ENCLOSURE 2

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Licensing Topical Report

NEDO-33910, Revision 0 Supplement 2, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection

Non-Proprietary Information

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GE Hitachi Nuclear Energy

NEDO-33910
Revision 0 Supplement 2
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Non Proprietary Information

Licensing Topical Report

BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection

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This is a non-proprietary version of the GE Hitachi Nuclear Energy (GEH) document NEDC-33910P Revision 0 Supplement 2, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

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REVISION SUMMARY

Revision Number	Description of Change
0	Initial Issue
Supplement 1	Revised to incorporate the following responses to NRC Requests for Additional Information (eRAIs): • NRC eRAI 9730, Question 03.09.06-1, revised Section 1.1 and added new Sections 4.1.5 and 4.1.13 to address compliance with 10 CFR 50, Appendix A, GDC 2 and GDC 37, and to address addition of Generic Letter 95-07 as new Section 4.5.2 • NRC eRAI 9730, Question 03.09.06-1, supplemental response revised new Section 4.1.5 to address [[ ]]. • NRC eRAI 9730, Question 03.09.06-3, replaced the use of terms such as "consideration" and "considered" with appropriate terms in Sections 2.4, 2.4.1, 2.5, 2.6, 2.7, renumbered 4.1.6, and renumbered 4.3.2. • NRC eRAI 9730, Question 03.09.06-4, added the design requirements for the use of positive mechanical means in the design of the valve actuators to maintain these valves in their required post-accident valve positions in Sections 2.5 and 3.1.2. • NRC eRAI 9730, Question 03.09.06-5, revised Section 4.1.1 to describe requirements for 10 CFR 50.34(f)(2)(x) as not technically relevant rather than not required, but to also indicate that the [[ ]] will be qualified by testing to perform their safety-related design function for fluid conditions expected under operating conditions, transients and accidents including ATWS events. • NRC eRAI 9730, Questions 03.09.06-6, 03.09.06-7, and 03.09.06-14, clarified statements that no alternative approach, exception, or exemption from certain regulatory requirements is required providing a commitment to meeting the applicable regulatory requirements during detailed design in Sections 2.7, 4.1.2, 4.1.3, 4.1.4, renumbered 4.1.6, renumbered 4.1.7, renumbered 4.1.8, renumbered 4.1.9, renumbered 4.1.10, renumbered 4.1.12, 4.2.1, 4.2.2, 4.2.3, and 4.2.4.

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

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

Acronyms and Abbreviations

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

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

1.0 INTRODUCTION

1.1 Purpose

The purpose of this report is to provide the design requirements, acceptance criteria, and regulatory basis for the BWRX-300 Reactor Pressure Vessel (RPV) isolation and overpressure protection design functions, specifically for the following areas:

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

1.2 Scope

The scope of this report includes the following:

- A technical evaluation of the BWRX-300 RPV isolation and overpressure protection design features and design functions, acceptance criteria, regulatory bases, and references to existing proven design concepts based upon previous Boiling Water Reactor (BWR) designs, including the Advanced Boiling Water Reactor (ABWR) and Economically Simplified Boiling Water Reactor (ESBWR).
- A regulatory review of the BWRX-300 RPV isolation and overpressure protection design features and design functions to describe compliance with regulatory requirements and to describe the bases for any exemptions to regulatory requirements or approaches to regulatory guidance that may be referenced during future licensing activities either by GEH in support of a 10 CFR 52 Design Certification Application (DCA) or by a license applicant for requesting a Construction Permit (CP) and Operating License (OL) under 10 CFR 50 or a Combined Operating License (COL) under 10 CFR 52.

2.0 TECHNICAL EVALUATION OF RPV ISOLATION

2.1 General Introduction

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

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

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

2.1.1 Reactor Pressure Vessel

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

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

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

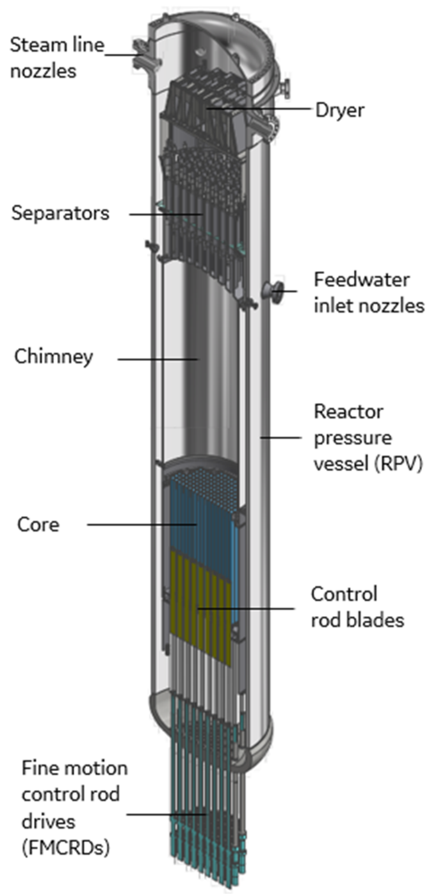

Figure 2-1: BWRX-300 Reactor Pressure Vessel and Internals

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

The major reactor internal components include:

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

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

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

2.1.2 Isolation Condenser System

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

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

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

[[

]]

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

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

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

2.2 General Overview of the Reactor Pressure Vessel Isolation Concept

[[

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[[

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

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

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

]]

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

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

]]

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

[[

]]

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

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

[[

]]

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

2.3 Reactor Pressure Vessel Design Requirements

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

2.4 Reactor Pressure Vessel Nozzle Design Requirements

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

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

]]

Design Requirements:

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

]]

2.4.1 Connection of Reactor Pressure Vessel Isolation Valves to Reactor Vessel

[[

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

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

[[

]] These [[

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

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

[[

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

[[

]]

[[

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

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

2.5 Reactor Pressure Vessel Isolation Valve Design Requirements

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

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

Design Requirement:

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

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

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

Design Requirements:

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

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

]]

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

Design Requirements:

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

  ]]

The ICS RPV isolation valves automatic isolation function uses logic and functionality similar to the ESBWR ICS containment isolation valves, which is described in the last two paragraphs of

ESBWR DCD Tier 2 Subsection 5.4.6.2.2 [Reference 5.2]. However, the BWRX-300 Instrumentation and Control (I&C) system has three divisions of safety-related I&C. [[

]]

2.6 Reactor Pressure Vessel Isolation Valve Actuator Design Requirements

Design Requirement:

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

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

Design Requirement:

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

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

2.7 Categories of Pipe Breaks

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

- [[
-]]

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

]]

Design Requirements:

- [[

-

]]

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

]]

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

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

Table 2-1: Pipe Break Categories

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

Small leaks in the RCPB that are [[
]] (e.g., leakage from flanges or cracks in piping or other components) do not exceed the capability of the nonsafety-related high-pressure CRD system used as normal reactor coolant

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

2.8 LOCA Acceptance Criteria

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

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

3.0 TECHNICAL EVALUATION OF OVERPRESSURE PROTECTION

3.1 General Overview of the Overpressure Protection Concept

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

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

3.1.1 Reactor Protection System Design Requirements

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

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

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

Design Requirements:

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

3.1.2 Isolation Condenser System Design Requirements

The BWRX-300 ICS is based on the ESBWR ICS design [Reference 5.2]. The ICS is designed as a safety-related system to remove decay heat passively and with a minimal loss of reactor coolant following reactor shutdown and isolation. [[

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

The ICS contains IC heat exchangers that condense steam on the tube side and transfer heat to the IC pool. The IC heat exchangers, connected by piping to the RPV, are placed at an elevation above the source of steam (RPV) and, when the steam is condensed, the condensate is returned to the RPV via a condensate return pipe.

The steam side connections between the RPV and the IC heat exchangers are normally open, and the condensate lines are normally closed. This allows the IC heat exchangers and drain piping to fill with condensate, which is maintained at a subcooled temperature by the IC pool water during normal reactor operation.

The ICS is placed into operation by opening condensate return valves and draining the condensate to the RPV, thus causing steam from the reactor to fill the tubes which transfer heat to the cooler IC pool water. [[
]] with valve actuators designed to maintain the valves in their open position by positive mechanical means.

Design Requirements:

- [[
-
-
-
  ]]

3.2 ASME Requirements for Overpressure Protection

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

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

b. The use of the reactor shutdown system

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

    1. The overpressure does not exceed 1.1 times design pressure for the expected system pressure transient condition
    2. The calculated stress intensity and other design limitations for Service Level C are not exceeded for the unexpected system excess pressure transient condition

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

4.0 REGULATORY EVALUATION

4.1 10 CFR 50 Regulations

4.1.1 10 CFR 50.34(f)

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

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

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

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

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

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

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

]]. The intent of this requirement is to minimize the potential for loss of reactor coolant through inadvertent operation of relief and safety valves, [[

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

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

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

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

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

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

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

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

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

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

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

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

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

Based on the above discussions, these requirements are not technically relevant to the BWRX-300. These statements of compliance may be used as the bases for this conclusion during future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant for requesting a CP and OL under 10 CFR 50 or a COL under 10 CFR 52.

4.1.2 10 CFR 50.46

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

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

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

  - The required ECCS design functions of the [[ ]] to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(4) as further described below. Another design function of the [[ ]] is for overpressure protection which is further required to maintain a coolable geometry to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria in 10 CFR 50.46(b)(4). The worst-case single failure affecting the [[ ]] does not prevent fulfillment of the required ECCS design functions. Because of the relatively large volume of reactor coolant above the reactor core during normal operation, there is no need for [[ ]] following the worst-case postulated LOCA assuming failure of [[ ]]. Following the worst-case postulated LOCA, the [[ ]] continues to provide long-term cooling to meet the BWRX-300 acceptance criteria in response to a LOCA which bound the acceptance criteria of 10 CFR 50.46(b)(5) and only requires operator action to [[ ]] after approximately seven days.

  - [[ ]] for any breaks

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

For the BWRX-300, [[

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

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

]] Following [[

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

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

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

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

4.1.3 10 CFR 50.55a

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

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

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

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

4.1.4 10 CFR 50 Appendix A, GDC 1

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

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

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

4.1.5 10 CFR 50 Appendix A, GDC 2

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

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

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

4.1.6 10 CFR 50 Appendix A, GDC 4

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

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

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

4.1.7 10 CFR 50 Appendix A, GDC 14

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

Statement of Compliance: [[

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

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

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

4.1.8 10 CFR 50 Appendix A, GDC 15

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

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

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

4.1.9 10 CFR 50 Appendix A, GDC 30

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

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

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

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

4.1.10 10 CFR 50 Appendix A, GDC 31

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

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

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

4.1.11 10 CFR 50 Appendix A, GDC 33

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

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

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

4.1.12 10 CFR 50 Appendix A, GDC 35

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

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

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

[[

]]

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

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

4.1.13 10 CFR 50 Appendix A, GDC 37

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

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

  ]].

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

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

4.2 Regulatory Guides

4.2.1 Regulatory Guide 1.26

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

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

4.2.2 Regulatory Guide 1.29

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

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

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

4.2.3 Regulatory Guide 1.45

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

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

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

4.2.4 Regulatory Guide 1.84

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

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

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

4.2.5 Regulatory Guide 1.147

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

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

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

4.2.6 Regulatory Guide 1.192

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

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

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

4.3 NUREG-0800 Standard Review Plan Guidance

4.3.1 Standard Review Plan 3.9.6

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

As described in Section 4.1, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 2, GDC 4, GDC 14, GDC 15, and GDC 37, and the requirements of 10 CFR 50.55a, during detailed design activities with specific requirements for the [[ ]] to be provided during future licensing activities. In addition, Section 4.1.3 describes that the BWRX-300 RPV isolation valves and overpressure protection design features are to be designed using the standards approved in 10 CFR 50.55a(a) in effect within six months of any license application, including any application for a construction permit under 10 CFR 50 or design certification application under 10 CFR 52. These requirements are to be implemented during detailed design of the safety-related components of the [[ ]]. Therefore, the existing SRP provides adequate guidance to use during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

4.3.2 Standard Review Plan 5.2.2

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

The design features of the BWRX-300 RCPB include the use of the RPS and [[ ]] for overpressure protection, [[ ]]. As this is a non-traditional approach to meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 15, GDC 30, and GDC 31, alternate guidance applicable to SRP 5.2.2 for the BWRX-300 is recommended to provide guidance to use during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

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

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

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

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

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

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

  This area of review [[ ]].

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

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

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

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

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

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

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

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

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

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

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

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

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

    ]].

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

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

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

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

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

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

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

    ]].

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

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

    These acceptance criteria should be revised to [[

    ]].

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

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

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

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

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

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

  ]].

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

  These review procedures should be revised to [[

  ]].

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

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

]].

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

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

  ]].

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

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

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

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

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

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

  These evaluation findings should be revised to [[

]].

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

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

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

4.3.3 Standard Review Plan 5.4.13

SRP 5.4.13, Isolation Condenser System (BWR), Rev. 0, states that the areas of review include the system design bases, design criteria, components, support systems, and instrumentation and controls employed in the system. The areas of review, review interfaces, acceptance criteria, review procedures, evaluation findings, and references are acceptable for use for the BWRX-300 based on the design description and design requirements discussed in Subsection 2.1.2 of this LTR. Therefore, the existing SRP provides adequate guidance to use during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

4.3.4 Standard Review Plan 6.3

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

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

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

The design functions of the [[

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

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

Although this is a non-traditional approach for the design of the ECCS for past LWRs, no active or passive injection of additional water inventory is required following the worst-case LOCA to meet these BWRX-300 acceptance criteria and to meet the requirements of 10 CFR 50 Appendix A, GDC 1, GDC 14, GDC 30, GDC 31, and GDC 35. Therefore, the second and third areas of review regarding ADS and passive ECCS injection are not applicable. Other than these areas of review, the review interfaces, acceptance criteria, review procedures, evaluation findings, and references are acceptable for use for the BWRX-300 recognizing that those discussions related to the ADS and active or passive ECCS injection are not applicable. Based on these factors, the existing SRP provides adequate guidance to use during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

4.3.5 Standard Review Plan 15.6.5

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

The design functions of the [[

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

]] to meet the requirements of 10 CFR 50 Appendix A, GDC 35, 10 CFR 50.46, and 10 CFR 50 Appendix K, and the applicable general design requirements discussed in SRP Section 6.3. Based on these factors, the existing SRP provides adequate guidance to use during future review of a BWRX-300 10 CFR 52 design certification application if pursued (as required by 10 CFR 52.47(a)(9)), or for future 10 CFR 50 license applications.

4.4 Generic Issues

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

4.4.1 NUREG-0737

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

4.5 Operational Experience and Generic Communications

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

4.5.1 Generic Letter 83-02

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

4.5.2 Generic Letter 95-07

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

5.0 REFERENCES

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

5.2 26A66412AR, Rev 10, "ESBWR Design Control Document, Tier 2, Chapter 5 Reactor Coolant System and Connected Systems", GE Hitachi Nuclear Energy, April 2014

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

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

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

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

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