ENCLOSURE 2

M200122

Licensing Topical Report

NEDO-33912, Revision 0 Supplement 1,
BWRX-300 Reactivity Control

Non-Proprietary Information

IMPORTANT NOTICE

This is a non-proprietary version of the BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection Licensing Topical Report, from which the proprietary information has been removed. The header of each page in this enclosure carries the notation "Non-Proprietary Information." Portions of the enclosure that have been removed are indicated by an open and closed bracket as shown here [[ ]].

GE Hitachi Nuclear Energy

NEDO-33912
Revision 0 Supplement 1
September 2020

Non-Proprietary Information

Licensing Topical Report

BWRX-300 Reactivity Control

PROPRIETARY INFORMATION NOTICE

This is a non-proprietary version of the GE Hitachi Nuclear Energy (GEH) document NEDC-33912P Revision 0 Supplement 1, 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

Please Read Carefully

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

TABLE OF CONTENTS

LIST OF FIGURES

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 9761, Question NONE-4, revised Section 4.1.11 for BWRX-300 compliance to 10 CFR 50, Appendix A, GDC 28, and to add new References 5.7 and 5.8 for ESBWR Request for Additional Information (RAI) 4.6-23 (ADAMS Accession Nos. ML081090147 (Public) and ML081090148 (Non-Public) and NEDE-33885P-A, Revision 1, Global Nuclear Fuels (GNF) CRDA Application Methodology, respectively. • NRC eRAI 9761, Question NONE-5, revised Section 4.1.9 for BWRX-300 compliance to the statement "independent reactivity control systems of different design principles" in 10 CFR 50, Appendix A, GDC 26. • Revised Section 5 in response to NRC eRAI 9761, Question NONE-4, to add references to ESBWR RAIs and GNF LTR NEDE-33885P-A. • Corrected wording in the Purpose section to read: Design requirements are specified … and the backup means to automatically or manually insert control rods to ensure reactor shutdown. • Editorial correction in last sentence of Section 4.3.1. Changed wording from "[[ ]]" to "[[ ]]". • Information regarding the FMCRDs has been reclassified as non-proprietary and is identified with change bars in Sections 2.2.2.1, 3.5, 3.7.1 and 4.1.1.

Acronyms and Abbreviations

Term	Definition
ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ALWR	Advanced Light-Water Reactor
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ASME	American Society of Mechanical Engineers
[[	]]
ATWS	Anticipated Transient Without Scram
B&PV	Boiler & Pressure Vessel
BL-DBA	Baseline Design Basis Analysis
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CDF	Core Damage Frequency
CN-DBA	Conservative Design Basis Analysis
COL	Combined Operating License
CP	Construction Permit
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DCA	Design Certification Application
DEC	Design Extension Condition
DL	Defense Line
ECCS	Emergency Core Cooling System
ESBWR	Economically Simplified Boiling Water Reactor
EX-DBA	Extended Design Basis Analysis
FMCRD	Fine Motion Control Rod Drive

Term	Definition
FSF	Fundamental Safety Function
GDC	General Design Criteria
GEH	GE Hitachi Nuclear Energy
HCU	Hydraulic Control Unit
HGNE	Hitachi-GE Nuclear Energy Ltd.
HVAC	Heating, Ventilation, and Air-conditioning
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICS	Isolation Condenser System
LOCA	Loss-of-Coolant Accident
LTR	Licensing Topical Report
MSIV	Main Steam Isolation Valve
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OL	Operating License
PDC	Principal Design Criterion
PIE	Postulated Initiating Event
PWR	Pressurized Water Reactor
RCPB	Reactor Coolant Pressure Boundary
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SAFDL	Specified Acceptable Fuel Design Limits
SLCS	Standby Liquid Control System
SMR	Small Modular Reactor
SRNM	Source Range Neutron Monitor
SRP	Standard Review Plan
SSC	Structure, System, and Component

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 reactivity control design functions, specifically for the following areas:

- Design requirements are specified for the Reactor Protection System (RPS) and the [[ ]] such that they satisfy the defense-in-depth (D-in-D) and diversity requirements to protect from common cause failure (CCF) of the RPS. Design requirements are also specified for other associated functions such as Alternate Rod Insertion (ARI) to ensure that the automatic hydraulic reactor scram will meet specified reliability requirements. The design of the RPS, [[ ]] and associated D-in-D features meet the requirements of 10 CFR 50.62 with justification provided for a [[ ]]. In addition, the requirements of 10 CFR 50 Appendix A, General Design Criteria (GDC) 12, GDC 20, GDC 21, GDC 22, GDC 23, GDC 24, GDC 25, GDC 26, GDC 28, and GDC 29 are met, with justification provided for a proposed exemption to the specific requirements of GDC 27 as proposed in Principal Design Criterion (PDC) 27.
- Design requirements are specified for the [[ ]] and the backup means to automatically or manually insert control rods to ensure reactor shutdown. [[

  ]] These design features meet the requirements of 10 CFR 50.62 with the [[ ]]. In addition, the requirements of 10 CFR 50 Appendix A, GDC 12 and GDC 26 are met, with justification provided for PDC 27.

1.2 Scope

The scope of this report includes the following:

- A technical description of the BWRX-300 RPS, [[ ]] and other D-in-D design features and design functions to ensure the capability to shutdown the reactor and provide for control of reactivity and reactor fuel thermal limits, including 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 RPS, [[ ]] and other D-in-D design features and design functions to ensure the capability to shutdown the reactor and provide for control of reactivity and reactor fuel thermal limits, including acceptance criteria, to describe compliance with regulatory requirements and to describe the bases for any exemptions to regulatory requirements or alternative approaches to regulatory guidance that may be referenced in future licensing activities either by GEH in support of a 10 CFR 52 Design Certification Application (DCA) or by a license applicant for requesting a Construction Permit (CP) and Operating License (OL) under 10 CFR 50 or a Combined Operating License (COL) under 10 CFR 52.

2.0 TECHNICAL DESCRIPTION OF REACTIVITY CONTROL

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 design features and functions for the mitigation of loss-of-coolant accidents (LOCAs) and small pipe breaks, and for ensuring overpressure protection requirements for the reactor pressure vessel (RPV), are delineated in Licensing Topical Report (LTR) NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 5.1].

The reactivity control design features and functions incorporate design, analysis, and operating experience from the BWR operating fleet, the ABWR, and the ESBWR. Additional D-in-D design features and functions, and design improvements, have been incorporated in the design of the BWRX-300.

2.2 Systems and Components for Control of Reactivity

The BWRX-300 uses control rods in order to change reactivity and therefore reactor power. Additionally, the reactor core is designed to include burnable poisons to allow for efficient fuel loading and management of the cycle excess reactivity.

Design Requirements:

- The core design along with the control rod negative reactivity results in ample shutdown margin in order to ensure that the reactor can remain shutdown in a cold, xenon-free condition throughout the cycle by use of control rods alone with the highest worth control rod pair associated with an individual HCU withdrawn.
- The control rods are positioned in fine increments for normal power adjustments and are also used for rapid insertion by multiple means to achieve shutdown.

2.2.1 Control Rods

The BWRX-300 employs bottom-entry, cruciform-shaped control rods as is typical of BWRs since 1961. This design concept benefits from nearly six decades of service in fleets of operating BWRs around the world.

The control rods intended for use in BWRX-300 are based on the designs used in the operational BWR fleet. This means the methods to design, evaluate and analyze the control rods, in their role as the primary means of reactivity control, are well understood. They have been exercised repeatedly, improved over time, and remain in use today.

The application of this large base of operating experience to the BWRX-300 design supports extremely high reliability of the control rods (and supporting systems) in their reactivity control role.

Design Requirements:

- Diverse sources of control rod motive force and diverse sets of control and actuation logic are provided in the design to provide extremely high confidence that the control rods can be inserted into the reactor core when necessary.
- The control rods are used for power shaping, power level adjustments, and insertion of negative reactivity to achieve shutdown.
- Control rods are the primary means of achieving shutdown in normal operations, Anticipated Operational Occurrences (AOOs), Postulated Accidents, and beyond design basis events and severe accident scenarios.
- Control rods include a bayonet style coupling to prevent inadvertent uncoupling from the FMCRD.

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, including the location and relative placement of the control rod blades and associated fine motion control rod drives (FMCRDs).

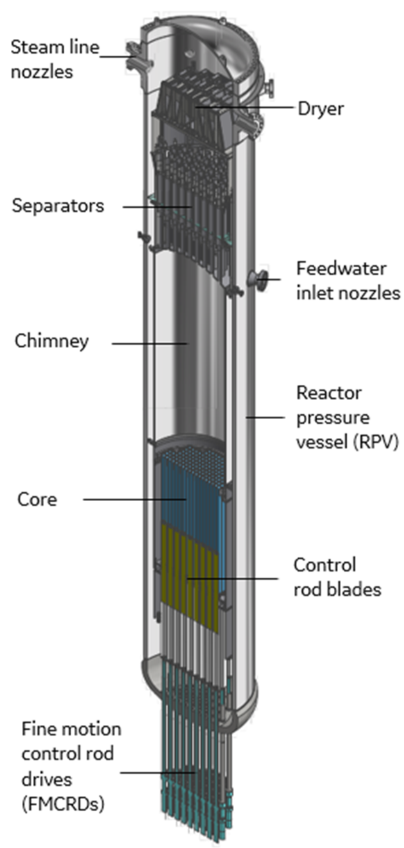

Figure 2-1: BWRX-300 Reactor Pressure Vessel and Internals

Each control rod includes a top handle, an absorber blade section, and a bottom coupling, assembled into a cruciform shape. A typical BWRX-300 control rod is shown in Figure 2-2. Each wing of an absorber blade is an array of stainless-steel tubes filled with boron carbide powder or a combination of boron carbide powder and hafnium rods. While moving vertically within the core, the absorber blade section travels through the cruciform envelope between surrounding fuel bundles. Handle pads guide the control rod along channels, and bottom connector rollers guide the control rod within a guide tube as the control rod is inserted and withdrawn from the core.

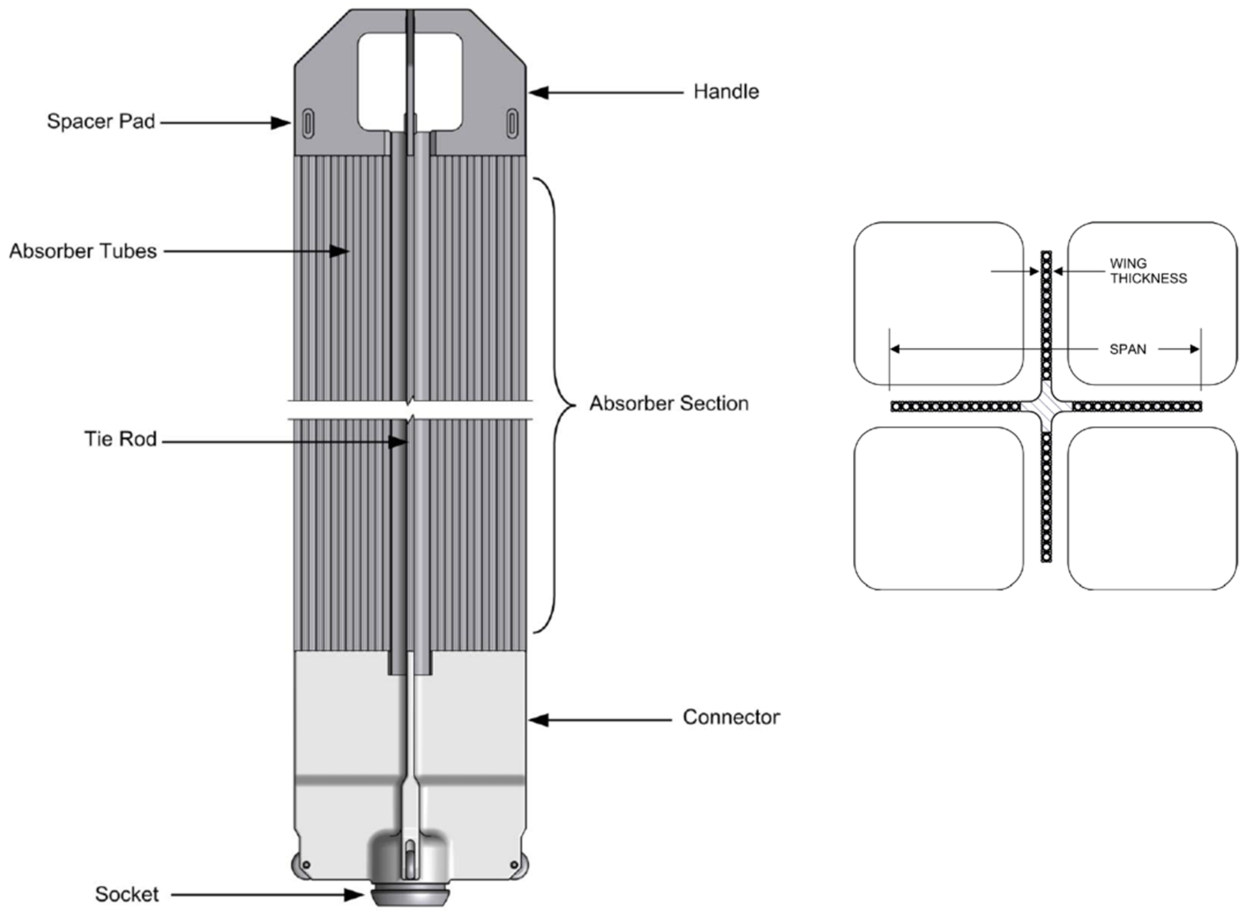

Figure 2-2: BWRX-300 Control Rod and Core Cell Arrangement

The BWRX-300 design includes 57 control rods distributed throughout the reactor core, which has 240 fuel bundle assemblies, as shown in Figure 2-3. The BWRX-300 core uses an N-Lattice configuration. This lattice design provides a larger dimension between fuel bundle centerlines than predecessor designs such as the BWR-6. This dimension includes the space for the control rod. A limited number of the control rods are used for normal power changes. The associated fuel cells are designated as control cells. The other control rods are primarily repositioned during significant power changes including startups and shutdowns. The control rods provide ample shutdown margin when inserted into the core during all conditions including when cold and xenon-free when the pair of control rods of the highest worth controlled by a single Hydraulic

Control Unit (HCU) is assumed to remain in the fully withdrawn position. A reactor core reload analysis is performed prior to every fuel cycle to define the core operating strategy for that cycle.

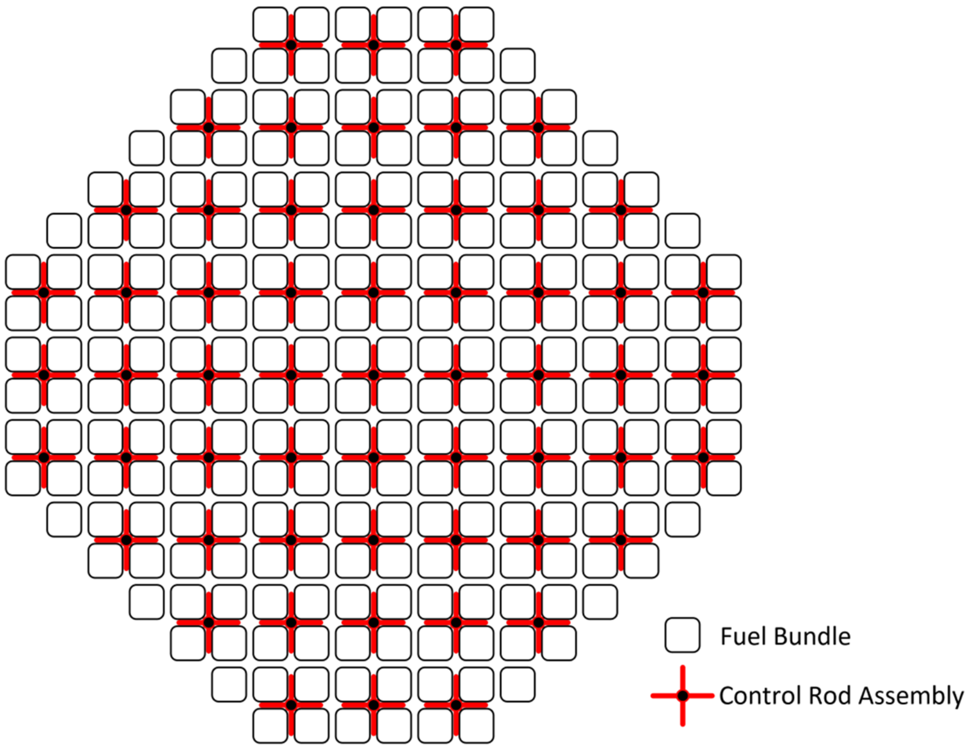

Figure 2-3: BWRX-300 Control Rod Locations Within Core

2.2.2 Control Rod Drives

Each Control Rod Assembly (i.e., control rod) is coupled to a Control Rod Drive (CRD) which is used to position the control rod.

Design Requirements:

- There are two diverse motive forces for the CRD and the associated control rod.
  - The control rods are normally positioned with an electric motor drive.
  - When a rapid shutdown is desired, the control rods are inserted hydraulically by use of high-pressure water.

The CRDs that are used for BWRX-300 are called FMCRDs to indicate the dual diverse means of movement as opposed to the predecessor locking piston hydraulically driven CRDs.

2.2.2.1 Fine Motion Control Rod Drive System

Design Requirements:

- During power operation, changes in core reactivity are controlled by movement and positioning of the control rods within the core, in fine increments, using FMCRD electric motors (one motor per control rod).
- The FMCRD motors also provide continuous run-in functionality to achieve shutdown.

- In the event of a postulated initiating event (PIE) that requires a rapid reactor shutdown, and if the reactor scram fails or is delayed, then the reactor is shut down by the electric motor run-in of FMCRDs function.
- [[

  ]]
- The FMCRDs include separation detection sensors that sense that the hollow piston along with the associated control rod are resting on the ball nut. These separation detection sensors actuate a control rod block signal.
- Rod withdrawal block signals prevent control rod withdrawal when required to enforce established control rod patterns.
- A rod withdrawal block signal prevents withdrawal of FMCRDs based upon an SRNM high period signal during startup.

The fine positioning and shutdown capabilities are achieved with a ball-nut and ball-screw arrangement driven by the FMCRD motor. The ball-nut is keyed to a guide tube to prevent its rotation and traverses through the guide tube vertically as the ball-screw is rotated. A hollow piston, connected to a control rod, rests on the ball-nut. The weight of the control rod keeps the hollow piston and ball-nut in contact during positioning in both insert and withdraw positioning. A schematic of the FMCRD including motor is illustrated in Figure 2-4.

[[

]]

Figure 2-4: BWRX-300 FMCRD Schematic

2.2.2.2 Control Rod Drive Hydraulic Scram System

In BWR designs, a fast insertion of control rods using stored hydraulic energy is referred to as a scram or hydraulic scram. The force required for hydraulic scram is provided by 29 HCUs that include nitrogen-charged accumulators. A hydraulic scram is initiated by opening 29 scram valves, one on each accumulator water discharge path.

Design Requirements:

- Scram valves open by spring action and are normally held closed by pressurized control air.
- To cause hydraulic scram, a de-energizing reactor trip signal is provided to solenoid-operated pilot valves that vent the control air from the scram valves for opening.
- The design also includes a [[
  ]].
- The scram valves are "fail safe" in that loss of either electrical power to the solenoid pilot valve or loss of control air pressure results in opening the scram valve, and hydraulic insertion of all control rods.
- When an accumulator water discharge path through a scram valve is opened, high-pressure nitrogen raises a piston within the accumulator forcing water through the scram piping at high pressure.
- A single accumulator's water is directed to a scram inlet connection on each of two CRDs, with exception of the one CRD located in the center of the core which has its own dedicated accumulator.
- Inside each CRD, the high-pressure water bypasses the ball-nut and lifts the hollow piston, driving the control rod into the core.
  - The scram water is discharged directly into the reactor vessel via clearances between CRD components.
  - A spring washer buffer assembly stops the hollow piston at the end of its stroke.
- Departure from the ball-nut releases spring-loaded latches in the hollow piston to engage slots in the guide tube.
  - These latches support the hollow piston in the fully inserted position.
  - Following a hydraulic scram insertion, the control rod cannot be withdrawn until the ball-nut is driven up, re-engaged, and the hollow piston de-latched from the guide tube.
- The design also includes ARI pilot valves on the control air header, which serves all 29 scram valves.
  - The ARI pilot valves are energized-to-actuate and provide an alternate path to vent control air and open all scram valves resulting in hydraulic insertion of all control rods.
- Any time an automatic hydraulic scram is initiated, a "scram follow" signal is generated such that each FMCRD motor drives the ball-nut to a position just below the fully inserted hollow piston.

– This establishes a second means (in addition to the spring-loaded latches described above) to prevent any control rod from dropping out of its fully inserted position.

Figure 2-5 shows a simplified view of an FMCRD depicting hydraulic scram.

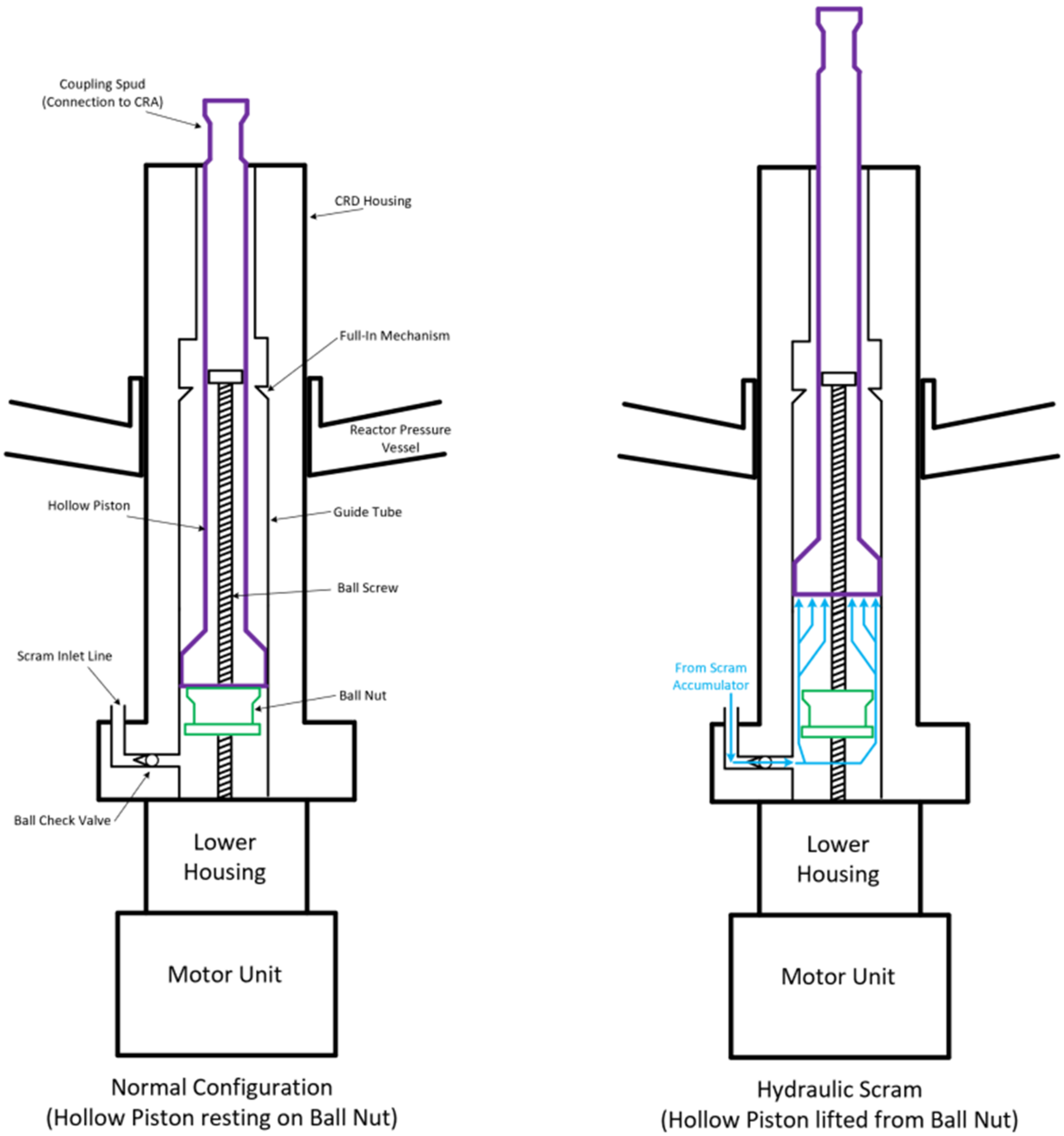

Figure 2-5: BWRX-300 Simplified View of FMCRD with Hydraulic Scram

Figure 2-6 shows hydraulic scram and ARI operation.

[[

]]

Figure 2-6: BWRX-300 Hydraulic Scram and ARI Schematic

The scram functions are categorized as defense line (DL) functions as further described in Section 3.0 to provide D-in-D for scram. Hydraulic scram by the RPS is a DL3 Safety Category 1 safety-related function, ARI hydraulic scram is a DL4a Safety Category 2 function, [[ ]]. The scram follow is designated as a Safety Category 3 function and is not needed for success of hydraulic scram. [[ ]]

2.3 BWRX-300 Associated Mitigating Systems

Design Requirements:

- In the case of a PIE that requires a rapid reactor shutdown, and if the reactor scram fails or is delayed, then the [[ ]].

The ICS is described in LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 5.1]. A brief description of these features and the associated effects are described below.

2.3.1 Isolation Condenser System

The arrangement of one IC heat exchanger situated in an IC pool is shown in Figure 2-7. The [[ ]] is discussed in NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 5.1].

[[

]]

Figure 2-7: BWRX-300 Isolation Condenser System (Only One Train Shown)

Design Requirements:

- 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 and evaporating water in the IC pool, which is vented to the atmosphere.
- [[
  ]]
- To start an IC train, the IC condensate return valve is opened whereupon the standing condensate drains into the reactor. [[
-

–

–

–

]]

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

]].

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 [[

]]

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.

3.0 DEFENSE-IN-DEPTH OF REACTIVITY CONTROL FUNCTIONS

3.1 General Overview of Defense-in-Depth (D-in-D) Concept

A plant-level D-in-D concept is applied to the BWRX-300 design. The BWRX-300 design is arranged in defense lines, consistent with the levels of defense defined in International Atomic Energy Agency (IAEA) SSR-2/1 [Reference 5.3]. The D-in-D concept defines the design and analysis rules governing that arrangement such that the defense lines have good alignment with the safety assessments defined in a BWRX-300 safety assessment framework used to demonstrate plant safety. The D-in-D concept is applicable to the BWRX-300 systems and equipment responsible for performing functions assigned to each DL. This includes both the primary systems directly acting on the nuclear and heat generation processes and their supporting systems. The functions are assigned to DLs based on their roles in the layered, deterministic design basis analyses including Baseline Design Basis Analysis (BL-DBA), Conservative Design Basis Analysis (CN-DBA), and Extended Design Basis Analysis (EX-DBA).

The BWRX-300 uses a D-in-D integrated [[

]] This is further described in LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 5.1].

Multiple defense lines provide layered protection against unacceptable releases of radiation. The defense lines include engineering and operational practices, plant features, and plant functions. These features, functions and practices are designed such that:

- The existence of pre-conditions which could lead to accident scenarios are minimized.
- The normal operation of the plant is monitored and controlled such that PIEs can be mitigated before evolving into accident scenarios.
- If an accident scenario does develop, the consequences are limited.
- Multiple defense lines are independently capable of performing the plant's fundamental safety functions (FSFs).

Five defense lines are adopted for the BWRX-300 D-in-D concept, consistent with the IAEA lines of defense approach. The first and fifth defense lines do not include performance of plant functions. The first line minimizes potential for accidents to occur in the first place and minimizes potential for failures to occur in subsequent defense lines by applying high-quality and conservatism in design, construction and operation. The fifth line involves off-site emergency preparedness to protect the public in case a substantial radioactive release occurs. The second, third, and fourth lines comprise plant functions which act to prevent PIEs from leading to significant radioactive releases.

Among the second, third and fourth defense lines, two independent and diverse lines can mitigate any PIE caused by a single failure or by a single operator error.

Among the second, third and fourth defense lines, at least one line can mitigate any PIE caused by a CCF in another line, with the mitigation means being independent from the effects of the initiating CCF.

The fourth defense line additionally contains independent provisions for prevention and/or mitigation of severe accidents.

The adequacy of the defense lines is assessed and demonstrated using layered, deterministic safety analyses which are designed to exercise the different defense lines. The PIEs to be considered in these analyses are selected based on rigorous and systematic failure modes and effects analyses of the plant systems, as well as internal and external hazard evaluations, and human operation hazards.

Functional and design requirements are derived from the deterministic safety analyses, and from the D-in-D concept itself, to ensure that the defense line functions are implemented in the design consistent with their role in the D-in-D concept, and the credit taken for them in the safety analyses.

The above concept states that for any PIE due to a CCF, at least one diverse and independent defense line must be able to mitigate the PIE's effects. This approach is consistent with the general IAEA guidance to provide "several" layers of protection. When DL1 and DL4b are explicitly considered, several layers of protection are provided:

- DL1 is provided to protect against occurrence of the CCF in the first place,
- An additional defense line, among DL2-4a, is provided to mitigate the effects of the CCF if it were to occur, and
- DL4b is provided as further protection against the sequence becoming a severe accident.

Figure 3-1 shows a representation of the D-in-D concept. The BWRX-300 functions to support D-in-D associated with reactivity control are discussed below. The combination of these functions ensures that the BWRX-300 adequately controls reactivity during normal, abnormal, design basis, and CCF events. The functions associated with reactivity management and associated events are listed for each defense line in the sections below.

[[

]]

Figure 3-1: BWRX-300 Simplified Defense-in-Depth Concept

3.2 Defense Line 1

DL1 includes the quality measures taken to minimize potential for failures and initiating events to occur in the first place and to minimize potential for failures to occur in subsequent lines of defense. These quality measures cover the design, construction, operation and maintenance of the plant. DL1 also includes the use of appropriate conservatism in design and analyses. DL1 does not include plant functionality; the plant functions themselves belong to subsequent lines of defense. However, the DL1 quality and conservatism measures are inherent in the design performance, and reliability of subsequent defense line functions, support the effectiveness of those functions, and provide assurance that opportunities to defeat multiple defense lines are minimized.

DL1 measures result in fewer challenges to the RPS. By incorporating operating experience into the design and operation as well as including appropriate quality measures, the plant experiences fewer trips. The BWRX-300 design benefits from the experience gained from the prior generations of BWRs and other power plants for select features. The continual learning incorporated into the design and operation results in elimination of some events and the reduction of frequency and severity of other events.

Although there are many DL1 activities that are applicable to reactivity control, some examples include the following:

- Technical Specification operational controls
- N-Lattice core less likely to experience control rod binding
- Advances in channel materials and core design/operation minimize probability of channel bow
- Normal power changes are with control rods – continuous observation of normal function
- Reliability measures included in design minimize probability of PIEs and failure of mitigation
- [[ ]]
- Seismic qualification ensures core geometry maintained
- FMCRDs similar to ABWR and ESBWR
- Control Rod Blades
  - Same as ABWR
  - Almost identical to latest design for BWR fleet
- ABWR fleet has 22+ years of operating experience with control rod blades and FMCRDs
- Bayonet style coupling of control rod to FMCRD to prevent inadvertent uncoupling

3.3 Defense Line 2

DL2 contains plant functions designed to control or initiate responses to PIEs, especially AOOs, before any parameters reach a DL3 actuation setpoint. The effectiveness of DL2 functions is assessed in the BL-DBA, with the DL2 functions, and equipment performing those functions, subject to functional and design requirements derived from the BL-DBA.

Those functions which normally operate to control the plant parameters on a continuous basis are part of DL2. Other functions such as blocking control rod motion and anticipatory plant trips are also part of DL2. The functions in DL2 are assigned to Safety Category 3 and performed by (at least) Safety Class 3 equipment as defined in IAEA guidance. These functions are considered non-safety related in the US, but the appropriate quality and reliability measures are applied to ensure functional performance as a D-in-D measure. The functions in DL2 must be performed independently from DL3 functions, and any portion of DL2 functions subject to CCF must be performed diversely from corresponding portions of functions in DL3.

The DL2 functions are used as normal plant controls and therefore operate regularly and with a high degree of reliability. Because of their importance to plant operation, the D-in-D functions are supported by operational reliability measures. These features control or prevent many PIEs. They minimize the probability that a PIE will challenge DL3.

DL2 features that are important for reactivity control include the following:

- [[
-
-
  ]]
- The rod control system provides normal control of control rods within established movement patterns
- Control rod blocks mitigate incorrect rod withdrawals

3.4 Defense Line 3

DL3 contains plant functions which act to mitigate a PIE by preventing fuel damage when possible, assuring the integrity of the barriers to release, and placing the plant in a safe state. DL3 also includes functions credited to maintain the plant in a safe condition following mitigation of PIEs, until normal operations are resumed. The effectiveness of DL3 functions is assessed in the CN-DBA, with the DL3 functions, and equipment performing those functions, subject to functional and design requirements derived from the CN-DBA.

DL3 functions typically include reactor scram and actuation of engineered safety features. The DL3 functions are needed when DL2 is not effective at intercepting a PIE or when a PIE is simply beyond the capabilities of the DL2 functions. The functions in DL3 are assigned to Safety Category 1 and performed by Safety Class 1 equipment as defined by IAEA guidance. These features are treated as safety-related in the US.

The systems and equipment involved in performance of DL3 functions are as simple as possible. Examples of the desired simplicity include eliminating the need for active support systems (e.g., power supply; heating, ventilation, and air-conditioning (HVAC); and cooling water) and minimizing the need for active control functions (pumps, motors, and actively controlled valve positioners).

DL3 features that are important for reactivity control include the following:

- RPS hydraulic scram
- [[

–

–

]]

3.5 Defense Line 4

DL4 includes two subsets of functions, designated as DL4a and DL4b functions.

DL4a functions are those which can place and maintain the plant in a safe state in case of PIEs with failure of the DL3 functions. The DL4a functions should prevent the progression of accidents or radioactive release to the public. The need for DL4a functions generally arises from specific, postulated CCFs occurring in DL3. The effectiveness of DL4a functions is assessed in the EX-DBA, with the DL4a functions, and equipment performing those functions, subject to functional and design requirements derived from the EX-DBA.

DL4a functions are assigned to Safety Category 2 and performed by (at least) Safety Class 2 equipment as defined by IAEA guidance. These features are considered non-safety related in the US, but the appropriate reliability and quality control measures are included to ensure that they can be relied upon for D-in-D.

DL4b functions are those explicitly provided to prevent or mitigate an accident involving substantial melting of the nuclear fuel (i.e., a severe accident) while keeping radioactive releases to acceptable levels. DL4b also protects for events that exceed DL1 assumptions regarding PIEs as a result of extreme events, multiple events, or multiple failures.

DL4 features that are important for reactivity control include the following:

- ARI provides hydraulic scram in event of HCU actuation failure
- Electric motor run-in of FMCRDs

3.6 Defense Line 5

DL5 includes emergency preparedness measures to cope with potential unacceptable releases in case the first four defense lines are not effective. These are largely off-site measures taken to protect the public in a scenario involving substantial release of radiation.

DL5 measures are not in the scope of this LTR.

3.7 Specific Reactivity Control Events Considered in Defense-in-Depth Concept

The D-in-D approach that has been applied to the BWRX-300 results in elimination of select events from previous designs, reduction in the frequency of other events, and improved mitigation of events. This section provides a summary of select events. It is not considered an all-inclusive list of BWRX-300 events sequences.

3.7.1 Anticipated Transient Without Scram (ATWS)

As defined in 10 CFR 50.62, an ATWS is an AOO followed by a failure of the reactor trip system. Section 4.0 of this report includes the regulatory evaluation of compliance with the regulatory requirements of 10 CFR 50.62. This section describes the D-in-D features of the BWRX-300 that prevent or mitigate these specific reactivity control events.

As described in Section 3.2, DL1 activities minimize potential for failures and initiating events to occur in the first place that require actuation of the RPS. Industry data supports the fact that the operating nuclear fleet has effectively reduced the challenges to plant operation that require RPS actuation. BWRX-300 benefits from the predecessor BWR's operating experience.

For BWRX-300 [[ ]] A hydraulic scram signal with scram follow is actuated to quickly effect a shutdown. Additionally, the [[ ]]

For a case of an event such as described above, if the [[ ]] were to fail to perform the reactor shutdown, the RPS which is in DL3 will sense a [[ ]], such as high reactor pressure, and cause a shutdown through a hydraulic scram signal with scram follow. Additionally, for [[ ]]

If a scram signal is applied to the HCUs to cause the control rods to insert, but does not release the stored energy, a backup DL4 ARI signal is applied to the ARI solenoids causing the air header to depressurize, thereby resulting in a release of the HCU stored energy and insertion of control rods.

As further D-in-D against a failure to scram, the [[ ]]

Because the BWRX-300 uses natural circulation for reactor coolant flow, the action of tripping reactor recirculation pumps to limit core flow and power is not applicable. [[ ]]

The combined effects of the features described above provide multiple layers of defense to ensure an effective scram when needed and to control reactor conditions while the shutdown is being completed by any of the methods listed. These diverse shutdown methods and mitigating features result in a [[ ]].

The BWRX-300 acceptance criteria for evaluating the effectiveness of the reactivity control diverse shutdown methods and mitigating features include the following:

- Pressures in the reactor coolant system and main steam system are maintained below 120% of the reactor coolant pressure boundary (RCPB) design pressure (ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Service Level C limit);
- Peak cladding temperature is within the 10 CFR 50.46 limit of 1204ºC (2200°F);

- Peak cladding oxidation is within the requirements of 10 CFR 50.46;
- Peak containment pressure and temperature do not exceed containment design pressure and temperature;
- A coolable geometry is maintained; and
- Radiological releases are maintained within 10 CFR 100 allowable limits.

The BWRX-300 design requirements associated with ATWS prevention and mitigation include the following:

- [[
-
  ]]
- RPS initiates a reactor scram based on signals and setpoints needed to support safety analysis credited trips.
- [[
-
-
  ]]
- ARI provides a diverse means to actuate the HCUs upon sensing a failure to scram.
- FMCRDs receive an electric motor run-in signal upon sensing a parameter requiring a scram.
- [[
  ]]
- FMCRD insertion time limits are established based on meeting the acceptance criteria of the safety analyses.

3.7.2 Control Rod Drop Accident

As with the ESBWR, the BWRX-300 has features to prevent a Control Rod Drop Accident (CRDA). The FMCRDs have a different coupling design to attach the drive assembly to the control rod than was used for the locking piston control rod drives. The FMCRD uses a bayonet style coupling that requires a 45-degree rotation to uncouple it. Since the FMCRD is firmly bolted into its position under the reactor vessel and the control rod is constrained from rotation by the fuel assemblies, it is not possible for the control rod to become uncoupled from the FMCRD during reactor operation. The hollow piston is the component within the FMCRD that is coupled to the control rod. The hollow piston normally rests on the ball nut internal to the FMCRD. There are dual separation detection devices that sense that the hollow piston along with the associated control rod are resting on the ball nut. If the sensor detects that the hollow piston is no longer on the ball nut, then control rod withdrawal is blocked. Additionally, the hollow piston has latches that prevent inadvertent withdrawal of the assembly when not attached to the ball nut.

These BWRX-300 features prevent inadvertent uncoupling of the control rod from the FMCRD, block withdrawal of the drive assembly if separation of the hollow piston and ball nut occurs and latches a hollow piston that is not resting on the ball nut. Since these items do not allow a separation distance to occur between the FMCRD and the control rod or from the ball nut to an unlatched hollow piston, it is not possible for a control rod drop accident to occur.

The BWRX-300 design requirements associated with CRDA prevention and mitigation include the following:

- FMCRDs include bayonet style coupling to prevent inadvertent uncoupling.
- FMCRDs include separation detection sensors that actuate a control rod block.

3.7.3 Rod Withdrawal Error

The protection system along with other mitigating design features assure that specified acceptable fuel design limits (SAFDLs) are not exceeded for rod withdrawal error events. In order to prevent a rod withdrawal error, the rod control system has redundancy to limit the effect of single failures. Additionally, the rod patterns are enforced for rod withdrawals. If there is a malfunction of the rod control system that results in a rod withdrawal error during startup, a rod block is initiated based upon a startup range neutron monitor (SRNM) high signal. If the rod withdrawal error were to result in a further increase to the SRNM based setpoint, a reactor scram occurs. If a rod withdrawal error were to occur at higher power, the average power range monitor (APRM) scram will terminate the event if it were to continue to its setpoint.

The BWRX-300 design requirements associated with rod withdrawal error prevention and mitigation include the following:

- The SRNMs provide a period based rod block function during startup.
- The SRNMs provide a period based reactor scram signal during startup.
- The APRMs provide a high flux reactor scram signal during power operations.

4.0 REGULATORY EVALUATION

4.1 10 CFR 50 Regulations

4.1.1 10 CFR 50.62

10 CFR 50.62, Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants, addresses an AOO as defined in 10 CFR 50, Appendix A, followed by the failure of the reactor trip portion of the protection system specified in 10 CFR 50 Appendix A, GDC 20. The following requirements are included for traditional BWRs:

- Regulatory Requirement: 10 CFR 50.62(c)(3) requires that each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

  Statement of Compliance: The BWRX-300 includes an ARI system as described in Section 2.2.2.2. The ARI system provides a diverse means of depressurizing the scram air header to ensure that the HCU stored energy is released to cause a reactor scram. Therefore, the BWRX-300 design will meet this requirement of 10 CFR 50.62(c)(3).

- Regulatory Requirement: 10 CFR 50.62(c)(4) requires that each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.

  Statement of Compliance: [[

  ]] Acceptance criteria as described in NUREG-0800, Standard Review Plan (SRP) 15.8, states that for evolutionary plants like the BWRX-300 some of the equipment required to satisfy the rule may not apply. For example, passive BWRs do not have recirculation pumps; therefore, these designs cannot provide equipment to trip them as required by the rule. For these designs provision of an equivalent action such as reducing the vessel water level may be acceptable.

  As discussed in Section 3.7.1, the BWRX-300 includes a comprehensive D-in-D approach to ensure that the reactor is effectively shut down when specified conditions are reached. The BWRX-300 includes a [[

]]. Additionally, the reactor can be shutdown by using the diverse FMCRD electric motor run-in. [[

]]

The BWRX-300 will meet the stated risk goal of 10 CFR 50.62 by providing a [[

]]. 10 CFR 50.62 was primarily issued to address a generic safety issue potentially affecting currently operating plants. The basis for ATWS rule requirements, which are outlined in SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," [Reference 5.4], which concluded that additional ATWS safety requirements were justified and included the stipulation to reduce the risk of core damage because of ATWS to be less than 10^{-5} per reactor year. [[

]]

Requirements for evolutionary plant designs beyond the original 10 CFR 50.62 regulatory requirements are addressed in more detail in SECY-93-087, "Policy, Technical, And Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR Designs)" [Reference 5.5] and addressed in SRP 15.8. The BWRX-300 design will meet the diversity and D-in-D guidelines for ATWS described in SECY-93-087, Item II.Q, Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems, without the use of an SLCS.

The effectiveness of the ATWS Rule requirements was evaluated in NUREG-1780 [Reference 5.6], which states that during the ATWS rulemaking the NRC staff set a goal that P(ATWS) should be no more than 1.0E-05/RY. P(ATWS) was defined as the annual frequency of an ATWS leading to plant conditions that exceed certain design parameters that can result in core melt, containment failure, and the release of radioactivity and can be viewed as the expected core damage frequency (CDF) of an unmitigated ATWS. Updating the original generic ATWS regulatory analysis, using operating data since the ATWS rule was implemented, found that on a generic basis, all four reactor types achieved the ATWS rule risk goal. The risk of core damage from a single CCF to scram is further reduced by reducing challenges to the RPS. NUREG-1780 notes that the initiating event frequency has been reduced by a factor of eight demonstrating that the Commission's recommendation to reduce the number of automatic reactor scrams has been very effective in reducing P(ATWS) (the probability of an ATWS). [[

]]

This statement of compliance may be used as the bases for the necessary exemption in future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant for requesting a CP and OL under 10 CFR 50 or a COL under 10 CFR 52.

- Regulatory Requirement: 10 CFR 50.62(c)(5) requires that each boiling water reactor must have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

  Statement of Compliance: Because the BWRX-300 uses natural circulation for reactor coolant flow, the action of tripping reactor recirculation pumps to limit core flow and power is not applicable. [[

  ]]

  Therefore, this requirement is not applicable to the BWRX-300. The above statement of compliance may be used as the bases for this conclusion in future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant for requesting a CP and OL under 10 CFR 50 or a COL under 10 CFR 52.

4.1.2 10 CFR 50 Appendix A, GDC 12

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 12, Suppression of reactor power oscillations, requires that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.

  Statement of Compliance: The BWRX-300 design addresses stability utilizing an RPV chimney that increases natural circulation core flow so that a margin to instability is maintained for all modes of operation. In accordance with GDC 12, it is required for the operating BWRs and the BWRX-300 that the power response to a sudden insertion of reactivity (typically a pressurization event) results in a coupled power and core flow response such that SAFDLs are not exceeded. For some reduced flow scenarios in forced circulation BWRs, a scram is required when the power and flow response is not damped to prevent SAFDLs related to critical power ratio (CPR) from being exceeded. Additionally, forced circulation BWRs have enforced operational exclusion zones to avoid regions of high power and low flow. The BWRX-300 maintains a coupled power and flow response such that any initial perturbation that does not cause an immediate scram decays quickly to steady state even at the limiting point of the cycle. Also, the relatively small core of the BWRX-300 causes it not to be susceptible to regional modes of oscillation. Therefore, a special stability detection and trip system is not required. The natural circulation driving head provided by the BWRX-300 chimney along with the core orifice design causes core flow and power perturbations to be naturally damped.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 12.

4.1.3 10 CFR 50 Appendix A, GDC 20

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 20, Protection system functions, requires that the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that SAFDLs are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

  Statement of Compliance: The BWRX-300 uses a D-in-D approach to prevent and mitigate events. In addition to the prevention features included in the rod control system, the [[

  ]]. The RPS provides timely and appropriate protection to provide a reactor scram for events exceeding limits. These systems ensure that SAFDLs are not exceeded. Scram trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but not be subject to spurious scrams. The analyses to demonstrate compliance will be provided during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 20.

4.1.4 10 CFR 50 Appendix A, GDC 21

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 21, Protection system reliability and testability, requires that the protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

  Statement of Compliance: The BWRX-300 uses Safety Class 1, safety-related equipment to ensure that high quality is achieved. The system includes redundancy to ensure that trips are reliably enforced, even in the case of a failure of a portion of the system. The ability to test and verify operability is included in the design.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 21.

4.1.5 10 CFR 50 Appendix A, GDC 22

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 22, Protection system independence, requires that the protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Statement of Compliance: The BWRX-300 uses a D-in-D approach to ensure that safety is maintained. The reactivity control system prevents inappropriate reactivity additions from the control rods. [[ ]] RPS provides a reactor scram when pre-set limits are reached. In addition to the diversity provided by these multiple layers of defense, the appropriate redundancy is included to ensure that reliability is maintained even in the event of failures. The RPS is Safety Class 1, safety-related equipment to ensure that high quality is achieved. The RPS and associated sensors and actuation devices are protected from natural phenomena and are designed as fail-safe to ensure that the safety function is maintained.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 22.

4.1.6 10 CFR 50 Appendix A, GDC 23

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 23, Protection system failure modes, requires that the protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

  Statement of Compliance: The BWRX-300 protection system, RPS, is designed such that it fails in a safe state. Upon loss of electrical power or motive force (i.e., air to the HCUs), a reactor scram occurs. The HCUs use stored energy for control rod insertion that are activated by the loss of electrical power to the actuating solenoids. This design ensures a safe state is achieved.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 23.

4.1.7 10 CFR 50 Appendix A, GDC 24

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 24, Separation of protection and control systems, requires that the protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

  Statement of Compliance: The BWRX-300 protection system, RPS, is separated from the control systems such as the rod control system such that the RPS effectively performs its function independent of the control systems.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 24.

4.1.8 10 CFR 50 Appendix A, GDC 25

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 25, Protection system requirements for reactivity control malfunctions, requires that the protection system shall be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

  Statement of Compliance: The BWRX-300 protection system along with other mitigating design features assure that SAFDLs are not exceeded for reactivity control malfunctions. In order to prevent a rod withdrawal error, the rod control system has redundancy to limit the effect of single failures. Additionally, the rod patterns are enforced for rod withdrawals. If there is a malfunction of the rod control system that results in a rod withdrawal error during startup, a rod block is initiated based upon a SRNM high signal. If the rod withdrawal error were to result in a further increase to the SRNM based setpoint, a reactor scram will occur. If a rod withdrawal error were to occur at higher power, the APRM scram terminates the event if it were to continue to its setpoint. The analyses to demonstrate compliance will be provided during future licensing activities.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 25.

4.1.9 10 CFR 50 Appendix A, GDC 26

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 26, Reactivity control system redundancy and capability, requires that two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, SAFDLs are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

  Statement of Compliance:

  Two independent reactivity control systems of different design principles are provided.

  Like previous BWR designs, including the ABWR and ESBWR, one of the two required independent reactivity control systems of different design principles is the hydraulic insertion of the control rods. A positive means for inserting the control rods is the highly reliable HCUs in both operating BWRs and the BWRX 300. In addition, the BWRX-300 has redundant and independent systems ([[ ]]), RPS, and ARI) to ensure that a hydraulic scram is initiated when required.

Additionally, BWRX-300 has the capability of electric motor-driven control rod movement that was not present in operating BWRs with only hydraulic control rod drive systems. This capability also exists for ABWR and ESBWR. Normal power changes in the BWRX-300 are made by the rod control system using fine motion control of specific control rods using the electric motor-driven control of the FMCRDs and by burnable poison, gadolinium, that is included in the fuel pellets in an axial and radial distribution within the core. The BWRX-300 rod control system and associated motor-driven control rod movement is a means of reactivity control that uses an independent control and motive force other than the hydraulic HCU insertion system.

The control blades are capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. Burnable gadolinium in the fuel and the FMCRDs provide for slow reactivity adjustments. The ganged movement of blades via the motors and the scram function provide protection when needed for AOOs and design basis accidents (DBAs).

In BWRs, the second means of reactivity control is the ability to respond to power shape changes by changing core flow and recirculation ratio. In the operating BWR fleet, the recirculation flow is forced using either internal or external recirculation pumps. The BWRX-300, like the ESBWR, does not employ forced circulation, but the core flow does change naturally in response to changes in reactivity and axial power shape in the same way as a jet pump BWR that is operating in natural circulation. For natural circulation, the core flow depends on the downcomer water level, and reducing the water level reduces core flow and thus core power. The BWRX-300 has the ability to reduce power via this mechanism. The Feedwater Level Control System is used to control feedwater flow and therefore controls reactor water level. This system can be used to make adjustments to water level when in normal power operation, and additional means are available for water level adjustments when in other modes of operation.

One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

The requirement that one of the systems shall be capable of holding the reactor core subcritical under cold conditions is satisfied in operating BWRs and the BWRX-300 by the insertion of the control blades. The use of gadolinium burnable poison in the fuel pellets is a diverse means of reactivity control that controls the power profile at the beginning of core life and enables the control rods alone to have adequate shutdown margin. The evaluation to demonstrate compliance considers the highest worth control rod pair associated with an individual HCU to be fully withdrawn and will be provided during future licensing activities.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 26.

4.1.10 10 CFR 50 Appendix A, GDC 27

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 27, Combined reactivity control systems capability, requires that the reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Statement of Compliance: As described in LTR NEDC-33910P, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection [Reference 5.1], the required emergency core cooling system (ECCS) design functions of the [[

]] to meet the criteria in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(4). Another design function of the [[
]] which is further required to maintain a coolable geometry to meet the 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 requirements of 10 CFR 50.46(b)(5) and only requires operator action to [[ ]] after approximately seven days. The evaluation to demonstrate compliance considers the highest worth control rod pair associated with an individual HCU to be fully withdrawn and will be provided during future licensing activities. [[
]] The control rods, FMCRDs, and actuation systems ensure adequate shutdown margin, capability, redundancy, and diversity such that there is no need for combined reactivy control systems as required by GDC 27. [[

]]

Based on the above discussions, the special circumstance as specified in 10 CFR 50.12(a)(2)(ii) is present justifying an exemption to these specific requirements of 10 CFR 50 Appendix A, GDC 27. The application of the regulation in these particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. Instead, the following PDC 27 is proposed:

> PDC 27, Reactivity control system capability, the BWRX-300 reactivity control system shall be designed to have the capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

These statements of compliance and proposed PDC 27 may be used as the bases for the necessary exemption in future licensing activities either by GEH in support of a 10 CFR 52 DCA or by a license applicant for requesting a CP and OL under 10 CFR 50 or a COL under 10 CFR 52.

4.1.11 10 CFR 50 Appendix A, GDC 28

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 28, Reactivity limits, requires that the reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Statement of Compliance: The combined features of the CRD system and the rod control system incorporate appropriate limits on the potential amount and rate of reactivity increase. The fine motion movement capability of the FMCRD allows reactivity additions from rod withdrawal to be limited. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worth. The BWRX-300 design prevents rod drop and rod ejection events through positive design means. Control rod drop is prevented by use of a bayonet style coupling, CRD mechanism latches, and CRD separation switches. Control rod ejection is prevented by physical constraints including the attachment of the control rod guide tube to the core plate and the CRD connection to the control rod guide tube. The FMCRD includes a brake that further prevents inadvertent rod withdrawal. The FMCRD also includes an internal ball check valve, which reduces the chances of rapid rod withdrawal. The ball check valve functions as a safety related component because it prevents reverse flow from the scram inlet port against the pressure and flow conditions caused by a break of the scram line, preventing the loss of pressure from the underside of the hollow piston and the generation of loads on the drive that could cause a rapid rod withdrawal and associated reactivity insertion. Normal rod movement and the rod withdrawal rate are limited by the FMCRD.

The rod control system controls rod patterns and provides control rod blocks to limit the rate and amount of reactivity addition for control rod movement. Compliance with GDC 28 was demonstrated in ESBWR Design Control Document Section 15.4.7.3 by analysis of the consequences of a postulated CRDA [Reference 5.2]. ESBWR Request for Additional Information (RAI) 4.6-23 (ADAMS Accession Nos. ML081090147 (Public) and ML081090148 (Non-Public)) summarized the results of the analysis [Reference 5.7]. In NUREG-1966, Volume 3, Section 15.4.7.3, the Staff Evaluation noted significant conservatism in the ESBWR analysis. There are no significant differences between the ESBWR approved analysis and the design and response of the BWRX-300.

The safety analyses to demonstrate compliance will be provided during future licensing activities and will evaluate PIEs and resulting transients or accidents including the steam line break and events that could change the reactor coolant temperature and pressure including cold water additions. The reactivity control systems include appropriate mitigating features for these events. The BWRX-300 CRD coupling is the same as the ESBWR. The CRDA event applied to an equilibrium cycle will be analyzed as a Special Event during future licensing activities using the approved Global Nuclear Fuels (GNF) CRDA Methodology (NEDE-33885P-A) [Reference 5.8]. Future licensing activities will confirm the expected similarities and support a conclusion that additional cycle-by-cycle CRDA evaluations are not warranted.

The design meets the requirements of GDC 28 by providing reactivity control systems features that mitigate postulated reactivity accidents that could result in damage to the

RCPB greater than limited local yielding or damage that impairs core cooling capability significantly.

Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 28.

4.1.12 10 CFR 50 Appendix A, GDC 29

- Regulatory Requirement: 10 CFR 50 Appendix A, GDC 29, Protection against anticipated operational occurrences, requires that the protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

  Statement of Compliance: The BWRX-300 rod control system enforces control rod withdrawal limits and prevents inappropriate control rod withdrawal. For events that result in AOOs such as a turbine trip, the [[
  ]]. If the event reaches the RPS trip setpoints, a scram is enforced by the RPS. The combination of these system provides D-in-D protection for AOOs and more significant events. The RPS is Safety Class 1, safety-related equipment to ensure that high quality is achieved. The RPS and associated sensors and actuation devices are protected from natural phenomena and are designed as fail-safe to ensure that the safety function is maintained.

  Therefore, the BWRX-300 design will meet the requirements of 10 CFR 50 Appendix A, GDC 29.

4.2 NUREG-0800 Standard Review Plan Guidance

4.2.1 Standard Review Plan 4.3

SRP 4.3, Nuclear Design, Rev. 3, states that the areas of review include confirmation that design bases are established as required by the appropriate GDC. Areas concerning core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality of the reactor during refueling, stability, analytical methods, and pressure vessel irradiation are to be reviewed. GDC specified in the SRP 4.3 acceptance criteria relevant to this LTR include GDC 12, GDC 20, GDC 25, GDC 26, GDC 27, and GDC 28. The BWRX-300 will meet the requirements of 10 CFR 50 Appendix A, GDC 12, GDC 20, GDC 25, GDC 26, and GDC 28 as described in Section 4.1, except for the exemption justification provided for GDC 27 and proposed PDC 27 in Section 4.1.10.

The areas of review, review interfaces, acceptance criteria, review procedures, evaluation findings, and references specified for evolutionary plants are applicable to the BWRX-300, [[

]] Therefore, GEH recommends that the existing SRP may be used 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.

4.2.2 Standard Review Plan 7.2

SRP 7.2, Reactor Trip System, Rev. 6, states that the areas of review include confirming that the reactor trip system satisfies the requirements of the acceptance criteria and guidelines applicable to the protection system and performs its safety functions for all plant conditions under which the safety functions are required. GDC specified in the SRP 7.2 acceptance criteria relevant to this LTR include GDC 20, GDC 21, GDC 22, GDC 23, GDC 24, GDC 25, and GDC 29. The BWRX-300 will meet the requirements of 10 CFR 50 Appendix A, GDC 20, GDC 21, GDC 22, GDC 23, GDC 24, GDC 25, and GDC 29 as described in Section 4.1.

Therefore, the areas of review, review interfaces, acceptance criteria, review procedures, evaluation findings, and references are applicable to the BWRX-300 based on the design description and design requirements discussed in Sections 2.0 and 3.0 of this LTR for the RPS. Therefore, GEH recommends that the existing SRP may be used 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.

4.2.3 Standard Review Plan 15.8

SRP 15.8, Anticipated Transients Without Scram, Rev. 2, states that certain light-water-cooled plants have prescribed systems and equipment that have been determined to reduce the risks attributable to ATWS events, for each of the nuclear steam supply system (NSSS) vendor's designs, to an acceptably low level. The rule also requires applicants to demonstrate the adequacy of their plants' prescribed systems and equipment.

10 CFR 50.62 was issued to address a generic safety issue potentially affecting current operating plants. The basis for the ATWS rule requirements, which are outlined in SECY-83-293, include the stipulation to reduce the risk of core damage due to an ATWS to be less than 10^{-5} per reactor year. The BWRX-300 will meet the stated risk goal of 10 CFR 50.62 by [[

]].

During DCA reviews of evolutionary plant designs, the NRC developed additional requirements criteria (i.e., [[ ]] or to demonstrate that the consequences of ATWS events are acceptable). The BWRX-300 is an evolutionary plant, and SRP 15.8 has designated acceptance criteria for evolutionary plants. The specific acceptance criteria for evolutionary plants is included in the guidance provided in SRP 15.8 (Items 3.A. through 3.C.).

SRP Acceptance Criteria:

A. For evolutionary plants where the ATWS rule does not explicitly require a diverse scram system, the applicant may provide either of the following:

   i. A diverse scram system satisfying the design and quality assurance criteria specified in SRP Section 7.2.

   ii. Demonstrate that the consequences of an ATWS event are within acceptable values.

BWRX-300 discussion:

The BWRX-300 includes an [[

]]. Therefore, the BWRX-300 design conforms to this guidance.

SRP Acceptance Criteria:

B. For evolutionary plants, some of the equipment required to satisfy the rule may not apply. For example, passive BWRs do not have recirculation pumps; therefore, these designs cannot provide equipment to trip them as required by the rule. For these designs provision of an equivalent action such as reducing the vessel water level may be acceptable.

BWRX-300 discussion:

The specific equipment requirements described in 10 CFR 50.62, and the way the BWRX-300 addresses these requirements, are discussed in Section 4.1.1. This includes 10 CFR 50.62(c)(3) requiring an ARI system which is applicable to the BWRX-300 design, [[ ]] and 10 CFR 50.62(c)(5) requiring equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS which is not applicable to the BWRX-300 design. Therefore, the BWRX-300 design conforms to this guidance where applicable.

SRP Acceptance Criteria:

C. Applicants must demonstrate that the failure probability of failing the ATWS success criteria is sufficiently small because either: (1) the criteria are met, or [[ ]]. The analysis leading to the ATWS rule in NUREG-0460 used the following ATWS success criteria, which have their bases in the Commission regulations and GDC listed above. Applicant's design shall maintain:

i. Coolable geometry for the reactor core. If fuel and clad damage were to occur following a failure to scram, GDC 35 requires that this condition should not interfere with continued effective core cooling. 10 CFR 50.46 defines three specific core-coolability criteria: (1) Peak clad temperature shall not to exceed 1221°C (2200°F), (2) Maximum cladding oxidation shall not to exceed 17% the total cladding thickness before oxidation, and (3) Maximum hydrogen generation shall not to exceed 1% of the maximum hypothetical amount if all the fuel clad had reacted to produce hydrogen.

ii. Maintain reactor coolant pressure boundary integrity. Appendix A to WASH-1270 states that in evaluating the reactor coolant system boundary for ATWS events, "the calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the 'emergency conditions' as defined in the ASME Nuclear Power Plant Components Code, Section III." The acceptance criteria for reactor coolant pressure, based upon the ASME Service Level C limits, are approximately 10.3 MPa (1500 psig) for BWRs and approximately 22MPa (3200 psig) for Pressurized Water Reactors (PWRs).

iii. Maintain containment integrity. Following a failure to scram, the containment pressure and temperature must be maintained at acceptably low levels based on GDC 16 and 38. The containment pressure and temperature limits are design dependent; but to satisfy GDC 50, those limits must ensure that containment design leakage rates are not exceeded when subjected to the calculated pressure and temperature conditions resulting from any ATWS event.

BWRX-300 discussion:

> The BWRX-300 will meet the stated risk goal of 10 CFR 50.62 to reduce the risk of core damage due to an ATWS to be less than 10^{-5} per reactor year [[
>
>]]. As discussed in Section 3.7.1 and Section 4.1.1, these acceptance criteria are met through the defense in depth measures to insert control rods through hydraulically-driven or electric motor-driven means in combination with the [[ ]]. Therefore, the BWRX-300 design conforms to this guidance.

The areas of review, review interfaces, acceptance criteria, review procedures, evaluation findings, and references specified for evolutionary plants are applicable to the BWRX-300, with the exception of requiring [[ ]] and equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS which are not applicable to the BWRX-300 design, based on the design description and design requirements discussed in Sections 2.0 and 3.0 of this LTR. Therefore, GEH recommends that the existing SRP may be used 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 Generic Issues

The following generic issues do not represent the total listing required to support a 10 CFR 52 design certification application if pursued or for future 10 CFR 50 license applications but are provided based on their relevance to the scope of this LTR.

4.3.1 NUREG-1780

NUREG-1780, Regulatory Effectiveness of the Anticipated Transient Without Scram Rule, states that during the ATWS rulemaking the NRC staff set a goal that P(ATWS) should be no more than 1.0E-05/RY. As described in Section 4.1.1, the BWRX-300 will meet the stated risk goal of 10 CFR 50.62 as further defined in NUREG-1780 by [[

]].

5.0 REFERENCES

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

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 International Atomic Energy Agency, "Safety of Nuclear Power Plants: Design," Specific Safety Requirements No. SSR-2/1, Rev. 1

5.4 SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," USNRC, Washington, DC, July 19, 1983

5.5 SECY-93-087, "Policy, Technical, And Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR Designs)," ADAMS Accession Number ML003708021, USNRC, Washington, DC, July 21, 1993

5.6 NURGEG-1780, "Regulatory Effectiveness of the Anticipated Transient Without Scram Rule," USNRC, Washington, DC, September 2003

5.7 General Electric Hitachi Letter, "Response to NRC Request for Additional Information Letters No. 115 and No. 137 – Related to ESBWR Design Certification Application – RAI Numbers 4.6-23 Supplement 2 and 4.6-38, Respectively," April 14, 2008 (ADAMS Accession Nos. ML081090147 (Public) and ML081090148 (Non Public))

5.8 NEDE-33885P-A, Revision 1, Global Nuclear Fuels (GNF) CRDA Application Methodology, March 2020