Systems, Rev. 1, describes methods acceptable to the NRC Staff for use in implementing the regulatory requirements of GDC 55, Reactor coolant pressure boundary penetrating containment, GDC 56, Primary containment isolation, and GDC 57, Closed system isolation valves, with regard to establishing piping systems that penetrate the primary reactor containment be provided with isolation capabilities that reflect the importance to safety of isolating these piping systems.

The requirements and recommendations for the containment isolation of fluid systems that penetrate the primary containment of light-water-cooled reactors, as specified in ANSI N271-1976, are generally acceptable and provide an adequate basis for use.

Sections 2.2.8, 5.1.22, 5.1.23, and 5.1.24 of this LTR describes how the design of the BWRX-300 CIVs complies with the requirements of GDC 55, GDC 56, and GDC 57. Compliance with the requirements of 10 CFR 50.55a is described in Section 5.1.3.

Therefore, the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.141.

5.2.5 Regulatory Guide 1.147

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

Section 4.1.3 of LTR 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 B&PV Code Section XI, Division 1, specifically apply during construction and operation activities of nuclear power plants for performance of inservice inspection activities. The GEH design process and associated administrative controls considers operating plant compliance to RG 1.147 guidance in performing examinations, inspections and tests of installed systems and components, and are 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 B&PV Section XI Code Cases endorsed 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.

5.2.6 Regulatory Guide 1.155

RG 1.155, Rev. 0, Station Blackout, describes methods acceptable to the NRC for complying with 10 CFR 50.63, Loss of All Alternating Current Power, that requires nuclear power plants be capable of coping with an SBO for specified duration, so that SSCs important to safety continue to function. "Station blackout" refers to the complete loss of alternating current electric power to the essential and nonessential switchgear buses concurrent with turbine trip and failure of the onsite emergency ac power system, but not the loss of available ac power to buses fed by station batteries through inverters or loss of power from "alternate ac sources". 10 CFR 50.63 requires all licensees and applicants to assess the capability of their plants to maintain adequate core cooling and appropriate containment integrity during an SBO and to have procedures to cope with such an event. This guide further presents a method acceptable to the NRC for determining the specified duration for which a plant should be able to withstand an SBO in accordance with these requirements.

The BWRX-300 is designed to safely shut down without ac power. Safety-related CIV position indication and closure are provided by safety-grade control power, closure and position indication in case of SBO.

Therefore, the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.155.

5.2.7 Regulatory Guide 1.163

RG 1.163, Performance-Based Containment Leak Rate Test, Rev. 0, describes acceptable cost-effective methods, including setting test intervals, for implementing the safety objectives for performing containment leak testing in order to meet the requirements of 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." This regulatory guide approves an industry guideline that describes in detail a performance-based leak-test program, leakage-rate test methods, procedures, and analyses; the NRC Staff has determined this industry guideline to be an acceptable means of demonstrating compliance with the requirements of 10 CFR 50, Appendix J.

The BWRX-300 design is to include a containment leak rate testing program that addresses containment integrated leakage rate (Type A tests), containment penetration leakage tests (Type B tests), and CIV leakage rates (Type C tests) and complies with 10 CFR 50, Appendix J, Option A or Option B as per RG 1.163 and GDC 52, GDC 53, and GDC 54. The leakage rate testing capability is consistent with the testing requirements of ANS-56.8. Type A, B, and C tests are performed prior to operations and periodically thereafter to assure that 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.

Therefore, the BWRX-300 design conforms to the guidance, including the regulatory positions of RG 1.163.

5.2.8 Regulatory Guide 1.192

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

Section 4.1.3 of LTR 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 IST activities. GEH design process and associated administrative controls considers operating plant compliance to RG 1.192 guidance in performing examinations, inspections and tests of installed systems and components, and are 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.

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.

5.2.9 Regulatory Guide 1.203

RG 1.203, Transient and Accident Analysis Methods, Rev. 0, describes a process that the NRC Staff considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. An additional benefit is that evaluation models that are developed using these guidelines will provide a more reliable framework for risk-informed regulation and a basis for estimating the uncertainty in understanding transient and accident behavior.

The Regulatory Position section describes a multi-step process for developing and assessing evaluation models, and provides guidance on related subjects, such as quality assurance, documentation, general purpose codes, and a graded approach to the process. The Implementation section then specifies the target audience for whom this guide is intended, as well as the extent to which this guide applies, and the Regulatory Analysis section presents the NRC Staff related rationale and conclusion. For convenience, this guide also includes definitions of terms that are used herein. Finally, Appendix A provides additional information important to Emergency Core Cooling System (ECCS) analysis, and Appendix B presents an example of the graded application of the evaluation model development and assessment process (EMDAP) for different analysis modification scenarios.

Section 3.4 of this LTR describes how the GOTHIC methodology code utilizes the Code Scaling, Applicability and Uncertainty in NUREG/CR-5249 and RG 1.203, and the Phenomenon Identification and Ranking Table graded approach of RG 1.203 for analyzing BWRX-300 containment response to transient and accident behavior.

Therefore, the BWRX-300 design conforms to the guidance, including regulatory positions of RG 1.203.

5.3 NUREG-0800 Standard Review Plan Guidance

5.3.1 Standard Review Plan 3.6.2

Standard Review Plan (SRP) 3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Rev. 3 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 GDC 4 Environmental and Dynamic Effects Design Bases be considered in the design structures, systems and components. This SRP 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. The guidance provides specific areas for review:

1. Defining break and crack locations and configurations
2. Analytical methods to define forcing functions, including jet thrust reaction at the postulated pipe break or crack location and jet impingement loadings on adjacent safety-related SSCs
3. The dynamic analysis methods used to verify the integrity and operability of mechanical components, component supports, and piping systems, including restraints and other protective devices under postulated pipe rupture loads
4. The implementation of criteria used in defining pipe break and crack locations and configurations
5. The criteria for dealing with special features such as augmented inservice inspection programs
6. The acceptability of the analysis results, including jet thrust and impingement forcing functions and pipe-whip dynamic effects
7. The design adequacy of SSCs to ensure that the intended design functions will not be impaired to an unacceptable level of integrity or operability as a result of pipe whip or jet impingement loadings.

The BWRX-300 containment isolation system SSCs will conform to the guidance of the SRP, as well as meeting the requirements of GDC 4. The design of the containment structure, component arrangement, pipe runs, pipe whip restraints, and compartmentalization are 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 are engineered to preclude damage based upon break evaluation. A complete description of compliance to the SRP and associated branch technical positions, 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.

5.3.2 Standard Review Plan 3.9.6

Standard Review Plan (SRP) 3.9.6, Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints, Rev. 4, states that the areas of review include the functional design and qualification provisions and IST programs for safety-related pumps, valves, and dynamic restraints (snubbers) designated as Class 1, 2, or 3 under ASME B&PV Code Section III. The review includes other pumps, valves and dynamic restraints not categorized as ASME BPV Code Class 1. 2 or 3 that have safety-related function. Conformance with the specific guidance in Subsection II of this SRP section will provide reasonable assurance that the functional design and qualification of pumps, valves and dynamic restraints within the scope of this SRP section and their associated IST programs satisfy the applicable requirements of Section 50.55a, "Codes and Standards," of Title 10 of the Code of Federal Regulations, particularly the IST program requirements of the ASME Code for Operation and Maintenance of Nuclear

Power Plants (OM Code) 4; General Design Criterion (GDC) 1, "Quality Standards and Records," GDC 2, "Design Bases for Protection against Natural Phenomena," GDC 4, "Environmental and Dynamic Effects Design Bases," GDC 14, "Reactor Coolant Pressure Boundary," GDC 15, "Reactor Coolant System Design," GDC 37, "Testing of Emergency Core Cooling System," GDC 40, "Testing of Containment Heat Removal System," GDC 43, "Testing of Containment Atmosphere Cleanup Systems," GDC 46, "Testing of Cooling Water System," and GDC 54, "Systems Penetrating Containment," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities;" Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50; 10 CFR 52.47(b)(1), 10 CFR 52.79(a)(11), and 10 CFR 52.80(a).

The containment isolation valves are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, including any application for a construction permit under 10 CFR 50 or design certification application under 10 CFR 52. The requirements of 10 CFR 50.55a, are to be implemented during detailed design of the safety-related components of containment isolation. Therefore, the existing SRP provides adequate guidance to use during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

5.3.3 Standard Review Plan 6.2.1

SRP 6.2.1, Containment Functional Design, Rev. 3, states that the areas of review include the containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant, steam line, or feedwater line break accidents. The containment structure must also maintain functional integrity in the long term following a postulated accident; i.e., it must remain a low leakage barrier against the release of fission products for as long as postulated accident conditions require.

The design and sizing of containment systems are largely based on the pressure and temperature conditions which result from release of the reactor coolant in the event of a LOCA. The containment design basis includes the effects of stored energy in the reactor coolant system, decay energy, and energy from other sources such as the secondary system, and metal-water reactions including the recombination of hydrogen and oxygen. The containment system is not required to be a complete and independent safeguard against a LOCA by itself, but functions to contain any fission products released while the ECCS cools the reactor core. The evaluation of a containment functional design includes calculation of the various effects associated with the postulated rupture in the primary or secondary coolant system piping. The subsequent thermodynamic effects in the containment resulting from the release of the coolant mass and energy are determined from a solution of the incremental space and time-dependent energy, mass, and momentum conservation equations. The basic functional design requirements for containment are given in GDC 4, GDC 16, GDC 50, and 10 CFR 50, Appendix K.

The various containment types and aspects to be reviewed under this SRP section have been separated and assigned to a set of other SRP sections. The BWRX-300 containment design is affected by the guidance provided in SRP 6.2.1.1.A, SRP 6.2.1.1.C, SRP 6.2.1.2, SRP 6.2.1.3, SRP 6.2.1.4, and SRP 6.2.1.5. The following SRPs are not applicable to the BWRX-300 design and discussed specifically in subsequent LTR sections:

SRP 6.2.1.1.C Pressure-Suppression Type BWR Containments – the BWRX-300 does not utilize a pressure-suppression pool for maintaining containment pressure and temperature from the dynamic effects of LOCA.

SRP 6.2.1.2 Subcompartment Analysis – the BWRX-300 does not have subcompartments in the design that contain large bore high energy lines.

SRP 6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures – the BWRX-300 design does not utilize secondary system piping.

SRP 6.2.1.5 Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies – the BWRX-300 does not utilize emergency core cooling for maintaining containing pressure during design basis events. Containment pressure is maintained by the PCCS for AOOs, IEs and DBAs.

The design features of the BWRX-300 containment include:

- Underground (subterranean) steel or reinforced concrete PCV
- Dry containment with no suppression pool
- Nitrogen-inerted containment
- Passive containment heat removal for PCCS for design basis events; fan coolers for normal operations
- No subcompartments with large bore high energy lines
- ICS pools and reactor cavity pool for PCCS located above containment
- Fewer penetrations

Specific discussions under Section I, Areas of Review, are addressed in meeting the intent of the affected individual SRP sections previously delineated.

Specific discussions under Section II, Acceptance Criteria, for generic acceptance criteria that are affected are discussed in the affected individual SRP sections previously delineated.

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 Section 2.0 of this LTR. Therefore, GEH recommends that the existing SRP may be used during future review of a BWRX-300 10 CFR 52 DCA if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

5.3.4 Standard Review Plan 6.2.1.1.A

SRP 6.2.1.1.A, PWR Dry Containments, Including Subatmospheric Containments, Rev. 3, states that the areas of review include: (1) the temperature and pressure conditions in the containment due to a spectrum (including break size and location) of postulated loss-of-coolant accidents (i.e., reactor coolant system pipe breaks) and secondary system steam and feedwater line breaks; (2) the maximum expected external pressure to which the containment may be subjected; (3) the minimum containment pressure that is used in analyses of ECCS capability; (4) the effectiveness of static and active heat removal mechanisms; (5) the pressure conditions within subcompartments that act on system components and supports due to high energy line breaks; and (6) the range and accuracy of instrumentation that is provided to monitor and record containment conditions during and following an accident.

The BWRX-300 containment is nitrogen-inerted with no suppression pool to mitigate the dynamic effects of DBAs. Therefore, SRP 6.2.1.1.C no longer applies to this GEH design. As a result, SRP 6.2.1.1.A was selected to use as guidance inasmuch as the guidance and acceptance criteria described within reflect the BWRX-300 design. It should be noted that while SRP 6.2.1.1.A better reflects the design of the BWRX-300, portions of this guidance document are also 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; (3) there are no secondary systems utilized in the BWRX-300 design. The design requirements for the PCCS to reject heat to the reactor cavity pool above containment during DBAs is described in Section 2.2.8.

Section 3.0 discusses the TRACG and GOTHIC computer code methodologies utilized to analyze mass and energy release from the RPV that provide boundary conditions for the GOTHIC code to analyze the containment response for a spectrum of break sizes and locations for postulated loss of coolant accidents. The GOTHIC computer methodology for measuring containment response is provided in LTR NEDC-33922P BWRX-300 Containment Evaluation Method [Reference 6.5]. The containment performance acceptance criteria are discussed in Section 4.0. All instrumentation is to be provided with accuracy and ranges for the most severe accident scenario and record containment conditions during and following an accident. Requirements for beyond design basis events and severe accidents are to be addressed in LTR NEDC-33921P, BWRX-300 Severe Accident Management [Reference 6.9].

Specific discussions under Section I, Areas of Review, are addressed in meeting the intent of the affected individual SRP sections previously delineated.

Specific discussions under Section II, Acceptance Criteria, for generic acceptance criteria that are affected are discussed in the affected individual SRP sections previously delineated.

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

5.3.5 Standard Review Plan 6.2.1.1.C

SRP 6.2.1.1.C, Pressure-Suppression Type BWR Containments, Rev. 7, provides guidance in evaluating the temperature and pressure condition effects in the drywell and wetwell of BWR containments incorporating a suppression pool.

The BWRX-300 design does not employ the use of a drywell and wetwell incorporating a suppression pool. Therefore, the acceptance criteria associated with these guidelines are not applicable for the BWRX-300 design.

5.3.8 Standard Review Plan 6.2.1.4

SRP 6.2.1.4, Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures, Rev. 2, provides guidance for the review of the mass and energy release for secondary system pipe ruptures to evaluate the containment and subcompartment functional design in order to comply with GDC 50 for postulated pressurized-water reactor PWR secondary system pipe ruptures to ensure the reactor containment structure, including access openings, penetrations, and the containment heat removal system can withstand the calculated pressure and temperature conditions resulting from any LOCA.

The BWRX-300 design does not employ the use of any secondary systems for feedwater or steam production. Containment temperature and pressure are removed by the PCCS for all postulated DBAs. See Section 2.2.8 for complete discussion of PCCS heat removal capability. Therefore, the acceptance criteria associated with these guidelines are not applicable for the BWRX-300 design.

5.3.9 Standard Review Plan 6.2.1.5

SRP 6.2.1.5, Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies, Rev. 3, provides guidance for compliance to 10 CFR 50.46 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 during the time until the core is reflooded.

The BWRX-300 design includes the use of RPV isolation valves and the ICS to perform the ECCS design functions as described in LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 6.2]. For large break LOCAs, containment pressure does not affect the performance of the ECCS design functions as the [[

]] Therefore, the acceptance criteria associated with these guidelines are not applicable for the BWRX-300 design.

5.3.10 Standard Review Plan 6.2.2

SRP 6.2.2, Containment Heat Removal Systems, Rev. 5, provides guidance for the review of containment heat removal under post-accident conditions to ensure conformance with the requirements of GDC 38, GDC 39, GDC 40, and 10 CFR 50.46(b)(5).

Specific Areas of Review under Section I include: 1. the consequences of single component malfunctions; 2. analyses of Net Positive Suction Head (NPSH) to the ECCS and containment heat removal pumps; 3. the analyses of the heat removal capability of the spray water system; 4. the analyses of the heat removal of the Residual Heat Removal (RHR) and fan cooler heat exchangers; 5. the potential for surface fouling and flow blockage of the fan cooler, recirculation , and RHR heat exchangers and the effect on heat exchanger performance; 6. the design provisions and

5.4 Generic Issues

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

5.4.1 NUREG-0737

NUREG-0737, Clarification of TMI Action Plan Requirements, November 1980, contains requirements approved for implementation by the NRC Commissioners as a result of the accident at TMI Unit 2. The NRC Commission subsequently recommended that certain of these requirements be added to the 10 CFR 50 regulations, which were subsequently implemented in 10 CFR 50.34(f). Compliance with the items that are related to containment performance are discussed in Section 5.1.1. Compliance with the items that are related to RPV isolation and the mitigation of the effects of a LOCA are discussed in Section 4.1.1 of LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 6.2].

5.5 Operational Experience and Generic Communications

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

5.5.1 Generic Letter 83-02

Generic Letter 83-02, NUREG-0737 Technical Specifications, dated January 10, 1983, contains a request for information for the current BWR licensees regarding NUREG-0737 items for which technical specifications are required, including guidance on the scope of a specification which the 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.

5.5.2 Generic Letter 95-07

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

6.0 REFERENCES

6.1 26A6642AP Revision 10, "ESBWR Design Control Document, Tier 2, Chapter 4 Reactor," GE Hitachi Nuclear Energy, April 2014

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

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

6.4 26A6642AT Revision 10, "ESBWR Design Control Document, Tier 2, Chapter 6 Engineered Safety Features," GE Hitachi Nuclear Energy, April 2014

6.5 NEDC-33922P, "BWRX-300 Containment Evaluation Method"

6.6 NEDC-33083P-A Revision 1, "TRACG Application for ESBWR," September 2010

6.7 NEA/CSNI/R3(2014), "Containment Code Validation Matrix," May 2014

6.8 SMSAB-02-02, "An Assessment of CONTAIN 2.0: A Focus on Containment Thermal Hydraulics (Including Hydrogen Distributions)," July 2002

6.9 NEDC-33921P, "BWRX-300 Severe Accident Management"