ENCLOSURE 2

M200121

Licensing Topical Report

NEDO-33911, Revision 0 Supplement 1,
BWRX-300 Containment Performance

Non-Proprietary Information

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GE Hitachi Nuclear Energy

NEDO-33911
Revision 0 Supplement 1
September 2020

Non-Proprietary Information

Licensing Topical Report

BWRX-300 Containment Performance

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

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The design, engineering, and other information contained in this document is furnished for the purpose of obtaining Nuclear Regulatory Commission (NRC) review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, GEH makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

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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 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 change bars for Section 2.1.2, last paragraph; 2.2, 2^{nd} paragraph; 2.2.2, 4^{th} bullet; 2.2.7.3; 3.1, 4^{th}, 5^{th} and 6^{th} paragraphs; 5.1.4; 5.1.5; 5.1.6; 5.1.7; 5.1.24; 5.3.3; 5.3.4, 2^{nd} paragraph; 5.3.9, 2^{nd} 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.

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	Economically 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
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 Economically 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 for requesting a Construction Permit (CP) and Operating License (OL) under 10 CFR 50 or a Combined Operating License (COL) under 10 CFR 52.

2.0 TECHNICAL EVALUATION OF 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.

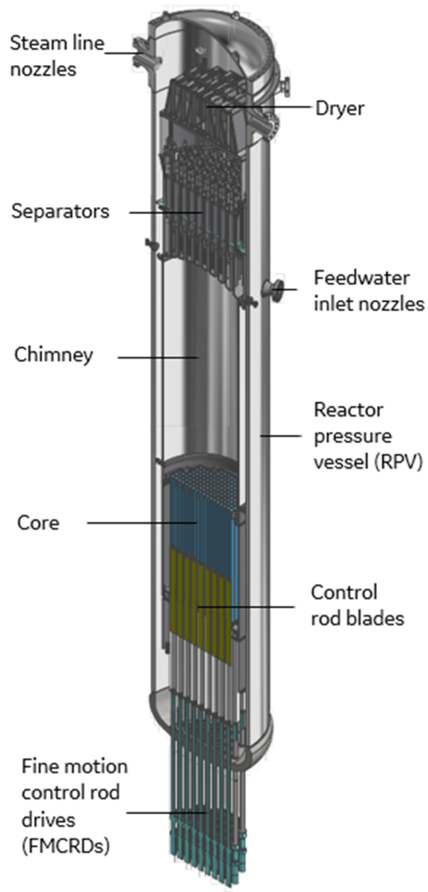

Figure 2-1: BWRX-300 Reactor Pressure Vessel and Internals

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 in 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 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 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 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 [Reference 6.9], 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

[[

]]

Figure 2-6: CIVs Connected to RPV Boundary

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.

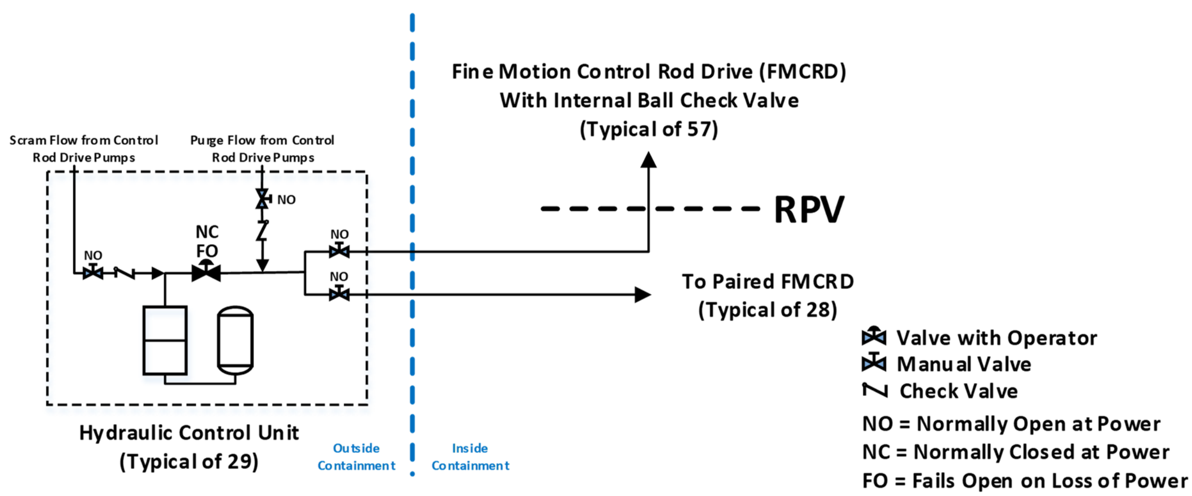

Figure 2-8: FMCRD CIVs Connected to RPV Boundary

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 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.

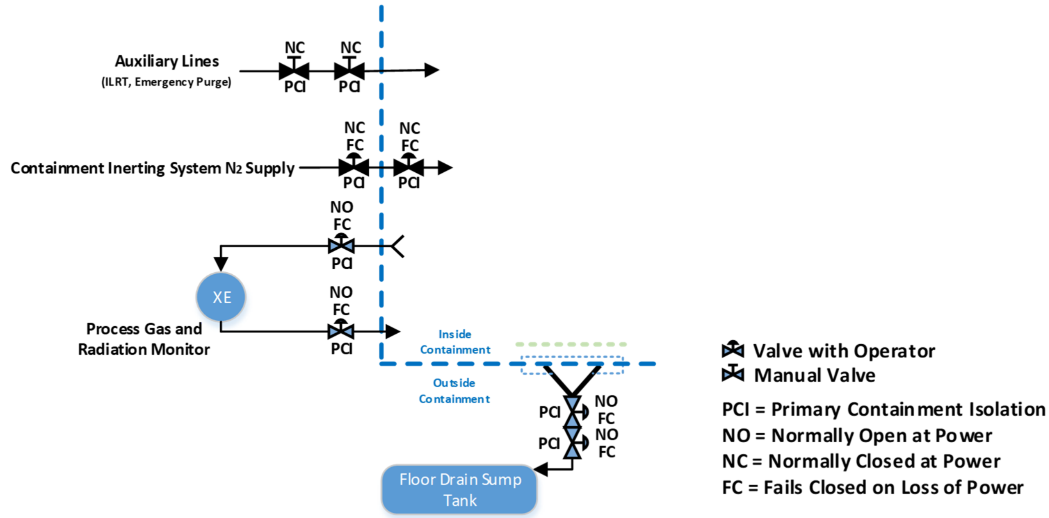

Figure 2-9: CIVs Connected to Containment Atmosphere

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.

[[

]]

Figure: 2-10 CIVs Connected to Closed Systems

2.2.8 Passive Containment Cooling System (PCCS)

The PCCS is based upon proven concepts and [[

]]

2.2.8.1 PCCS Design Functions

The PCCS transfers heat [[

]]

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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[[

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

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[[

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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.

3.5 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].

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"