ENCLOSURE 1

M230183

Presentation Slides for Pre-Application Meeting for Planned Submittal of GE Hitachi Nuclear Energy BWRX-300 Safety Strategy Topical Report NEDC-33934P

Non-Proprietary Information

Pre-Submittal Meeting For GE Hitachi Licensing Topical Report

NEDC-33934P
BWRX-300 Safety Strategy

March 8, 2023

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Agenda

- BWRX-300 Safety Strategy Topical Report Overview
- Safety Strategy Introduction
- Defense Line Architecture for Defense-in-Depth (D-in-D)
- BWRX-300 Safety Classification
- Improvements in the BWRX-300 from the ESBWR Design Where Additional Layers of Safety Added for D-in-D
- BWRX-300 Illustrative I&C Example
- BWRX-300 Illustrative Fault List Methodology
- Conformance With Standard Review Plan 19.3
- Conformance With Standard Review Plan Sections 3.1 and 3.2 – Classification of SSCs
- Conformance With Standard Review Plan Section 15.0, 15.1-15.9
- Proposed BWRX-300 Principal Design Criteria
- Technical Specifications Related to Defense Lines
- Conclusions
- Questions & Comments

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BWRX-300 Safety Strategy Topical Report Overview

Licensing Topical Report Purpose

- Demonstrate that the deterministic application of BWRX-300 multi-layer Defense-in-Depth (D-in-D) approach meets the safety goals for a risk-informed approach described in NUREG-0800 Standard Review Plan (SRP) Section 19.3
- Demonstrate compliance with the applicable requirements of 10 CFR 50 Appendix A General Design Criteria (GDC) and other 10 CFR regulations or request exemptions and/or propose BWRX-300 Principal Design Criteria (PDC)
- Demonstrate conformance with SRP for Chapter 15, Section 19.3 and Section 3.2
- Compliance with Technical Specification Selection Criteria of 10 CFR 50.36
- Identify improvements in the BWRX-300 design adding additional D-in-D from the ESBWR approved approach where nonsafety-related systems were credited for some anticipated operational occurrences (AOOs)
- Classification of BWRX-300 Structures, Systems, and Components (SSCs) as they relate to Defense Lines

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Licensing Topical Report Scope

BWRX-300 Safety Strategy Topical Report to address:

- Defense Line Architecture for Defense-in-Depth (D-in-D)
- Conformance With Standard Review Plan Section 19.3
- Conformance With Standard Review Plan Section 15.0
- Conformance With Standard Review Plan Sections 15.1-15.9
- Method Used in Establishing Frequency of Postulated Initiating Events (PIEs)
- Compliance With 10 CFR 50.36 (c)(2)(ii)
- Conformance With Standard Review Plan Section 3.2 – Classification of SSCs
- Improvements in BWRX-300 design from the ESBWR DESIGN where additional D-in-D from the ESBWR design is provided
- Exemptions and proposed PDC for selected GDCs and other regulations

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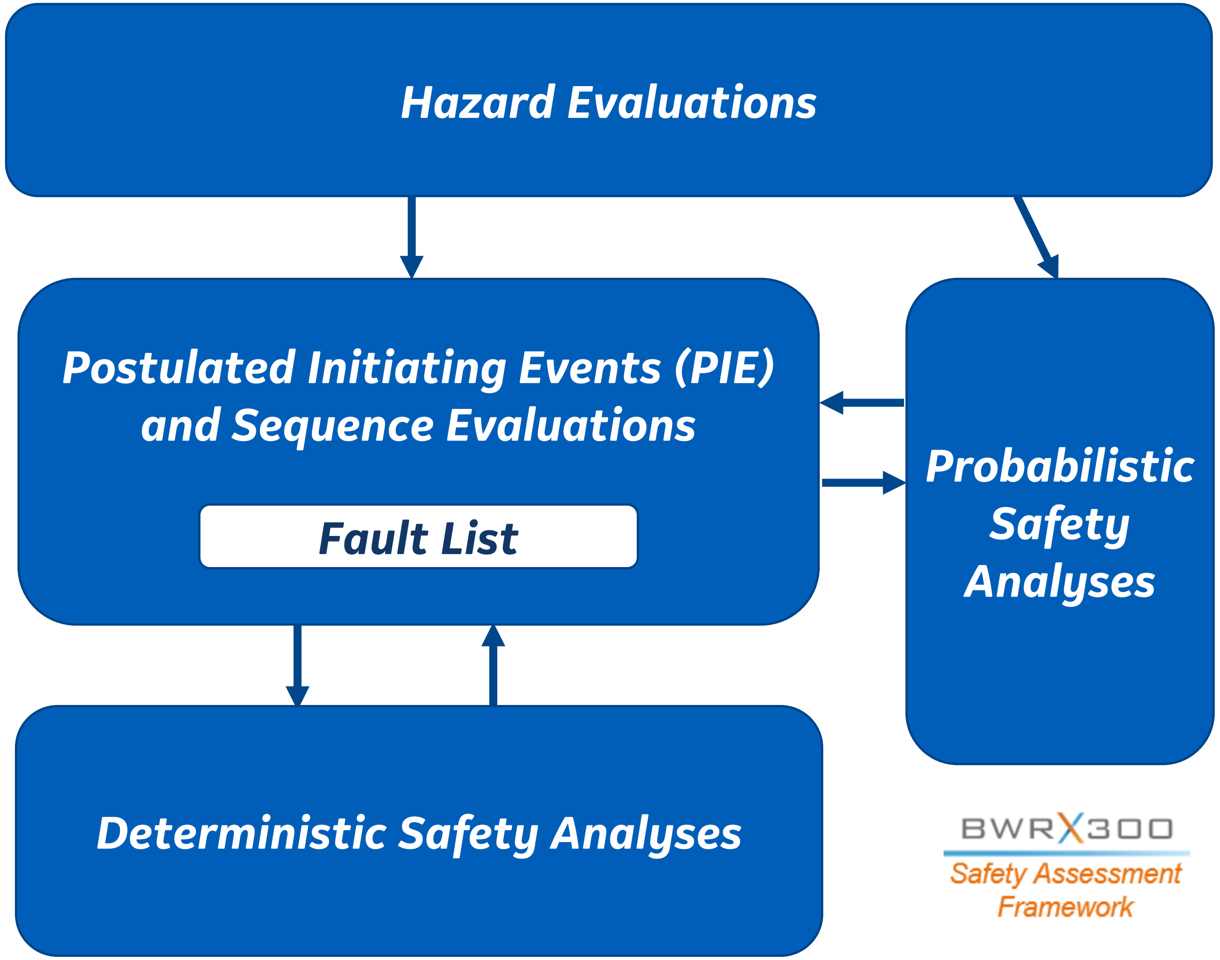
BWRX-300 Safety Strategy Introduction

BWRX-300 Safety Strategy



- Objective of plant design is to have a very low likelihood of reactor core damage or large release
- This safety and reliability goal is met by an inherently safe reactor design that relies on passive safety systems
- Design priorities in meeting the safety goals:
  - Inherent Safety
  - Passive Safety
  - Active Safety
  - Procedural Controls

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BWRX-300 Safety Strategy

- Perform hazard evaluation to define initiating events (failures, internal/external hazards and human error)
- Determine postulated initiating events (PIEs)
- Categorize PIEs based on probability (i.e., Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs), and Design Extensions Conditions (DECs))
- Define fault sequences based on PIEs and postulated failures
- Perform deterministic analyses to validate plant response
- Iterate as needed during the design phases

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BWRX-300 Defense Line Architecture for D-in-D

Defense Line Definitions

BWRX-300 Defense Line (DL)	Objective	Defense Line Implementation
DL1	Prevention of abnormal operation and defense line failures	Conservative design and high quality in fabrication and construction, including monitoring and control systems
DL2	Control of AOOs	Normal controls systems (e.g., control rod blocks, select control rod run in (SCRRI)), anticipatory trip systems, and other limitation features
DL3	Control of DBA PIEs and DBA event sequences, including CCF of DL2 functions	Reactor Trip Systems and engineered safety features
DL4a	Control of DEC PIEs and DEC event sequences to prevent core damage, including CCF of DL3 functions	Safety features such as the Diverse Protection System (DPS) for DECs without core damage
DL4b	Control of DEC event sequences to prevent or mitigate the consequences of severe accidents	Safety features such as containment overpressure vent, FLEX, or skid mounted diesels for refilling isolation condenser pools (ICS) for DECs, including events with core damage
DL5	Mitigation of radiological consequences of significant releases of radioactive materials	On-site and off-site emergency response facilities

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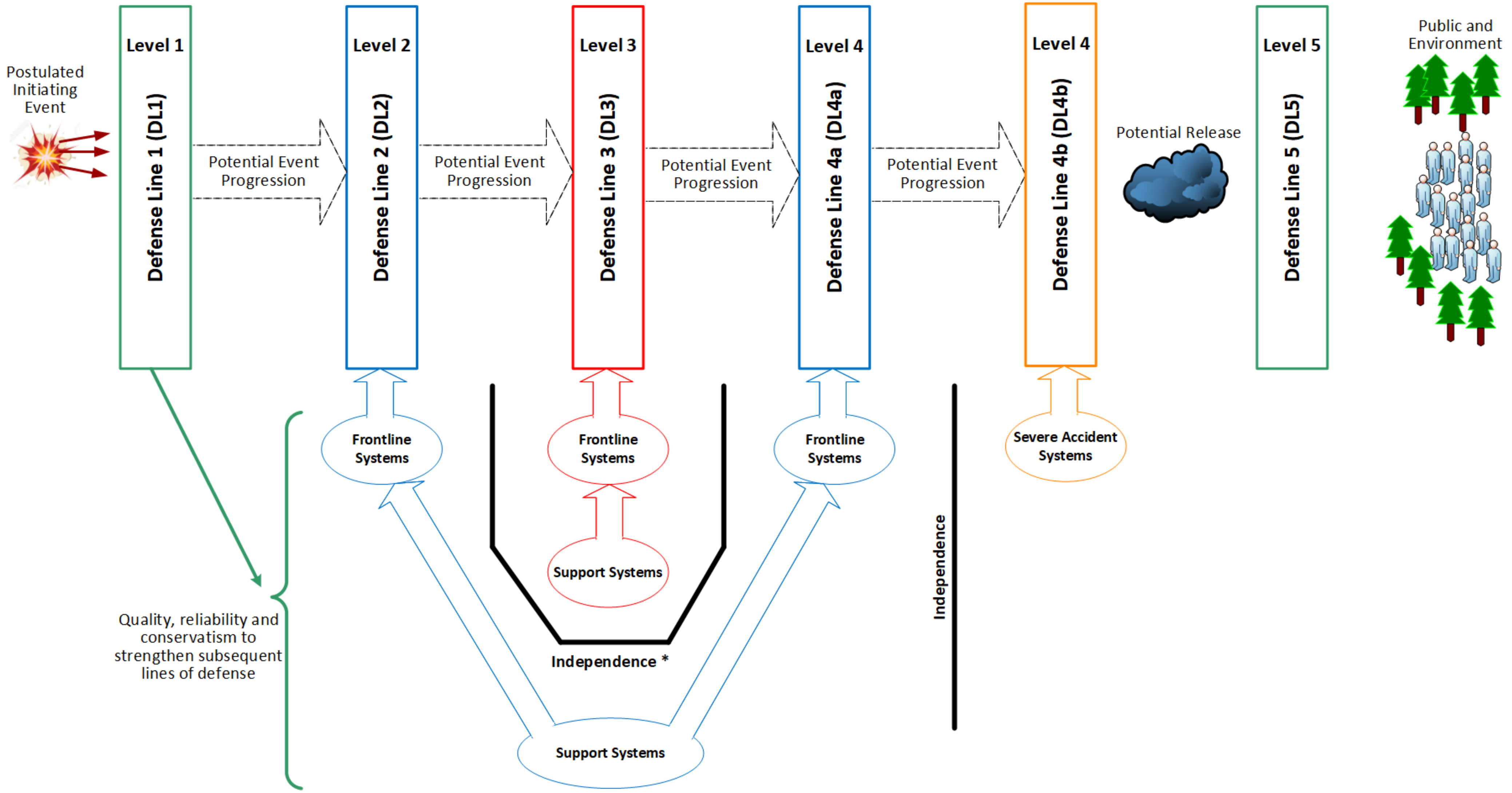
Defense in depth … built into the design from the start



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BWRX-300 Application of Defense Lines

All single failure PIEs with a frequency greater than 1E-05 can be mitigated by one of the following DL combinations:

- DL3 and independently by DL2
- DL3 and independently by DL4a
- DL3 and independently by DL4a plus unaffected DL2 functions

DL3 is independent from DL2 and DL4a to the extent practicable

DL2 and DL4a are not independently credited to mitigate the same PIEs and are not required to be independent from each other

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BWRX-300 Safety Classification

Safety Classification of Systems

Graded approach to SSC classification commensurate with the importance of the safety function ensures successful performance of the fundamental safety functions using a defense-in-depth (D-in-D) approach:

- Controlling reactivity
- Removing heat from the fuel and the fuel pool
- Confining radioactive materials

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Safety Category for Functions based on Defense Line Assignment

Safety Category	Defense Line 3 Functions	Defense Line 4a Functions	Defense Line 2/4b Functions
1	• Primary and integral support functions required within first 72 hours of event		
2	• Primary and integral support functions required after 72 hours but before 7 days	• Primary and integral support functions required within first 7 days of event	
3	• Primary and integral support functions required after 7 days (e.g., ICS replenishment) • Make-ready support functions (e.g., CRD Purge system)	• Primary and integral support functions required after 7 days • Make-ready support functions	• All primary and integral support functions (e.g., DL2-ATS and control rod blocks, maintain power, level and pressure; DL4b – containment vent or FLEX)
N			• Make-ready support functions

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Safety Classes of SSCs

- Components that are required to perform Safety Category 1 functions are assigned Safety Class 1 (SC1)
- Components that are required to perform Safety Category 2 functions are assigned at least SC2
- Components that are required to perform Safety Category 3 functions are assigned at least SC3
- An exception is (just as with functions) a time-dependency is introduced for components that perform or support DL3 and DL4a functions:
  - If the component is not needed until after 72 hours into the event, it can be classified as SC2 (instead of SC1)
  - If the component is not needed until after seven days into the event, it can be classified as SC3 (instead of SC1 or SC2)

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Functional Reliability Targets

Safety Category	Functional Reliability
1	1E-04 failures per demand
2	1E-03 failures per demand
3	1E-02 failures per demand

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Improvements in the BWRX-300 From the ESBWR I&C design Where Additional Layers of Safety Added for D-in-D

ESBWR and BWRX-300 Comparison

ESBWR	BWRX-300	Comments
AOOs and DBAs selected deterministically and classified by event frequency. Conservative event analysis methods used for all events. Transient scenarios involving multiple equipment failures were classified as infrequent events and addressed as postulated accidents. ESBWR DCD Section 15A.2 describes methodology used in determining event frequency.	AOOs and DBAs are both deterministically analyzed using risk-informed and experience-based insights and are classified by event frequency. Transient scenarios involving multiple equipment failures are classified as postulated accidents or design extension conditions based upon event frequency. A graded approach to event analysis methods is used based on event significance.	BWRX-300 Safety Strategy is a risk-informed approach that is different than both deterministic SRP approach and LMP risk-based approach.
Several AOOs are credited with control actions that limit event consequences (see ESBWR NRC FSER, Section 7.7.3.2.1 and 15.2.2.4). Failure of these control functions is accounted in event frequency (i.e., less frequent than an AOO). • Loss of Feedwater Heating (FWCS action) • Turbine Trip without Bypass (SCRRI action) DBAs are mitigated by safety-related protection system.	AOO PIE sequences are controlled by DL2 functions, SC3 safety-classified components. DBA PIEs and fault sequences not protected by DL2 CCF are mitigated by DL3, a SC1 safety-classified system.	The BWRX-300 analysis of AOOs is analogous to the NRC-approved ESBWR where credit for nonsafety-related control and block functions in determining event frequencies and analysis consequences was allowed. BWRX-300 approach is based on a methodology that utilizes a graded approach to defense line design requirements.

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ESBWR and BWRX-300 Comparison

ESBWR	BWRX-300	Comments
The rod block features are in a nonsafety-related control system.	The rod block features are in DL2, a SC3 safety-classified system.	BWRX-300 approach has a formal basis for the DL2 functions based on the Safety Strategy and Fault List Methodology.
Anticipatory trip signals are provided to safety-related protection system from non-safety related I&C control system and use of signals from turbine building components justified (see to ESBWR 15.2.2.7.3 for Closure of All MSIVs).	Anticipatory trip functions are allocated to DL2, a SC3 safety-classified system. BWRX-300 approach puts a high emphasis on DL3 (SC1) independence and diversity.	DL2 sensors are not shared with DL3. BWRX-300 approach to independence and diversity aligns with IAEA guidelines.
CCF of the safety-related digital I&C protection system was addressed using BTP 7-19 methods. AOO and DBA events were evaluated using 'best-estimate' (i.e., more realistic methods and assumptions) to DBA acceptance criteria.	CCF of DL2 is addressed in the BWRX-300 Safety Strategy. AOO event sequence with assumed CCF of DL2 is evaluated to DBA acceptance criteria and mitigated by DL3 , a SC1 safety-classified system due to change in event frequency. CCF of DL3 is addressed in the BWRX-300 Safety Strategy. DBA event sequence with assumed CCF of DL3 is evaluated to DEC acceptance criteria and mitigated by DL4a, a SC2 safety-classified system.	The BWRX-300 Safety Strategy provides a more holistic approach to treatment of CCF than the BTP 7-19 approach and aligns with IAEA guidelines.

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ESBWR and BWRX-300 Comparison

ESBWR	BWRX-300	Comments
Non-safety-related equipment that is diverse from the safety-related protection system or unaffected by the digital I&C CCF was credited to mitigate AOOs and DBAs coincident with digital CCF	The BWRX-300 approach provides SC1 safety-classified equipment to mitigate AOOs coincident with CCF of DL2. The BWRX-300 approach provides SC2 safety-classified equipment to mitigate DBAs coincident with CCF of DL3.	The BTP 7-19 method allows credit for non-safety-related equipment that is diverse from the safety-related protection system or unaffected by the digital I&C CCF. BTP 7-19 only requires mitigation of AOO events coincident with digital protection system CCF using non-safety-related equipment based on "realistic" analysis methods and accident acceptance criteria.
BTP 7-19 approach applies some limited quality assurance requirements to the diverse protection features credited analysis of CCFs based on Generic Letter 85-06, Quality Assurance Guidance for ATWS Equipment That Is Not Safety-Related.	BWRX-300 Classification Methodology applies SC1 design and quality requirements to DL3 equipment, SC2 design and quality requirements to DL4a equipment, and SC3 design and quality requirements to DL2 equipment.	BWRX-300 Classification Methodology applies design and quality requirements to diverse protection features credited in the analysis of CCFs than required by BTP 7-19. BWRX-300 provides stronger defense-in-depth by applying more rigorous design and quality requirements to equipment mitigating CCF events than required by BTP 7-19.

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BWRX-300 Illustrative I&C Example

C10 Reactor Protection Functions and Signals

Safety Class 1

Defense Line 3

inputs are per division

NR – narrow range
WR – wide range
TC – thermocouple
DP – differential pressure

"per IC" means that each IC is initiated or isolated by individual logics

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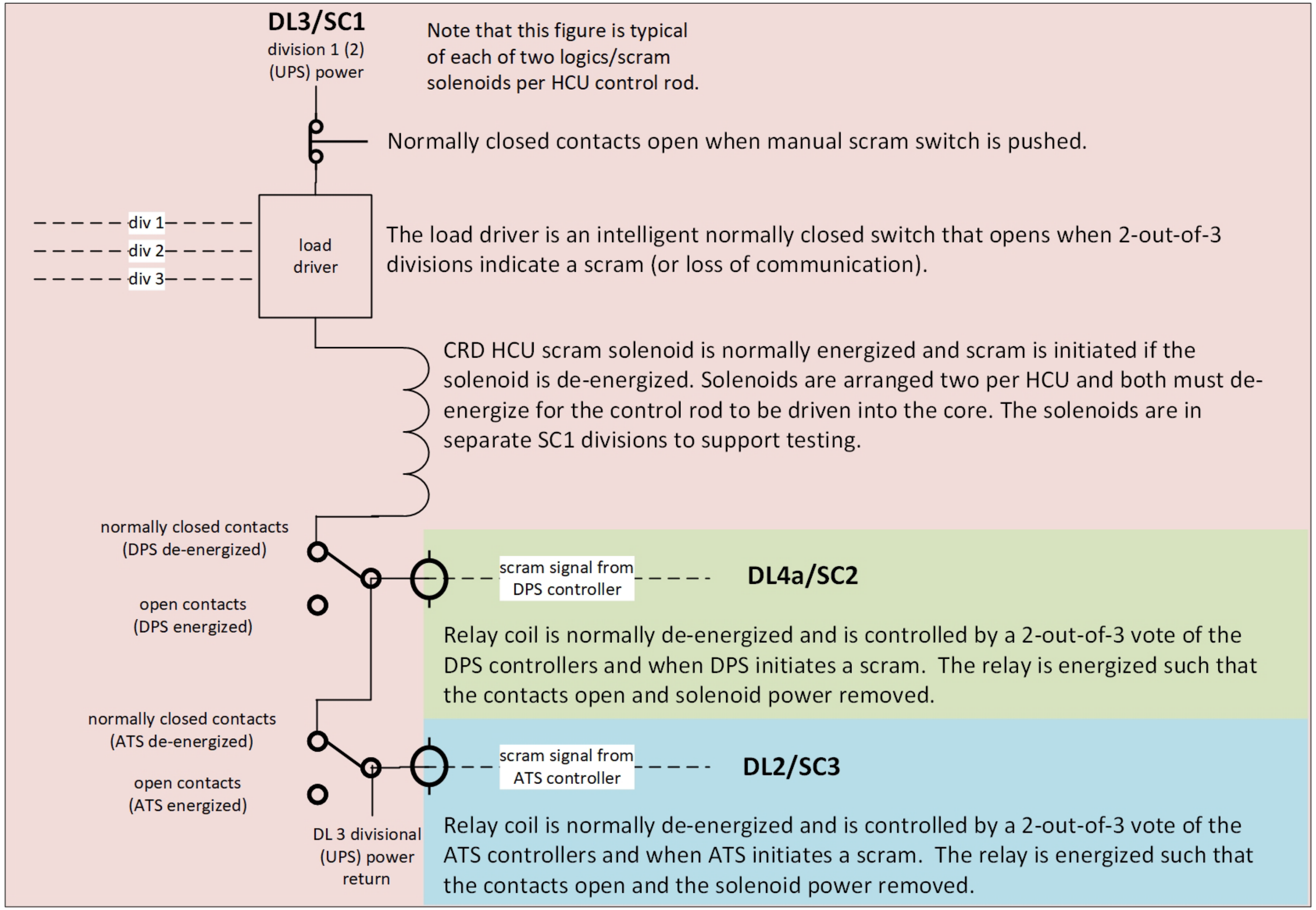
C10 Hydraulic Scram Solenoid Control and Power

Safety Class 1

Defense Line 3



Note:
A similar arrangement is used for reactor and containment isolation valve closure, and IC initiation. A different arrangement for IC reactor isolation valves.

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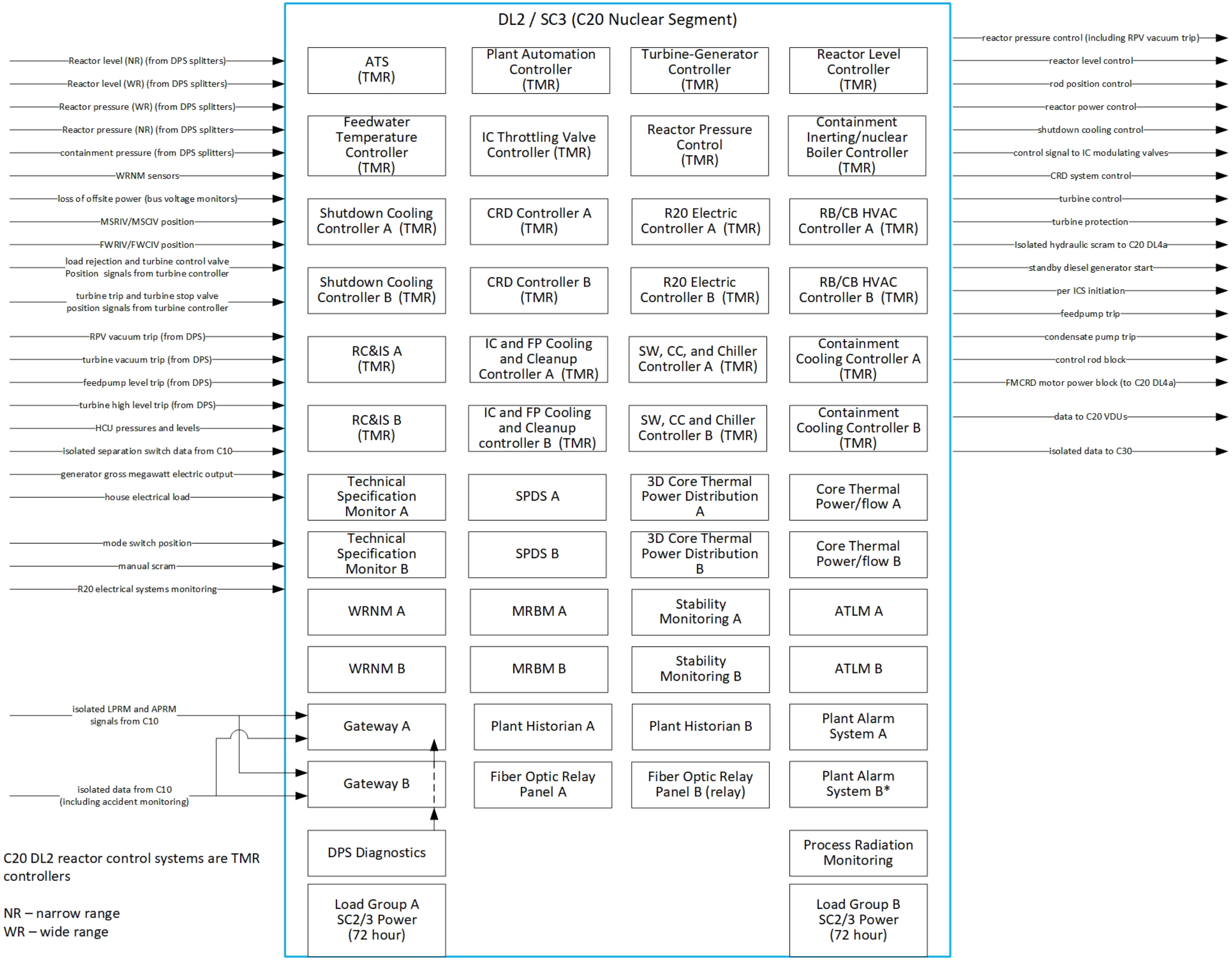
C20 Functions and Signals

Safety Class 3

Defense Line 2



C20 DL2 reactor control systems are TMR controllers

NR – narrow range
WR – wide range

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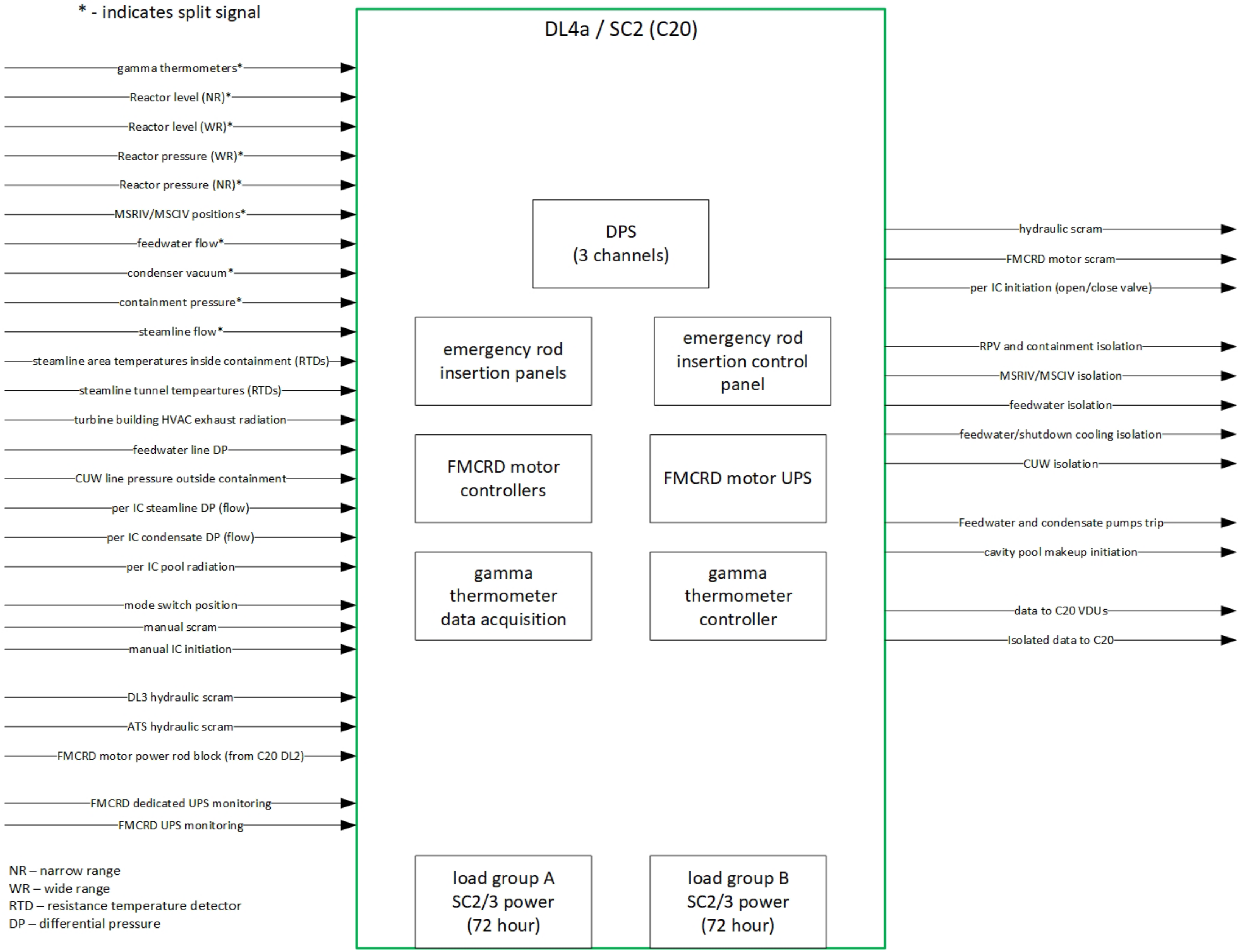
C20 Diverse Protection Functions and Signals

Safety Class 2

Defense Line 4a



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BWRX-300 Illustrative Fault List Methodology Examples

Fault Sequence Example: Loss of Preferred Power (LOPP)

Deterministic Safety Analysis	Functional Defense Line	DL	FSF-Control of Reactivity	DL	FSF-Fuel Cooling	DL	FSF-Long-Term Heat Removal	DL	FSF-Confinement of Radioactive Materials
BL	2	DL2-26	Anticipatory Hydraulic Scram on Low Electric Bus Voltage	DL2-01 DL2-09 DL2-14 DL2-31 DL2-43	Maintain Target Pressure TBV Fast Open on Turbine Trip Demand TBV Closure on High Main Condenser Pressure (HMCP3) ICS Pressure Control on High Reactor Pressure Feedwater Pump Check Valve Closure on Reverse FW Flow		ICS passive operation	DL2-43	Feedwater Pump Check Valve Closure on Reverse FW Flow
	3								
	4a								
	4b								
CN	2						ICS passive operation		
	3	DL3-04	Hydraulic Scram on High Neutron flux	DL3-12 DL3-13 DL3-39	ICS Train 2 Initiation on High RPV Pressure (HP3) ICS Train 3 Initiation on High RPV Pressure (HP4) FW Isolation on Indication of Loss of Normal Feedwater Flow			DL3-39	FW Isolation on Indication of Loss of Normal Feedwater Flow
	4a								
	4b								

Realistic

Conservative

DL2 I&C Implementation

Function		I&C System(s)	Function Inputs
DL2-26	Anticipatory Hydraulic Scram on Low Electric Bus Voltage	(SC3) Anticipatory Trip System	(SC3) Electric Bus Voltage Measurements
DL2-01	Maintain Target Pressure	(SC3) Reactor Control System	(SC2) Reactor Pressure Measurements
DL2-31	ICS Pressure Control on High Reactor Pressure		
DL2-14	TBV Closure on High Main Condenser Pressure		(SC3) Main Condenser Pressure
DL2-09	TBV Fast Open on Turbine Trip Demand	(SC3)Reactor Control System (SC3) T/G Control System	(SC3) Turbine Trip Demand
DL2-43	Feedwater Pump Check Valve Closure on Reverse FW Flow	Not an I&C Function	

Separate SC1 and SC2 pressure sensors

DL3 I&C Implementation

Function		I&C System(s)	Function Inputs
DL3-04	Hydraulic Scram on High Neutron flux	(SC1) Protection System	(SC1) Local Power Range Monitors
DL3-12	ICS Train 2 Initiation on High RPV Pressure (HP3)		(SC1) Reactor Pressure Measurements
DL3-13	ICS Train 3 Initiation on High RPV Pressure (HP4)		
DL3-39	FW Isolation - Loss of Normal Feedwater Flow		(SC1) FW Line Pressure Measurements

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Fault Sequence Example: Feedwater Pump Trip

Deterministic Safety Analysis	Functional Defense Line	DL	FSF-Control of Reactivity	DL	FSF-Fuel Cooling	DL	FSF-Long-Term Heat Removal	DL	FSF-Confinement of Radioactive Materials
BL	2		Inherent, reduction in core flow and core inlet subcooling results in a powe decrease.	DL2-01 DL2-02 DL2-25	Maintain Target Pressure Maintain Target Level Start Standby Feedwater Pump on Loss of Operating Feedwater Pump		Normal		Normal
	3								
	4a								
	4b								
CN	2						ICS passive operation		
	3	DL3-03	Hydraulic Scram on Low RPV level (L3)	DL3-17 DL3-39 DL3-11 DL3-12	MSRIV / MSCIV Isolation on Low RPV Pressure (LP1) FW Isolation on Loss of Normal FW Flow ICS Train 1 Initiation on High RPV Pressure (HP2) ICS Train 2 Initiation on High RPV Pressure (HP3)			DL3-17 DL3-39	MSRIV / MSCIV Isolation on Low RPV Pressure (LP1) FW Isolation on Loss of Normal FW Flow
	4a								
	4b								

Realistic

Conservative

DL2 I&C Implementation

Function		I&C System(s)	Function Inputs
DL2-01	Maintain Target Pressure	(SC3) Reactor Control System	(SC2) Reactor Pressure Measurements
DL2-02	Maintain Target Level		(SC2) Reactor Level Measurements
DL2-25	Start Standby FW Pump on Loss of Operating FW Pump		(SC3) Operating FW Pump status

DL3 I&C Implementation

Function		I&C System(s)	Function Inputs
DL3-39	FW Isolation on Loss of Normal FW Flow	(SC1) Protection System	(SC1) FW Line Pressure Measurements
DL3-03	Hydraulic Scram on Low RPV level (L3)		(SC1) Reactor Level Measurements
DL3-17	MSRIV / MSCIV Isolation on Low RPV Pressure (LP1)		(SC1) Reactor Pressure Measurements
DL3-11	ICS Train 1 Initiation on High RPV Pressure (HP2)		
DL3-12	ICS Train 2 Initiation on High RPV Pressure (HP3)		

Separate SC1 and SC2 pressure and level sensors

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Fault Sequence Example: Loss of Condenser Vacuum

Deterministic Safety Analysis	Functional Defense Line	DL	FSF-Control of Reactivity	DL	FSF-Fuel Cooling	DL	FSF-Long-Term Heat Removal	DL	FSF-Confinement of Radioactive Materials
BL	2	DL2-37 DL2-13	Anticipatory Hydraulic Scram on High Main Condenser Pressure (HMCP1) Turbine Trip on High Main Condenser Pressure (HMCP2)	DL2-01 DL2-02 DL2-09 DL2-14 DL2-31	Maintain Target Pressure Maintain Target Level TBV Fast Open on Turbine Trip Demand TBV Closure on High Main Condenser Pressure (HMCP3) ICS Pressure Control on High Reactor Pressure		ICS passive operation	DL2-14	TBV Closure on High Main Condenser Pressure (HMCP3)
	3								
	4a								
	4b								
CN	2						ICS passive operation		
	3	DL3-04	Hydraulic Scram on High Neutron flux	DL3-23 DL3-11 DL3-12	FW Isolation on High RPV Water Level (L9) ICS Train 1 Initiation on High RPV Pressure (HP2) ICS Train 2 Initiation on High RPV Pressure (HP3)			DL3-23	FW Isolation on High RPV Water Level (L9)
	4a								
	4b								
EX	2	DL2-37 DL2-13	Anticipatory Hydraulic Scram on High Main Condenser Pressure (HMCP1) Turbine Trip on High Main Condenser Pressure (HMCP2)	DL2-01 DL2-02 DL2-09 DL2-14	Maintain Target Pressure Maintain Target Level TBV Fast Open on Turbine Trip Demand TBV Closure on High Main Condenser Pressure (HMCP3)		ICS passive operation	DL2-14 DL2-41	TBV Closure on High Main Condenser Pressure (HMCP3) MSRIV/MSCIV Isolation on High Main Condenser Pressure (HMCP4)
	3								
	4a	DL4a-40 DL4a-41	CRD Fast Motor Run-In on High Flux After Scram Signal FW/CD Pump Trip on High Flux After Scram Signal	DL4a-XX	ICS initiation on High Flux After Scram Signal				
	4b								

Realistic

Conservative

BDBE

DL2 I&C Implementation

Function		I&C System(s)	Function Inputs
DL2-37	Anticipatory Hydraulic Scram on High Main Condenser Pressure	(SC3) Anticipatory Trip System	(SC3) Main Condenser Pressure
DL2-41	MSRIV/MSCIV Isolation on High Main Condenser Pressure		
DL2-14	TBV Closure on High Main Condenser Pressure	(SC3) Reactor Control System	
DL2-01	Maintain Target Pressure		(SC2) Reactor Pressure Measurements
DL2-31	ICS Pressure Control on High Reactor Pressure		
DL2-02	Maintain Target Level		(SC2) Reactor Level Measurements
DL2-09	TBV Fast Open on Turbine Trip Demand	(SC3)Reactor Control System (SC3) T/G Control System	(SC3) Turbine Trip Demand

Separate SC1 and SC2 pressure and level sensors

DL3 I&C Implementation

Function		I&C System(s)	Function Inputs
DL3-04	Hydraulic Scram on High Neutron flux	(SC1) Protection System	(SC1) Local Power Range Monitors
DL3-23	FW Isolation - High RPV Water Level		(SC1) Reactor Level Measurements
DL3-11	ICS Train 1 Initiation on High RPV Pressure (HP3)		(SC1) Reactor Pressure Measurements
DL3-12	ICS Train 2 Initiation on High RPV Pressure (HP4)		

DL4a I&C Implementation

Function		I&C System(s)	Function Inputs
DL4a-40	CRD Fast Motor Run-In on High Flux After Scram Signal	(SC2) Diverse Protection System (SC2) FMCRD Motor Control System	(SC2) Gamma Thermometer Measurements
DL4a-41	FW/CD Pump Trip on High Flux After Scram Signal		
DL4a-xx	ICS Initiation on High Flux After Scram Signal		

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BWRX-300 Conformance With SRP Section 19.3

BWRX-300 Conformance to SRP Section 19.3

The BWRX-300 design process includes the use of both probabilistic and deterministic criteria to achieve the following objectives associated with the Regulatory Treatment of Non-SC1 SSCs:

(1) Determine whether regulatory oversight for SC2 and SC3 SSCs is needed

(2) Identify risk important SSCs for regulatory oversight (if it is determined that regulatory oversight is needed)

(3) Decide on an appropriate level of regulatory oversight for the various identified SSCs commensurate with their risk importance

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BWRX-300 RTNSS Mapping With Safety Strategy

RTNSS Criterion	BWRX-300 Defense Lines (Possible RTNSS Functions and Defense Lines)
A - SSC functions relied upon to meet beyond design basis deterministic NRC performance requirements such as 10 CFR 50.62 for anticipated transient without scram (ATWS) mitigation and 10 CFR 50.63 for station blackout	N/A (DL3 72-hour batteries for coping with SBO are provided) DL4a (DPS scram functions, CRD fast motor run-in) DL2 (ATS functions)
B - SSC functions relied upon to resolve long-term safety (beyond 72 hours) and to address seismic events	Based upon 7-day passive pool cooling, no functions identified to date for this criterion
C - SSC functions relied upon under power-operating and shutdown conditions to meet the Commission's safety goal guidelines of a core damage frequency of less than 1.0E-4 each reactor year and large release frequency of less than 1.0E-6 each reactor year	Some DL4a (e.g., DPS) and possibly DL2 (none identified to date, evaluation on-going) Some DL4b (e.g., containment vent)
D - SSC functions needed to meet the containment performance goal (SECY-93-087, Issue I.J), including containment bypass (SECY-93-087, Issue II.G), during severe accidents	DL4a, DL4b (e.g., containment vent)
E - SSC functions relied upon to prevent significant adverse system(s) interactions	none identified in the design to date; evaluation on-going; any identified are eliminated in the design

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BWRX-300 Conformance to SRP Section 19.3

SECY 94-084

4. Selection of Important Non-Safety (non-SC1) Systems

The designers will determine any combinations of non-safety SSCs that are necessary to meet NRC regulations, safety goal guidelines, and the containment performance goal objectives. The designers will determine these combinations for both scope Criteria A and E where NRC regulations are the bases for consideration and scope Criteria C and D where PRA methods are the bases for consideration. To address the long-term safety issue in scope Criterion B, the designer will use PRA insights, sensitivity studies, and deterministic methods to establish the ability of the design to maintain core cooling and containment integrity beyond 72 hours. Non-safety SSC functions required to meet beyond design basis requirements (Criterion A), to resolve the long-term safety and seismic issues (Criterion B), and to prevent significant adverse interactions (Criterion E) are subject to regulatory oversight

For BWRX-300, "Nonsafety-related systems" include:

- SC2 and SC3
- SCN

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BWRX-300 Conformance to SRP Section 19.3

Criterion A – BWRX-300 design meets ATWS and SBO Regulations with DL2, DL3 and DL4a functions:

> "the staff finds use of the risk goal for the BWRX-300, as described in NEDC-33912, of P(ATWS) less than 1x10-5 per reactor year acceptable because it is consistent with the intent of 10 CFR 50.62 as described in SECY 83-293 and NUREG-1780."

- Limitation and Condition 5.1

5.1 Reliability analysis or testing, considering applicable operating experience and expected load follow conditions, of the BWRX-300 diverse scram features to demonstrate the probability of an ATWS is less than 1x10-5 per reactor year without the need for a SLCS

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BWRX-300 Conformance to SRP Section 19.3

Station Blackout

BWRX-300 is designed to mitigate an SBO event to meet the requirements of 10 CFR 50.63 using DL3 functions

- No operator actions or AC power are required for a SBO event for 72 hours
- Reactor water level is maintained above the top of active fuel by reactor isolation and operation of the ICS
- Containment integrity (pressure and temperature within design limits) is maintained by Passive Containment Cooling System (PCCS)

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BWRX-300 Conformance to SRP Section 19.3

Criterion B. Based upon 7-day passive pool cooling, no functions identified to date for this criterion

Seismic PRA is in progress

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BWRX-300 Conformance to SRP Section 19.3

Criterion C – PRA used to evaluate whether systems required to meet the NRC safety goals of CDF less than 1.0 E-4 per year and LRF less than 1.0 E-6 per year.

- PRA studies encompass at-power and shutdown modes for internal and external events and retain the same initiating event frequencies as the baseline PRA models
- PRA uses the internal and external event models to quantify the effects of SSCs on the safety goal guidelines
- In lieu of a focused PRA, the BWRX-300 uses the full PRA and risk-importance measures to identify risk-significant components

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BWRX-300 Conformance to SRP Section 19.3

Criterion D – Containment Performance Assessment

The containment performance goal in SECY-93-087, Issue I.J including containment bypass (SECY-93-087, Issue II.G), during severe accidents. This performance criteria is evaluated in the PRA to identify any applicable SSCs. BWRX-300 design includes the containment vent that would be applicable to this criterion

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BWRX-300 Conformance to SRP 19.3

Criterion E – Assessment of Significant Adverse Interactions

Systems interactions are usually well recognized and accounted for in the design and within the PRA model. However, there is the potential for unrecognized subtle dependencies among the various SSCs that could be significant. The term used to describe such dependencies is adverse systems interaction (ASI). It is broadly applied in terms of functional interactions, spatial interactions and human acts of commission. Such interactions are RTNSS candidates under Criterion E

- Independent defense lines in the design results in the inapplicability of this criterion to the BWRX-300 design
- Deterministic protection provisions are included in the Isolation Condenser System (ICS) (the safety-related Emergency Core Cooling System (ECCS)
- As part of the PRA inputs to the design programs and processes, potential ASI discovered during the construction of the PRA are assessed for applicability to the RTNSS process under Criterion E
- ASI exists if the action or condition of an active, interfacing system causes a loss of safety function of a passive safety-related system. A systematic process is used to analyze specific features and actions that are designed to prevent postulated adverse interactions, while taking into consideration the extensive operating experience that has been used in the current design criteria to prevent adverse systems interaction

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BWRX-300 Conformance to SRP Section 19.3

Conclusion:

Special treatments are applied to Safety Class 2 and 3 components that meet the RTNSS criteria consistent with SRP 19.3 acceptance criteria. As a result, the Safety Classifications are used in lieu of the RTNSS classifications.

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BWRX-300 SRP Section 19.3 Suggested Wording for Evaluation Findings

IV. EVALUATION FINDINGS

1. The staff concludes that the applicant's process using full PRA identified risk-significant SSCs meets the intent of the regulatory treatment of nonsafety-related equipment approved by the NRC and is therefore acceptable.
2. Based on its review, the staff concludes that the applicant has an acceptable process where non-SC1 equipment relied upon to meet beyond design basis deterministic NRC performance requirements is identified, and special treatments applied.

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BWRX-300 SRP Section 19.3 Suggested Wording for Evaluation

Findings

3. The staff finds that the applicant can identify SC2 and SC3 equipment ensuring that safety functions relied upon in the post-72-hour period and following seismic events have a reasonable likelihood of being successful.

Further, the staff finds that the SC2 and SC3 equipment relied upon in the post-72-hour period is designed in accordance with Commission policy and that the applicant has established appropriate availability controls (ACs) for this equipment.

4. The staff has reviewed the mission statements for SSCs including R/A missions. These statements correctly describe the missions of safety-classified SSCs that meet the intent of RTNSS criteria, and R/A missions are consistent with assumptions in the PRA; therefore, the staff finds them acceptable.

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BWRX-300 SRP Section 19.3 Suggested Wording for Evaluation

Findings

5. The applicant proposed a means for implementing controls on SC2 and SC3 SSCs that meet the RTNSS criteria in the form of administrative controls (ACs) for the SSCs as discussed in the SER Section. The availability controls manual (ACM) will be incorporated into the FSAR and documents the ACs.

6. The staff has reviewed the applicant's implementation of the SSC safety classification for consistency with the guidance in SRP 19.3 and determined that it envelopes the scope, criteria, and process described in SECY-94-084 and associated SRM, SECY-95-132 and associated SRM, and RG 1.206. Therefore, the staff finds the applicant's implementation approach to be acceptable.

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BWRX-300 Conformance With SRP Sections 3.1-3.2 Classification of SSCs

SRP 3.2.1 - Seismic Category Conformance

Seismic Category reflects SSC requirements during and after a seismic event and how the SSC is seismically designed and qualified. Seismic Category is assigned based on the SRP 3.2.1 and RG 1.29 regulatory guidance:

1. DL3 (Safety Class 1 – "Safety-Related") functions are credited with remaining operable during and after a seismic event associated with an SSE. Accordingly, SSCs that perform or support DL3 functions are categorized as Seismic Category I. Other SSC that are classified as Safety Class 1 are categorized as Seismic Category I.
2. Radioactive waste building that houses the management and storage of radiological materials is categorized as Seismic Category RW-IIa as noted in LTR NEDO-33914-A and consistent with Regulatory Guide (RG) 1.143. This RW-IIa structure is seismically qualified for one-half of the site-specific SSE based upon the consequences of failure, one-half of the site-specific SSE is justified as it would bound the ground motion spectra for seismic categories identified in ASCE/SEI 43 for SSCs used for handling and storage of highly radioactive materials.
3. Seismic Category II - SSCs whose failure during a seismic event could adversely affect the ability of any Seismic Category I SSC to accomplish its safety class function are classified as Seismic Category II.

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SRP 3.2.1 - Seismic Category Conformance (cont.)

3. The BWRX-300 complies with the requirements of GDCs 1, 2, and 60, 10 CFR Part 100, Appendix A, and 10 CFR Part 50, Appendix S as it relates to Safety Class 1 SSCs being designed to withstand the SSE and remain functional.
4. The BWRX-300 conforms to the following Regulatory Guides:
   - RG 1.29 establishes the seismic design classification to meet the requirements of GDC 2, 10 CFR Part 100, Appendix A, and 10 CFR Part 50, Appendix S.
   - RG 1.151 provides guidance regarding the seismic design provisions and classification of safety-related instrument sensing lines.
   - RG 1.143 provides guidance used to establish the seismic design provisions of radioactive waste management SSCs to meet the requirements of GDC 2 and GDC 60 as they relate to designing these SSCs to withstand earthquakes.
   - RG 1.189 provides guidance used to establish the design provisions of fire protection SSCs, including seismic design consideration and seismic classification for certain SSCs.

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SRP 3.2.2 - System Quality Group Classification

- BWRX-300 design utilizes a Quality Group designation per the guidance in RG-1.26 as a method for establishing the appropriate codes and standards based on the importance of the pressure-retaining function of the component. Items are classified as Quality Group A, B, C or D. The guidance and classification method are used with some clarification based on the unique design of the BWRX-300.
- Quality Group A, B, and C are SC1.
- Except for the Isolation Condenser System (ICS), piping between the outboard Reactor Pressure Vessel Isolation Valves (RIVs)[1] and the seismic restraints outboard of the Containment Isolation Valves (CIVs)[2] is Quality Group B:
  - Piping outboard of the seismic restraint outboard of the CIVs is Quality Group D except for ICS, CRD Hydraulic or areas of break exclusion
  - Pressure-retaining components upstream of the flow restrictor in the ICS are Quality Group A

Notes:

1. NEDC-33910P-A, *BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection*
2. NEDC-33911P-A, *BWRX-300 Containment Performance*

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SRP 3.2.2 -System Quality Group Classification (cont.)

GDC 1 and 10 CFR 50.55a require that systems and components be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. Regulations in 10 CFR 50.55a also incorporate by reference the applicable editions and addenda of the ASME BPV Code, which addresses pressure integrity of components.

An exemption to 10 CFR 50.55a is proposed for the BWRX-300 design where the latest 2021 ASME codes and standards will be used rather than the later editions of codes and standards that are currently allowed in 10 CFR 50.55a.

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BWRX-300 Conformance With SRP 15.0

Conformance with SRP 15.0

SRP 15.0 Conformance

- I.1.A. Categorization of Transients and Accidents
- I.1.B. Categorization According to Type
- I.2.A. AOO Analysis Acceptance Criteria
- I.2.B. Analysis Acceptance Criteria for Postulated Accidents
- I.3. Plant Characteristics Considered in the Safety Evaluation
- I.4 Assumed Protection and Safety System Actions
- I.5 Evaluation of Individual Initiating Events
- I.6.A. Identification of Causes and Frequency Classifications
- I.C. Core, System, and Barrier Performance

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Conformance with SRP 15.0

II. Acceptance Criteria Conformance

- GDCs 2, 4, 5, 13, 31, 34, 35, 60 and 61 – BWRX-300 conforms
- GDC 15 – PDC 15 proposed
- GDC 25 – PDC 25 proposed
- GDC 26 – PDC 26 proposed for Reactivity control redundancy and capability consistent with relief allowed by the Safety Evaluation to Licensing Topical Report (LTR) NEDC-33912P-A, R1
- GDC 28 – exemption requested as discussed in NEDC-33912P-A, R1 where the Safety Evaluation allowed for an exemption request to justify that the Control Rod Drop Accident was a beyond design basis accident that is documented in the Probability Risk Assessment
- GDC 55 – conforms as “other defined basis” as allowed by the Safety Evaluation to NEDC-33910P-A, R3, BWRX-300 Containment Performance

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Conformance with SRP 15.0

SRP 15.0 Exceptions and Exemptions

- I.6.B. Sequence of Events and System Operation – GEH is taking an exemption from components mitigating AOOs meeting the definition of "safety-related" per 10 CFR 50.2 as allowed by the SRP guidance technical justification for using nonsafety-related systems to mitigate AOOs
  - Technical justification will be provided for operation of DL2 (SC3) components (nonsafety-related) for mitigating AOOs
  - DL2 functions are assigned appropriate quality and reliability measures to ensure functional performance independently from DL3 (SC1) functions
  - Any DL2 common cause failure (CCF) is mitigated and performed diversely by a DL3 (SC1) function
  - DL2 functions protect the SSCs and prevent event progression to less frequent DBAs
  - DL2 functions include anticipatory trips, control rod blocks and maintaining target level, power and pressure
- Exemptions/Proposed PDCs– GDCs 10, 12, 17 (provided in a future licensing submittal), 19, 20, 27 and 29

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Proposed BWRX-300 Principal Design Criteria

BWRX-300 Principal Design Criteria

Definitions and Explanations

Safety-Classified - A classification generally applied to systems, structures, and components (SSCs) that perform functions designed to mitigate event sequences and that have been classified as Safety Class 1, 2, or 3 using a graded approach that accounts for the importance of the SSCs in a defense-in-depth (D-in-D) strategy. Safety-classified SSCs are designed using a graded approach intended to ensure a level of reliability consistent with the assumptions in the plant safety analyses.

Reactivity Control Systems - Reactivity control systems are a group of safety-classified control systems designed to perform the fundamental safety function of reactivity control during all modes of plant operations.

Actuation/Trip Systems- Actuation/trip systems are a group of safety-classified protection systems designed to perform the fundamental safety function of initiating reactor trips and engineered safety features actuations during all modes of plant operations.

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BWRX-300 Principal Design Criteria

Proposed PDCs

PDC 10 Reactor design. The reactor core and associated coolant, safety-classified reactivity control actuation/trip ~~protection~~ systems shall be designed with appropriate margin to assure that the specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operation occurrences.

PDC 12 Suppression of reactor power oscillations. Suppression of reactor power oscillations. The reactor core and associated coolant, safety-classified reactivity control, and actuation/trip ~~protection~~ systems shall be designed to assure that power oscillations which can result in conditions exceeding specified fuel design limits are not possible or can be reliably and readily detected and suppressed.

PDC 15 Reactor coolant system design. The reactor coolant system and associated auxiliary, safety-classified reactivity control, actuation/trip ~~protection~~ systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

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BWRX-300 Principal Design Criteria

Proposed PDCs

PDC 19 Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under design basis accident conditions, including loss-of-coolant accidents.

Adequate radiation protection shall be provided to permit access and occupancy of the control room under design basis accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent as defined in 10 CFR 50.2 ~~whole body, or its equivalent to any part of the body, for the duration of the accident~~.

Equipment at appropriate locations outside the control room shall be provided ~~(1)~~ with a design capability for ~~prompt hot~~ safe shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe shutdown condition ~~during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures~~."

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BWRX-300 Principal Design Criteria

Proposed PDCs

PDC 20 Reactivity control, actuation/trip ~~protection~~ system functions. The safety-classified reactivity control, actuation/~~and~~trip ~~protection~~ systems shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of safety-classified systems and components ~~important to safety~~."

PDC 21 Actuation/trip ~~Protection~~ system reliability and testability. The actuation/trip ~~protection~~ systems shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed.

PDC 25 Reactivity control, actuation/~~and~~ trip protection systems for reactivity control malfunctions. The safety-classified reactivity control, actuation/trip ~~protection~~ systems requirements for reactivity control malfunctions shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

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BWRX-300 Principal Design Criteria

Proposed PDCs

PDC 26 Reactivity control system redundancy and capability.

1. The control rods shall have appropriate margin to accommodate malfunctions such as stuck rods to assure a) specified acceptable fuel design limits design limits are not exceeded and shutdown is achieved and maintained during normal operation including anticipated operational occurrences and b) shall be capable of holding the reactor core subcritical under cold conditions.
2. A means of inserting negative reactivity shall be provided using a diverse motive force at a sufficient rate and amount to assure that the design limits for the fission product barriers are not exceeded during normal operation including anticipated operational occurrences.
3. A means with a diverse motive force shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the specified acceptable fuel design limits for the fission product barriers are not exceeded.

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BWRX-300 Principal Design Criteria

Proposed PDCs

PDC 27 ~~Combined~~ Reactivity control systems capability - 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.

PDC 28 Reactivity limits. The safety-classified actuation/trip and 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 drop (unless prevented by positive means) ~~out~~, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

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BWRX-300 Principal Design Criteria

Proposed PDCs

PDC 29 Protection against anticipated operational occurrences. The safety-classified actuation/ trip ~~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.

PDC 37 Testing of emergency core cooling system. 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 actuation/trip ~~protection~~ systems, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

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Technical Specifications Related to Defense Lines

Technical Specifications and Relation to Defense Levels

Defense Line	Type of Controls*
2	Availability Controls, as applicable
3	Technical Specifications
4a	Availability Controls, as applicable
4b	Availability Controls, as applicable
*Risk significant DL functions that are not SC1 will need to be in TS when identified per SRP 19.3 and 10 CFR 50.36(c)(2)(ii)(D)	

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Technical Specifications 10 CFR 50.36 (c)(2)(ii) Compliance

No.	Criterion	Compliance
1	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary	Instrumentation will be installed to meet General Design Criterion 30 and an LCO will require the instrumentation to be OPERABLE
2	A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.	LCOs will be established to include initial conditions of design basis accidents consistent with Chapter 15
3	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.	LCOs will require Defense Line 3 SSCs (SC1) be operable to mitigate the consequences of design basis accident analyzed in Chapter 15
4	A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety	LCOs are established for safety-classified SSCs that are determined to be risk-significant using risk importance measures such as Risk Achievement Worth (RAW) and Fussell-Vesely (F-V)

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BWRX-300 Conformance With SRP Sections 15.0.1-15.9

Conformance with SRP 15.0.1 Radiological Consequences Using Alternative Source Terms

SRP 15.0.1 Conformance and Exceptions/Exemptions:

- Exemption request and proposed PDC-19 (see Proposed PDCs)
- Exception to NUREG-0737, Item II.B.3 Post-Accident Sampling – BWR Owner's Group Topical Report NEDO-32991-A, "Regulatory Relaxation for BWR Post-Accident Sampling Stations (PASS)" August 2001 is applied to the BWRX-300. The three commitments from the Safety Evaluation (capability for classifying fuel damage events at ALERT level, contingency plan for obtaining and analyzing reactor coolant samples, and maintaining an I-131 site survey detection capability) will be fulfilled in a future licensing submittal
- Table 1 Accident Dose Criteria for LOCA – RG 1.183, Revision 0 forms the basis for calculating source terms for Fuel Handling Accident, Offgas failure release and LOCA breaks outside containment that bound all DBA breaks inside containment; however, in determining the EPZ for a BDBA, the limiting core melt scenario identified from the Level 1 PRA will be determined using a mechanistic source term using the guidance of RG 1.242, and the EPZ will meet the proposed ruling requirements from 10 CFR 50.160

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Conformance with SRP 15.0.2 Review of Transient and Accident Analysis Method

- RG 1.203 - Conformance provided in Safety Evaluation to GEH LTR NEDC-33922P-A, Revision 3, “BWRX-300 Containment Evaluation Method” Section 3.4.2 and Section 5.2.9
- II. Acceptance Criteria 10 CFR 50, Appendix B – See conformance in Safety Evaluation to LTR NEDC-33922P-A, Rev. 3
- II. Acceptance Criteria 10 CFR 50, Appendix K – See conformance in Safety Evaluation to LTRs NEDC-33911P-A, Rev. 3 Section 5.3.3 and NEDC-33910P-A, Rev. 0, Supplement 1, Section 4.1.2
- II.2 Evaluation Model – TRACG, GOTHIC and PANAC11 codes were reviewed and approved in LTR NEDC-33922P-A, Rev. 3; ANSI/ANS-18.1, RADTRAD and PAVAN have been approved for use in ESBWR Design Control Document Tier 2, 26A6642BP, Rev. 10, April 2014

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Conformance with SRP 15.1.1-15.1.4 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow , and Inadvertent Opening of a Steam Generator Relief or Safety Valve

Conforms with the following exceptions for II. Acceptance Criteria:

- Exemption requested with proposed PDC-10 for GDC-10 (see Proposed PDCs)
- Exemption requested with proposed PDC-20 for GDC-20 (see Proposed PDCs)
- Exemption requested with proposed PDC-26 for GDC-26 (see Proposed PDCs)

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Conformance with SRP 15.2.1-15.2.5 Loss of External Load; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)

Conforms with following exceptions:

- Exemption requested with proposed PDC 10 for GDC 10 (see Proposed PDCs)
- Exemption requested with proposed PDC 17 for GDC 17 that will be provided in a future licensing submittal
- Exemption requested with proposed PDC 26 for GDC 26 (see Proposed PDCs)

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Conformance to SRP 15.2.6 Loss of Nonemergency AC Power to the Station Auxiliaries

Conforms with the following exceptions:

- Exemption requested with proposed PDC 10 for GDC 10 (see Proposed PDCs)
- Exemption requested with proposed PDC 26 for GDC 26 (see Proposed PDCs)

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Conformance to SRP 15.2.7 Loss of Normal Feedwater Flow

Conforms with the following exceptions:

- Exemption requested with proposed PDC 10 for GDC 10 (see Proposed PDCs)
- Exemption requested with proposed PDC 17 for GDC 17 that will be provided in a future licensing submittal
- PDC 26 proposed
- II.6 TMI Action Plan Item II.K.2.19 – not applicable to BWRX-300 design

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Conformance with SRP 15.3.1-15.3.2 Loss of Forced Reactor Coolant Flow Including Trip or Pump Motor and Flow Controller Malfunctions

Not applicable to BWRX-300 with no forced cooling , passively cooled plant

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Conformance with SRP 15.4.1 Uncontrolled Control Rod Assembly Withdrawal From a Subcritical or Low Power Startup Condition

Conforms with following exceptions:

- Exemption requested with proposed PDC 10 for GDC 10 (see Proposed PDCs)
- Exemption requested with proposed PDC 17 for GDC 17 that will be provided in a future licensing submittal
- Exemption requested with proposed PDC 20 for GDC 20 (see Proposed PDCs)

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Conformance with SRP 15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

Conforms with following exceptions:

- Exemption requested with proposed PDC 10 for GDC 10 (see Proposed PDCs)
- Exemption requested with proposed PDC 17 for GDC 17 that will be provided in a future licensing submittal
- Exemption requested with proposed PDC 20 for GDC 20 (see Proposed PDCs)

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Conformance to SRP 15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

- Exemption requested from GDC 28 that the CRDA is a beyond design basis event in accordance with the CRDA analysis performed in the PRA per Section 5.2 of the Limitations and Conditions provided in the Final Safety Evaluation For GE-Hitachi Licensing Topical Report NEDC-33912P, Revision 0, “BWRX-300 Reactivity Control” dated January 12, 2021
- Exemption requested with proposed PDC 10 for GDC 10 (see Proposed PDCs)
- Exemption requested with proposed PDC 20 for GDC 20 (see Proposed PDCs)

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Conformance to SRP 15.4.4-15.4.5 Startup of An Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWRX Core Flow Rate

Conforms with the following exceptions:

- Exemption requested with proposed PDC 10 for GDC 10 (see Proposed PDCs)
- Exemption requested with proposed PDC 20 for GDC 20 (see Proposed PDCs)
- Exemption requested with proposed PDC 26 for GDC 26 (see Proposed PDCs)

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Conformance With SRP 15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

Conforms with No Exceptions

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Conformance with SRP 15.4.9 Radiological Consequences of Control Rod Drop Accident (BWR)

Exemption requested to 10 CFR 100 that the CRDA is a beyond design basis event in accordance with the CRDA analysis performed in the PRA. This exemption request is discussed in Section 5.2 of the Limitations and Conditions provided in the Final Safety Evaluation For GE-Hitachi Licensing Topical Report NEDC-33912P, Revision 0, “BWRX-300 Reactivity Control” dated January 12, 2021

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Conformance with SRP 15.5.1-15.5.2 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

Conforms with the following exceptions:

- Exemption requested with proposed PDC 10 for GDC 10 (see Proposed PDCs)
- Exemption requested with proposed PDC 26 for GDC 26 (see Proposed PDCs)

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Conformance With SRP 15.6.1 Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or BWR Pressure Relief Valve

This SRP is not applicable to the BWRX-300 SMR design that does not require pressure relief valves or a suppression pool for pressure relief.

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Conformance to SRP 15.6.2 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

Conforms with exception to:

- GDC 55 – conforms as "other defined basis" as allowed by the Safety Evaluation to NEDC-33910P-A, R3, BWRX-300 Containment Performance

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Conformance to SRP 15.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment

Conforms

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Conformance to SRP 15.6.5 Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

- Conforms based upon audit and extensive review resulting in Final Safety Evaluation for GE-Hitachi Licensing Topical Report NEDC-33922P, Revision 2, “BWRX-300 Containment Evaluation Method” ADAMs Accession Nos. ML22110A008 (letter) and ML22004A080 (Non-Public) Section 8.0
  - Safety Evaluation Limitations and Conditions: 7.1 Isolation Condenser Radiolytic Gas Removal; 7.2 Isolation Condenser Return Line Design and Further Demonstration of Transient Reactor Analysis Code General Electric Modeling Capability; 7.3 Demonstration of No Safety-Significant Break Flow Reversal During the First 72 Hours Into the Event; 7.4 Demonstration of the Applicability of the BWRX-300 Containment Evaluation Method to the Final Passive Containment Cooling System Licensing Basis Analysis
- II.1 10 CFR 50.46 Acceptance Criteria - Conforms based upon the Final Safety Evaluation for GE-Hitachi Licensing Topical Report NEDC-33910P. Revision 0, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection” ADAMS Accession No ML20310A153, Section 4.1.2
- II.3 GDC-35 Acceptance Criteria - Conforms based upon the Final Safety Evaluation for GE-Hitachi Licensing Topical Report NEDC-33910P. Revision 0, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection” ADAMS Accession No ML20310A153, Section 4.1.12
- II.4 10 CFR 100 Acceptance Criteria - Conforms based upon the Final Safety Evaluation for GE-Hitachi Licensing Topical Report NEDC-33910P, Revision 0, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection” ADAMS Accession No ML20310A153, Section 4.1.2.6

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Conformance with SRP 15.6.5, Appendix B, Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Features Outside Containment

- BWRX-300 design does not use containment spray, low pressure safety injection or high-pressure safety injection systems as engineered safety features, relying upon the isolation condenser system to mitigate pressure and temperature in the reactor from LOCAs
- The BWRX-300 isolation condenser system is part of the RCPB as discussed in LTR NEDC-33911P-A, Revision 3, BWRX-300 Containment Performance" and Final Safety Evaluation for GE-Hitachi Licensing Topical Report NEDC-33911P, Revision 0, Supplement 1, "BWRX-300 Containment Performance" ADAMS Accession Nos. ML21070A353 (letter) and ML21008A370 (Non-Public) Section 2.1.2
- SE Section 5.1.22 states: "Specifically, the NRC staff finds the ICS and FMCRD containment isolation meeting GDC 55 on the "other defined basis" to be acceptable. However, this design, using a closed loop outside containment instead of an outboard CIV, could introduce a postulated LOCA outside containment that must be evaluated in future licensing activities. The NRC staff will conduct a detailed evaluation to confirm that GDC 55 is satisfied when it receives an application for a BWRX-300

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Conformance to SRP 15.7.4 Radiological Consequences of Fuel Handling Accident and 15.7.5 Spent Fuel Cask Drop Accident

Conforms

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Conformance to SRP 15.8 Anticipated Transients Without Scram

- Due to the frequency of this event, this is a BDBA event that is addressed in LTR NEDC-33912P-A, Revision 1, “BWRX-300 Reactivity Control”
- II.1 10 CFR 50.62 Acceptance Criteria - Safety Evaluation Limitation and Condition 5.1 requires reliability analysis or testing, considering operating experience and expected load follow conditions of the BWRX-300 diverse scram features to demonstrate the probability of an ATWS is less than 1E-05 per reactor –year without the need for a SLCS
- II.2 10 CFR 50.46 Acceptance Criteria - SE Section 3.7.1 allows for demonstrating the effectiveness for ATWS events consistent with SRP Section 15.8 item (1) acceptable or alternatively use a probabilistic approach for demonstrating the reliability of the diverse scram system
- II.3 GDC -12 Acceptance Criteria - SE Limitation and Condition 5.3 requires a stability analysis in accordance with an approved methodology to demonstrate that the BWRX-300 maintains a coupled power-flow response such that any operational perturbation, maneuver or AOO does not cause an immediate scram is naturally damped and decays quickly to steady state for all modes of operation, preventing the SAFDLS from being exceeded. A further exemption request from GDC -12 using proposed PDC-12 previously discussed in the Principal Design Criteria
- GDC-27 - Exemption request with justification for the PDC 27 discussed previously in the PDCs
- GDC-28 – Exemption to GDC 28 for the control rod drop accident will be provided in future licensing activities that meets the Section 5.0 Limitations and Conditions

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Conformance to SRP 15.9 Boiling Water Reactor Stability

Conforms with following exceptions:

- Exemption requested with proposed PDC-10 for GDC-10 (see Proposed PDCs)
- Exemption requested with proposed PDC-12 for GDC-12 (see Proposed PDCs) ; in the SE for LTR NEDC-33912P-A, R1, Section 4.1.2, the NRC staff will conduct a detailed evaluation of the thermal-hydraulic codes, methods and analysis results used to demonstrate that GDC-12 is met without the need for a special stability detection and trip system for all modes of operation, including startup
- Exemption requested with proposed PDC-20 for GDC-20 (see Proposed PDCs); in the SE for LTR NEDC-33912P-A, R1, Section 4.1.3, "the staff finds the approach for the RPS, as described in NEDC-33912, consistent with GDC 20 and, therefore acceptable."
- Exemption requested with proposed PDC-29 for GDC-29 (see Proposed PDCs); in the SE for LTR NEDC-33912P-A, R1, Section 4.1, "The staff notes that the BWRX-300 design, as described in NEDC-33912, contains an [[ATS to provide an early diverse scram; however, this system is not safety-related,]] and NEDC-33912 does not describe the design and quality requirements applied to the system. Therefore, the staff bases its finding on the approach described above related to the safety-related RPS hydraulic scram function. The staff finds this approach consistent with GDC 29 and, therefore, acceptable.

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Conclusions

Conclusions

- TS will be established using Safety Class 1 SSCs and risk significant based on risk importance measures
- ESBWR credited nonsafety-related control and block functions in determining event frequencies and analysis consequences
- BWRX-300 will apply special treatments to SC2 and SC3 SSCs
- With a stronger defense-in-depth, BWRX-300 establishes layers with signals and actuation/trip systems that are independent

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Conclusions

Exemptions and Proposed PDCs

- An exemption to 10 CFR 50.55a will be provided in future licensing activities for the use of ASME 2021 codes in lieu of the approved ASME codes currently allowed by 10 CFR 50.55a
- Proposed new Definitions to 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants for Safety-Classified, Reactivity Control System and Actuation/Trip Systems provided in LTR NEDC-33934P BWRX-300 Safety Strategy
- Exemptions sought and proposed PDCs for 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants GDCs 10, 12, 15, 20, 21, 25, 27, 28, 29, and 37 for the use of safety-classified reactivity control, actuation and trip systems (DL2, DL3, DL4a and DL4b) in lieu of safety-related system(s); relief is sought to use DL2 (SC3) functions for protecting and bringing the plant to a steady state should an AOO occur that will be provided in LTR NEDC-33934P BWRX-300 Safety Strategy
- Exemption sought and PDC proposed for GDC 17 Electric power systems that will be provided in a future licensing activity

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Conclusions

Exemptions and Proposed PDCs

- Exemption to 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, GDC 19 Control room and proposed PDC 19 that would apply to design basis events where the main control room (MCR) habitability is not a safety function and the dose criteria in PDC 19 can be met with systems that are not “safety-related”. No operator actions are necessary for protecting and controlling AOOs and no operator actions are necessary for mitigating a DBA. If the MCR becomes inhabitable (fire, malevolent act, aircraft impact, BDBE, etc.), the operators could evacuate and there is no impact on design basis accident management if operators are not available in the MCR. This exemption and proposed PDC will be provided in a future licensing activity.
- Exemption to 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, GDC 26 Reactivity control and redundancy and PDC proposed consistent with relief allowed by the Safety Evaluation to LTR NEDC-33912P-A, Revision 1.

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Conclusions

Exemptions and Proposed PDCs

- Exemption sought to GDC 28 and 10 CFR 100 for the Control Rod Drop Accident as allowed by the Safety Evaluation to LTR NEDC-33912P-A, R1 BWRX-300 Reactivity Control that will be provided in a future licensing activity
- Exception to SRP 15.0.1 Radiological Consequences Using Alternative Source Terms where RG 1.183, Revision 0 Alternative Radiological SOURCE Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors will be used to access Fuel Handling Accident, Offgas System Failure Release and LOCAs outside containment. RG 1.242, Revision 0, "Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities" along with the proposed regulation 10 CFR 50.160 "Emergency preparedness for small modular reactors, non-light-water reactors, and non-power production or utilization facilities" will be used in developing the source term associated with the limiting and most probable beyond design basis accident core melt and establishing the emergency planning zone. This will be provided in LTR NEDC-33913P "BWRX-300 Severe Accident Management and Source Term Methodology"
- Special treatments are applied to SC2 and SC3 components that meet the RTNSS criteria consistent with SRP 19.3 acceptance criteria. As a result, the safety classifications are used in lieu of the RTNSS classifications.

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Questions or Comments

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