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BWRX-300 UK Generic Design Assessment (GDA) Chapter 15.2 – Safety Analysis – Identification, Categorisation, and Grouping of Postulated Initiating Events and Accident Scenarios

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EXECUTIVE SUMMARY

This document is Chapter 15.2, Identification, Categorisation, and Grouping of Postulated Initiating Events and Accident Scenarios, of the Preliminary Safety Report of the GEH BWRX-300 for the purposes of UK Generic Design Assessment. It presents the approach used to identify, categorise, and group the Postulated Initiating Events used in the safety analyses presented in the rest of Chapter 15.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ALARP	As Low As Reasonably Practicable
AOO	Anticipated Operational Occurrence
ATLM	Automatic Thermal Limits Monitor
ATS	Anticipatory Trip System
BL-AOO	Baseline Anticipated Operational Occurrence
BL-DBA	Baseline Design Basis Accident
BL-DSA	Baseline Deterministic Safety Analysis
CCF	Common Cause Failure
CIV	Containment Isolation Valve
CN-DBA	Conservative Design Basis Accident
CN-DSA	Conservative Deterministic Safety Analysis
CPR	Critical Power Ratio
CRD	Control Rod Drive
CUW	Reactor Water Cleanup System
DBA	Design Basis Accident
DEC	Design Extension Condition
D-in-D	Defence-in-Depth
DL	Defence Line
DL2	Defence Line 2
DL3	Defence Line 3
DL4a	Defence Line 4a
DL4b	Defence Line 4b
DSA	Deterministic Safety Analysis
EX-DEC	Extended Design Extension Condition
EX-DSA	Extended Deterministic Safety Analysis
FAP	Forward Action Plan
FFA	Functional Failure Analysis
FMCRD	Fine Motion Control Rod Drive
FSF	Fundamental Safety Function
FW	Feedwater
FWPT	Feedwater Pump Trip
GDA	Generic Design Assessment
GEH	GE Hitachi Nuclear Energy
HFEA	Human Failure Event Analysis
IAEA	International Atomic Energy Agency
ICRW	Inadvertent Control Rod Withdrawal

Acronym	Explanation
ICS	Isolation Condenser System
IE	Initiating Event
IH	Internal Hazard
II	Inventory Increase
IR	Inventory Reduction
LFWH	Loss of Feedwater Heating
LOCA	Loss-of-Coolant Accident
LOFW	Loss of Feedwater Flow
LOPP	Loss-of-Preferred Power
MRBM	Multi-Channel Rod Block Monitor
MS	Main Steam
MSL	Main Steam Line
MSRIV	Main Steam Reactor Isolation Valves
NPP	Nuclear Power Plant
ONR	Office of Nuclear Regulation
PCCS	Passive Containment Cooling System
PCSR	Pre-Construction Safety Report
PCT	Peak Cladding Temperature
PI	Pressure Increase
PIE	Postulated Initiating Event
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RGP	Relevant Good Practice
RI	Reactivity Increase
RIV	Reactor Isolation Valve
RPC	Reactor Pressure Control
RPV	Reactor Pressure Vessel
SA	Severe Accident
SAA	Severe Accident Analysis
SC1	Safety Class 1
SDC	Shutdown Cooling System
SSCs	Structures, Systems, and Components
TAF	Top of Active Fuel
TBV	Turbine Bypass Valve
TD	Temperature Decrease

Acronym	Explanation
TCV	Turbine Control Valve
TSV	Turbine Stop Valve
UK	United Kingdom

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Revision

15.2 INDICATION, CATEGORISATION, AND GROUPING OF POSTULATED INITATING EVENTS AND ACCIDENT SCENARIOS

Introduction

Initiating Events (IEs) are achieved through the systematic process of fault evaluation. The fault evaluation objective includes:

- Identification, categorisation and grouping of Postulated Initiating Events (PIEs)
- Identification of the plant functions credited in the safety analyses and their assignment to a functional Defence Line (DL) (Defence Line 2 (DL2), Defence Line 3 (DL3), Defence Line 4a (DL4a), and Defense Line 4b (DL4b))

Scope

The scope of this Preliminary Safety Report (PSR) Subchapter comprises the safety analyses presented in NEDC-34183P, "BWRX-300 UK GDA Ch. 15.5: Deterministic Safety Analyses," (Reference 15.2-1). The fault evaluation scope is the list of potential PIEs generated by the Functional Failure Analysis (FFA) and Human Failure Event Analysis (HFEA) as described in NEDC-34179P, "BWRX-300 UK GDA Ch. 15.1: General Considerations," (Reference 15.2-3), (see Subsection 15.1.1) and includes:

- The complete range of operating modes
- All radioactivity sources (reactor core and outside core)
- Single failure PIEs and Common Cause Failure (CCF) PIEs caused by equipment failure or single human failure events
- Event sequences postulated to demonstrate the adequacy of the BWRX-300 Defence-in-Depth (D-in-D) design. These sequences consist of a PIE, an assumed CCF of an entire DL (e.g., DL2 for Anticipated Operational Occurrences (AOOs) and DL3 for Design Basis Accidents (DBAs), and the success of a latter DL function in mitigating the event sequence (e.g., DL3 for AOOs and DL4a for DBAs)

The output of the fault evaluation is documented in a fault list. The fault list establishes traceability between the plant design and the safety analysis. The fault evaluations start in parallel with or prior to Deterministic Safety Analysis (DSA) and Probabilistic Safety Assessment (PSA) activities. DSA and PSA mature with the design and the fault list is updated accordingly.

The fault evaluation includes the following activities in NEDC-33934P, "BWRX-300 Safety Strategy," (Reference 15.2-4) Implementation Process:

- Deterministic PIE Selection
- Complex Sequence Selection
- Severe Accident (SA) Sequence Selection

The deterministic PIE selection is the systematic process in organizing and selecting events for deterministic safety analyses. The activities performed during the deterministic PIE selection are described in Subsection 15.2.1. Selected PIEs and event sequences are allocated to three DSA types in a fault list:

- PIE List for Baseline Deterministic Safety Analysis (BL-DSA)
- PIE Event Sequence List for Conservative Deterministic Safety Analysis (CN-DSA)
- PIE Event Sequence List for Extended Deterministic Safety Analysis (EX-DSA)

The complex sequence selection identifies event sequences involving failures of multiple mitigating features, which have not been included in the deterministic PIE selection but are identified in the Level 1 PSA as having the potential to lead to core damage with a frequency of occurrence or consequences judged to require analysis and DL mitigation function. These complex sequences are added to the fault list and analysed in the EX-DSA.

The scope of SA sequence selection corresponds to those sequences involving significant core damage, which could lead to a breach of containment and radioactive release in the Level 2 PSA that is described in Section 15.6.3 in NEDC-34184P, "BWRX-300 UK GDA Ch. 15.6: Probabilistic Safety Assessment," (Reference 15.2-2). To the extent that core damage is not practically eliminated, representative SA sequences Design Extension Conditions (DECs) with core damage are postulated and analysed in the Severe Accident Analysis (SAA). The primary objective of the SA sequence selection is to identify representative core damage scenarios and define corresponding plant damage states that are used as the basis for performing the SAA. The selected SA sequences are documented in the fault list and are analysed in the SAA in Subsection 15.6.3 of NEDC-34184P (Reference 15.2-2).

Purpose

The purpose of this subchapter is to present the approach to the identification, categorisation, and grouping of PIEs used in the DSA for the BWRX-300.

Country Specific Material – UK Step 2 GDA

The PSR is being submitted as part of Step 2 of Generic Design Assessment (GDA) by the United Kingdom (UK) Office for Nuclear Regulation (ONR). GDA is an up-front, non-site-specific assessment of a generic Nuclear Power Plant (NPP) design. It is intended to determine whether a proposed reactor type could be constructed, operated, and decommissioned in Great Britain. Step 2 is a fundamental assessment of the generic safety, security, and environment protection cases. It is intended to identify potential showstoppers that may preclude deployment of the design.

Specific nuclear safety requirements differ between countries. Significant unique aspects of the UK regulatory regime are the overriding requirement to demonstrate that risks have been managed and reduced to As Low As Reasonably Practicable (ALARP) and clarity on the Claims, Arguments, and Evidence employed in the safety argument. The demonstration of ALARP is typically achieved through the application of Relevant Good Practice (RGP).

However, these aspects are not directly relevant to this subchapter. Other UK specific aspects are handled through the identification of future work.

Future Work

The need for future work has been identified during the production of PSR Ch. 15. This arises principally for the following reasons:

- Continuing design development
- Development of the preliminary safety arguments
- Country-specific requirements

PSR Subchapter 15.2 does not directly support any forward actions, as per Appendix B Forward Action Plan (FAP). PSR Ch. 15 subchapters that do identify FAP items, are presented in FAP schedules in their respective subchapters. Subchapters with FAP items are; NEDC-34182P, "BWRX-300 UK GDA Ch. 15.4: Human Actions," (Reference 15.2-6), NEDC-34183P (Reference15.2-1), NEDC-34184P (Reference 15.2-2), NEDC-34185P, "BWRX-300 UK GDA Ch. 15.7: Internal Hazards," (Reference 15.2-7) and NEDC-34186P, "BWRX-300 UK GDA Ch. 15.8: External Hazards," (Reference 15.2-8). Each FAP item comprises a concise description of the required work along with a project phase for when it is needed by; an outline of the reason for raising the FAP is also presented.

Chapter Route Map

This subchapter is part of PSR Ch. 15 which presents the BWRX-300 Safety Analysis and comprises the following subchapters:

- PSR Ch. 15.1 – NEDC-34179P (Reference 15.2-3)
- PSR Ch. 15.2 – NEDC-34180P, "BWRX-300 UK GDA PSR Ch. 15.2: Safety Analysis - Identification, Categorisation, and Grouping of Postulated Initiating Events and Accident Scenarios" [*this subchapter*]
- PSR Ch. 15.3 – NEDC-34181P, "BWRX-300 UK GDA Ch 15.3: Safety Objectives Acceptance Criteria," (Reference 15.2-5)
- PSR Ch. 15.4 – NEDC-34182P (Reference 15.2-6)
- PSR Ch. 15.5 – NEDC-34183P (Reference 15.2-1)
- PSR Ch. 15.6 – NEDC-34184P (Reference 15.2-2)
- PSR Ch. 15.7 – NEDC-34185P (Reference 15.2-7)
- PSR Ch. 15.8 – NEDC-34186P (Reference 15.2-8)
- PSR Ch. 15.9 – NEDC-34187P, "BWRX-300 UK GDA Ch. 15.9: Summary of Results of the Safety Analyses," (Reference 15.2-9)

This layout mainly follows the structure set out in the International Atomic Energy Agency (IAEA) specific safety guide SSG-61 "Format and Content of the Safety Analysis Report for Nuclear Power Plants," (Reference 15.2-10) with the exception that Internal and External Hazards are discussed in two separate subchapters.

Subchapter Structure

This subchapter presents the approach to the identification, categorisation, and grouping of PIEs for the BWRX-300 Safety Analyses and comprises the following main sections:

- 15.2 – Introduction
- 15.2.1 – Basis for Categorisation of Postulated Initiating Events, Accident Scenarios, and Fault Evaluation
- 15.2.2 – Categorisation of Events According to their Frequencies
- 15.2.3 – Grouping of Events According to their Type
- 15.2.4 – List of Postulated Initiating Events and Accident Scenarios
- 15.2.5 – List of Internal and External Hazards

Interfaces with other Chapters

This subchapter interfaces with the following PSR Chapters:

- PSR Ch. 15 – Safety Analysis Analyses (all other subchapters)

15.2.1 Basis for Categorisation of Postulated Initiating Events, Accent Scenarios, and Fault Evaluation

The FFA and HFEA described in Subsection 15.1.1 of NEDC-34179P (Reference 15.2-3), results in the list of potential PIEs. These potential PIEs are evaluated, categorised, and grouped during the fault evaluation in the deterministic PIE selection. The FFA and HFEA address a complete range of plant modes of operation, all sources of radioactivity, and any consequential failure that occurs because of the PIE. During design development, the initial fault evaluation is validated, and PIE selection is updated accordingly.

Ultimately, the set of evaluations and analyses demonstrate that the D-in-D design approach leads to an overall plant design that meets specific acceptance criteria per event category. The approach is a layered, comprehensive, and systematic way to evaluate events, apply DL functions, and specify related requirements for the design in an iterative, progressive manner. This process provides high confidence that once these evaluations and analyses are finalized, the plant design can be demonstrated to meet overall safety goals and has successfully implemented NEDC-33934P (Reference 15.2-4).

The activities included in the deterministic PIE selection and their bases are presented below:

- Event Sequences development – an Event Sequence is developed starting with a PIE and considers the success or failure of the required mitigating functions. The DL of each credited mitigating function is established for each Event Sequence.
- Event sequences are grouped into fault groups based on similar impact on a certain plant parameter. For example, events that lead to pressure increase in the reactor such as inadvertent closure of the Turbine Stop Valves (TSVs) and/or Turbine Control Valves (TCVs) or inadvertent closure of the Main Steam Reactor Isolation Valves (MSRIVs) are grouped in the pressure increase fault group. Subsection 15.2.4 includes the output of the grouping activity.
- Event sequences are categorised within each fault group as AOO, DBA or DEC based on their frequency of occurrence. Subsection 15.2.2 describes this categorisation.
- Plant conditions are defined corresponding to each PIE supporting the scenario analysis.
- Any exceptions are applied or justified to the standard PIE selection.

Event sequences are allocated to three types of DSAs:

- BL-DSA
- CN-DSA
- EX-DSA

A bounding set of PIEs and event sequences that result in the most significant challenge to the Fundamental Safety Functions (FSFs) are selected for evaluation in the DSA. DSA layers and events categories are combined so that limiting baseline events are Baseline Anticipated Operational Occurrence (BL-AOO), the limiting Conservative Design Basis Accident (CN-DBA) and limiting Extended Design Extension Condition (EX-DEC). This notation is used to identify the layer and event category.

Subsection 15.2.4 describes the bounding event selection for each fault group that is captured in a fault list.

A description of the three deterministic safety analyses aligned with the functional DLs (DL2, DL3 and DL4a) is included below.

15.2.1.1 Baseline Deterministic Safety Analysis

The primary objective of the BL-DSA is to model the expected plant response to AOO and DBA PIEs assuming all functions, regardless of safety category, respond as designed (except for the initiating PIE) to mitigate the event. The scope of the BL-DSA includes single failure PIEs categorised as bounding BL-AOOs, Baseline Design Basis Accidents (BL-DBAs) and DBA PIEs caused by a CCF in DL2 or DL4a functions. The BL-DSA models the expected response of the plant (no failure is postulated) to demonstrate that the event meets applicable acceptance criteria. The analysis end point is the controlled state condition. The protecting or mitigating DL functions credited in the BL-DSA for AOO PIEs are DL2 functions such as Anticipatory Trip System (ATS), maintain target level, power or pressure, or control rod block functions.

15.2.1.2 Conservative Deterministic Safety Analysis

The primary objective of the CN-DSA is demonstrating the effectiveness of DL3 functions, inherent safety features, or passive functions in mitigating AOO and DBA PIEs. The CN-DSA scope includes events categorised as bounding CN-DBAs:

- AOOs PIEs caused by a single equipment failure or a single human failure event, plus an assumed CCF of DL2 functions credited for mitigation of the same PIE in the BL-DSA
- DBA PIEs caused by a single equipment failure plus an assumed CCF of any DL2 function(s) credited for mitigation of the same PIE in the BL-DSA
- DBA PIEs caused by a CCF in DL2 or DL4a functions, plus an assumed CCF of any non DL3 functions credited for mitigation of the same PIE in the BL-DSA

The CN-DSA is performed using conservative initial conditions with established acceptance criteria and applying a graded approach in quantifying the uncertainties as described in Subsection 15.5.1.1 of NEDC-34183P (Reference 15.2-1). Single failure criterion is applied to Safety Class 1 (SC1) Structures, Systems and Components (SSCs). CN-DSA credits only DL3 mitigation functions. DL2 and DL4a functions that make the event more severe are assumed to function unless failure of the function is a consequence of the PIE or event sequence. The end point of the analysis is a controlled state condition.

15.2.1.3 Extended Deterministic Safety Analysis

The primary objectives of the EX-DSA are to:

- Demonstrate a second functional DL against DBA PIEs, where DL3 functions were credited in both the BL-DSA and CN-DSA.
- Demonstrate the plant's capability to avoid core damage in very unlikely event scenarios, which involve combinations or types of mitigation failures that are beyond those deterministically postulated ("complex sequences").

DBAs initiated by CCF in lower class equipment are not deterministically included in the EX-DSA with an additional CCF in DL3 functions added as this would constitute two CCFs occurring simultaneously in independent and diverse technologies that is outside the deterministic plant design basis. The EX-DSA scope includes events categorised as DECs:

- AOO PIEs mitigated using DL2 and DL3 hydraulic scram functions in the BL-DSA and CN-DSA, respectively. The same AOO PIE is analyzed assuming a CCF of the mechanical equipment providing motive force for hydraulic scram.
- DBA PIEs caused by a single equipment failure where DL3 functions were credited in both the BL-DSA and CN-DSA. The same DBA PIE is analyzed assuming DL3 CCF.
- PIES initiated by DL3 CCF
- PIEs caused by single equipment failures that are in the DEC frequency category
- Complex sequences

The EX-DSA can credit unaffected functions in DL2, DL3, and DL4a. Unaffected functions are those not impaired by the initiating failure, by a mitigation failure postulated as part of an event sequence, or by environmental conditions present during an event sequence.

15.2.2 Categorisation of Events According to Their Frequencies

One fundamental element of the deterministic PIE selection and event sequence selection is the assignment of event sequences to categories based on their frequency of occurrence:

- AOO (frequency greater than 1E-02 per reactor-year)
- DBA (frequency between 1E-02 and 1E-05 per reactor-year)
- DEC (frequency less than 1E-05 per reactor-year)

Limited exceptions to this approach of categorisation are described fully in NEDC-33934P (Reference 15.2-4):

- Loss-of-Coolant Accident (LOCA) PIEs are analyzed as DBAs to demonstrate core cooling performance in accordance with 10 CFR 50.46 even if their frequency of occurrence falls in the range of DEC events. This is a conservative approach aligned with international regulatory requirements for water-cooled reactors.
- PIEs initiated by digital CCF in DL3 are analyzed as DEC events even though the estimated frequency of occurrence falls in the range of DBA events. This approach is aligned with industry precedent regarding treatment of digital CCF events.
- Event sequences comprising an AOO PIE and an assumed CCF of DL2 mitigation functions are analyzed as DBA events even if their estimated frequency of occurrence is in the DEC range. This is a conservative approach and assures a DL3 response to satisfy acceptance criteria for all AOO PIEs.
- Event sequences comprising a DBA PIE and an assumed CCF of DL2 mitigation functions are analyzed as DBA events even if their estimated frequency of occurrence is in the DEC range. This is a conservative approach and assures a DL3 response, as needed to satisfy acceptance criteria for all DBA PIEs.
- Design basis hazards, both internal and external, are not categorized according to the frequency values associated with the above event categories. The design basis frequencies are hazard-specific and reflect the regional location of the facility and regulatory expectations for categorization.

Qualitative frequencies are adopted as an interim measure and are used in the early design stages to progress the performance of deterministic analyses prior to availability of more mature PSA information. Quantitative frequencies based on Level 1 PSA results are adopted as the final, governing measure of the event sequence category. The methodology for selection of event frequency is described in Subsection 15.6.1.1 of NEDC-34184P (Reference 15.2-2).

An event sequence consists of a PIE, an assumed failure of a mitigating function(s), and the DL function success that mitigates the PIE. The event sequence category is based on the sequence frequency, not the frequency of the PIE which initiates the sequence. The event category assigned to an event sequence may be different than the event category assigned to the PIE that initiated the sequence because the event sequence may include additional failures that make the sequence less likely to occur.

In addition to the event categorisation frequency, the categorised events are allocated the following DSA types:

- BL-AOO
- CN-DBA
- EX-DEC

A fundamental concept of NEDC-33934P (Reference 15.2-4) is that an event sequence that assumes failure of DL functions capable of mitigating the sequence (i.e., a CCF of all functions in a DL that are credited to mitigate a given PIE) has a lower frequency of occurrence than if the DL functions were not assumed to fail. The assumed frequency of such an event sequence is the frequency of the PIE multiplied by the required maximum frequency of failure of the DL mitigation functions postulated to fail. For example, an event sequence for a PIE with a frequency in the AOO range where an assumed failure of DL2 functions to mitigate the PIE has a frequency equal to the PIE frequency multiplied by the target reliability value for DL2 functions results in a sequence frequency in the DBA range. Accordingly, such an event sequence is evaluated as a DBA using acceptance criteria associated with DBAs. The target reliability values used are:

- 1E-02 failures/demand for DL2
- 1E-04 failures/demand for DL3
- 1E-03 failures/demand for DL4a

For example, the reliability of DL2 is such that its assumed failure to mitigate the AOO PIE creates an event sequence in the DBA event category. A CN-DSA is then performed to demonstrate DL3 mitigation of the same PIE but assuming the DL2 failure occurs. Similarly, the reliability of DL3 is such that its assumed failure to mitigate a DBA PIE creates an event sequence in the DEC event category. An EX-DSA is then performed to demonstrate DL4a mitigation of the same PIE, but assuming the DL3 failure occurs.

15.2.3 Grouping of Events According to Their Type

One of the steps in fault evaluation is grouping the events according to their type. The fault evaluation includes human failure events, functional failures, Level 1 PSA complex sequences, and Level 2 PSA SA sequences.

PIEs (faults) are grouped according to the resultant change in plant parameter:

- Temperature decreases events – decrease in core coolant temperature
- Pressure increases events – increase in reactor pressure
- Reactivity increases events – reactivity and power distribution anomalies
- Inventory increase events – increase in reactor coolant inventory
- Inventory reduction events – decrease in reactor coolant inventory
- Non-reactor events – these events are non-core related such as fuel handling accident
- Radiological accidents having dose consequences

Once the fault groups are identified, then the anticipated core physics response associated with each group is then selected. Once each group is identified, then the bounding event sequence from that group is selected.

Table 15.2-1 provides the fault groups with an explanation of how anticipated core physics response (reactor response) was considered during development of the groups. Within these groups, event sequences with similar responses are compared and used to select the bounding events.

15.2.4 List of Postulated Initiating Events and Accident Scenarios

PIEs and event frequency are first determined qualitatively based on system conceptual design, previous similar designs, and operating experience. PIEs are evaluated in the fault evaluation where they are further screened for inclusion in the fault list.

The bounding event selection is performed for events that are initiated at full power conditions (Mode 1 operating condition) because they are expected to result in the most significant challenge to the fission product barriers.

Bounding events are selected in each fault group, for each event category (e.g., AOO, DBA, DEC without core damage) and for the applicable DSA layer (e.g., baseline, conservative and extended). The resulting events selected are listed in Table 15.2-2 and analysed in NEDC-34183P (Reference 15.2-1). DEC events with core damage are part of the PSA and SAA.

The bounding event selection is performed for two event categories:

- Transient or non-LOCA described in Subsections 15.2.4.1 through 15.2.4.5
- LOCA scenarios described in Subsection 15.2.4.6

Bounding events are selected for the transient (non-LOCA) DSA that pose the most challenges in meeting the derived acceptance criteria.

The selected bounding events are summarized in Table 15.2-2 which also points to a complete description of the bounding event in the DSA described in PSR Ch. 15.5.

15.2.4.1 Decrease in Core Coolant Temperature Bounding Event

Events that result in core coolant temperature decreases are grouped as Temperature Decrease (TD) faults. A reduction in coolant temperature (at the core inlet) has the potential to challenge the fuel cladding barrier due to increasing reactivity as a result of reducing the core coolant void fraction. The Reactor Coolant Pressure Boundary (RCPB) is not challenged because there is not a significant increase in steam flow and normal pressure control is not affected. Because this is a reactivity driven event, it is only of concern when the reactor is not shutdown.

The largest source of coolant supply is from the Feedwater (FW) pumps. FW flow enters the Reactor Pressure Vessel (RPV) downcomer through the FW piping. Extraction steam from the turbine is directed to heat FW in multiple stages of FW heaters. Failures such as loss of extraction steam can result in a reduction of the temperature entering the RPV.

The Isolation Condenser System (ICS), Control Rod Drive (CRD) System, Reactor Water Cleanup System (CUW) and the Shutdown Cooling System (SDC) also have inflows to or outflows from the RPV that have the potential to reduce the coolant temperature. Inadvertent ICS initiation PIEs are included in the increase in reactor coolant inventory fault group. These systems can only reduce the coolant temperature a small fraction relative to Feedwater Heating related PIEs. Therefore, only FW heater related PIEs are considered potentially bounding.

There is no fault group for increase in core coolant temperature for the following reasons:

- The FW temperature is near the highest temperature that is feasible during normal operation
- Any increase in FW temperature would increase the core void fraction and reduce core power due to the decrease in void reactivity
- An increase in FW temperature does not result in an increase in core temperature because the core is boiling and remains at saturated conditions

Other than an increase in the FW temperature, the core coolant temperature may increase due to the following conditions:

- Core power increase resulting in a small increase in pressure covered by the reactivity and power distribution anomalies fault group

- Reactor pressure increase that effects saturation temperature covered by increase in reactor pressure fault groups

Therefore, any possible increase in FW temperature is small and results in a small decrease in power and poses no threat to fission product barriers resulting in no need for an increase in core coolant temperature fault group.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

Bounding Anticipated Operational Occurrence Event Selection

The event effect on the fuel Critical Power Ratio (CPR) and core power increase is used in selecting the bounding AOO events for this group.

The AOO that results in the largest reduction in FW temperature is the Loss of Feedwater Heating (LFWH) event. This BL-AOO event described in, Subsection 15.5.3.1.1 of NEDC-34183P (Reference 15.2-1), is the bounding AOO event for this category.

Bounding Design Basis Accident Event Selection

The bounding CN-DBA event described in Subsection 15.5.4.1.1 of NEDC-34183P (Reference 15.2-1), resulting in the largest postulated reduction in FW temperature, is the CCF leading to loss of all FW heaters. The event effect on fuel Peak Cladding Temperature (PCT) is used in selecting the bounding DBA in this group.

Bounding Design Extension Condition Without Core Damage Event Selection

There are no DEC events in this fault group because the DLs are established in the baseline and conservative DSA. No complex sequences in the fault group are identified.

15.2.4.2 Increase in Reactor Pressure Bounding Event Selection

Events that result in RCPB pressure increase are referred to as Pressure Increase (PI) faults. During full power operation, steam generated in the reactor exits the RPV through the Main Steam Lines (MSLs). There are normally open valves in the each MSL: two MSRIVs and one Main Steam (MS) Containment Isolation Valve (CIV). Outside containment, there is a MS header upstream of the TSVs. Downstream of the MS header the TSVs are in series with the TCVs.

The MSRIVs and CIVs close to isolate the RPV or containment. The TSVs and TCVs close to protect the turbine. Turbine Bypass Valves (TBVs) located in the MS equalizing line in the turbine building allow steam to bypass the closed TSV or TCV. PI faults are generally caused by closure of valves in the steam flow path. Closure of a single MSRIV terminates steam flow in one of the two MSLs. Closure of any two of these valves in separate MSLs causes a complete isolation of the MSLs upstream of the TBVs. Closure of both TSVs and/or TCVs, terminates steam flow to the turbine. A turbine trip or load rejection signal initiates closure of the TSVs and TCVs, and fast opening of the TBVs. When RPV pressure increases, TBVs are opened by the normally operating Reactor Pressure Control (RPC), which allows up to ~25% of the rated steam flow to the main condenser.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

There is no fault group for decrease in reactor pressure because a pressure decrease reduces reactivity. The decrease in reactor pressure is caused by either:

- Reactor power decrease that is covered by the reactivity at power distribution anomalies fault group
- Reactor coolant inventory is lost that is covered by the decrease in reactor coolant inventory fault groups

Therefore, there is no need for a decrease in reactor pressure fault group.

Bounding Anticipated Operational Occurrence Event Selection

The event effect on the fuel CPR, core power increase, and the RCPB pressure are used in selecting the bounding AOO events for this group. There are several AOO events in the PI group in the fault list. Most of the AOOs are PIEs that result in closure of the TSVs, TCVs, or both. These events present similar challenge to the cladding and RCPB, and all of them are selected as potentially limiting (bounding) events:

- Turbine trip, load rejection
- Loss-of-Preferred Power (LOPP)
- Loss of condenser vacuum

Another AOO in this group is the BL-AOO closure of one MSRIV. The MSRIVs do not close as fast as the TCVs or TSVs. However, the MSRIVs are upstream of the TBVs, resulting in this event selected as potentially limiting.

Bounding Design Basis Accident Event Selection

There are several DBA events that result in increased pressure in the PI group. There are two main types assumed in the PI group:

- AOO events where the TSV or TCV closes with a CCF of the DL2 mitigation equipment
- CCF results in closure of the MSRIV and Feedwater Reactor Isolation Valve

Because the TCV fast closure is faster than the MSRIV closure, the closure of the TCVs with failure of the DL2 mitigation functions is a more significant pressure increase event. However, because several of these events result in similar challenges to acceptance criteria, several CN-DBA events (described in Subsection 15.5.4 of NEDC-34183P (Reference 15.2-1)) are chosen as potentially bounding.

Bounding Design Extension Condition Without Core Damage Event Selection

The event effect on PCT and the RCPB pressure are parameters used in selecting the bounding DEC events for this group.

The main types of PI group DEC events are:

- Bounding AOO events with failure of the hydraulic scram due to postulated CCF of the hydraulic components
- Complex sequence AOO event with half of the control rods fail to insert either by Hydraulic Control Units (HCUs) or with the CRD motors run-in
- CCF in DL3 functions initiating events

The events assume a CCF of the hydraulic scram that significantly challenges the fuel cladding and RCPB because the CRD motor run-in that backs up the hydraulic scram results in slower negative reactivity insertion. Several of these events are selected as potentially bounding. The event result from CCF in DL3 functions are not concerning because sufficient non-DL3 functions initiate the hydraulic scram. In the TSV and TCV closure events, pressure control is achieved (at least momentarily) via the TBVs to the main condenser. In the loss of condenser vacuum and LOPP AOO events with failure of the hydraulic scram, the main condenser is available for a limited amount of time. These events result in more severe pressurization and are selected as potentially bounding relative to the turbine trip and load rejection events. Closure of one MSRIV AOO with failure of the hydraulic scram event sequence is also selected for evaluation. A complex sequence is also considered in this fault group based on the PSA evaluation. This event is selected as a potentially limiting event.

15.2.4.3 Reactivity and Power Distribution Anomalies Bounding Event Selection

Events that result in reactivity and power distribution anomalies are grouped as Reactivity Increase (RI) faults. These events include failures in reactivity control that challenge the fuel cladding or RCPB integrity. The BWRX-300 controls reactivity with control rod movement. Normal control is accomplished using Fine Motion Control Rod Drive (FMCRDs), which are individually connected to the control rods. HCUs are used to quickly shutdown the reactor by inserting control rods, ensuring reactivity control during off normal operation. Increases in reactivity can result from control rod withdrawal or control rod drop.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

Bounding Anticipated Operational Occurrence Event Selection

The fault list includes only one Reactivity Increase-Anticipated Operational Occurrence (RI-AOO). This event is Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW). However, this event is not evaluated because a protection function of Automatic Thermal Limits Monitor (ATLM) initiates a control rod block. The event conditions remain within normal operation conditions. Therefore, no analysis is needed. The design description of the ATLM is found in Subsection 7.3.3.2 of NEDC-34169P, "BWRX-300 UK GDA Ch. 7: Instrumentation and Control," (Reference 15.2-11).

Bounding Design Basis Accident Event Selection

The event effect on PCT and cladding integrity are the parameters used in selecting the bounding events for this group.

There are several DBA PIEs in the RI group to postulate in identifying the bounding event including the withdrawal of single or multiple control rods. The DBA PIEs are all mitigated by active DL2 functions ATLM and Multi-Channel Rod Block Monitor (MRBM) that result in minimal change to core power in the baseline sequence. The ATLM blocks rod motion before departure from normal operation and MRBM blocks rod motion before significant fuel effects occur. These mitigation functions are reliable, resulting in a failure sequence in the DEC frequency range. These events with the failure of ATLM and MRBM are selected as bounding DECs without core damage events because of the low frequency of the sequence. The following event sequences are practically eliminated or evaluated to not be bounding:

- CCF – All Control Rod Insertion at Power: For some core conditions, this event may result in a momentary increase in reactivity in the top part of the core because higher density water moves up the core faster than the rod motion. As the rods insert, high density coolant increases the reactivity above the control rods. This is a momentary effect, does not challenge fuel cladding criteria or any other fission product barrier, and is not selected as a bounding event.
- Control Rod Drop Accident: This event postulates an event that allows separation of the control rod from the drive mechanism (design precludes a separation such as this due to a separation detection device). The control rod becomes stuck and remains in its position when the drive mechanism is attempted to be withdrawn. Before the stuck rod is detected and is inserted, the control rod becomes unstuck and falls. This event

is practically eliminated with the BWRX-300 bayonet-style FMCRD coupling, hollow piston latches, dual separation detection devices that limit separation of the rod from the drive. The practical elimination of this event is discussed in Subsection 15.5.5.1 of NEDC-34183P (Reference 15.2-1).

Bounding Design Extension Condition Without Core Damage Event Selection

The event effect on PCT and cladding integrity are the parameters used in selecting the bounding DEC events for this group. There are two control rod withdrawal error DEC events that are potentially limiting:

- Single rod withdrawal
- Inadvertent withdrawal of all rod groups

These DECs without core damage are compared to applicable acceptance criteria demonstrating that features to prevent core damage are adequate and no further complementary design features are required.

15.2.4.4 Increase in Reactor Coolant Inventory Bounding Event Selection

Events resulting in an increase in reactor coolant inventory are included as Inventory Increase (II) faults. These events may occur from a FW II. Inadvertent ICS initiation scenarios do not fit well into any fault group and are included in this increase in inventory event category. Inadvertent ICS initiation results in much less II than a FW increase. However, the dynamics of ICS flow into the chimney is different than an increase in FW flow. These scenarios are selected for analysis.

Other systems such as CRD, CUW, SDC or Boron Injection System may result in inventory increases. However, these systems have low flow rates compared to FW and are not considered as bounding.

The events selected result in the most significant challenge to fission product barrier in this fault group. The selected events are summarized in Table 15.2-2.

Bounding Anticipated Operational Occurrence Event Selection

There is one BL-AOO event, Inadvertent Isolation Condenser Initiation – One Train, and it is selected for evaluation.

Bounding Design Basis Accident Event Selection

The event effect on PCT and coolant inventory are the parameters used in selecting the bounding events in this group.

Two CN-DBA events are selected:

- Inadvertent initiation of all ICS trains (bounds one train) (Inadvertent Isolation Condenser Initiation – All Trains (CCF-IICI))
- Feedwater Flow Increase - All Pumps (CCF-FWFI). This event bounds the CN-DBA event for increase in flow of one FW pump

Bounding Design Extension Condition Without Core Damage Event Selection

There are no DEC events identified in the II fault group because for each single failure AOO or DBA PIE, there are two DLs established that mitigate the event. No AOO events credit the hydraulic scram. For all CCF PIEs, a DL is established.

15.2.4.5 Decrease in Reactor Coolant Inventory Bounding Event Selection

Events that result in a decrease in reactor coolant inventory are included as Inventory Reduction (IR) events. These events may occur from failures that result in:

- Reduction or loss in FW flow
- Opening of TBVs
- Pipe breaks (LOCA events) (discussed in Subsection 15.2.4.6)
- Misalignment of systems connected to the RPV
- LOPP (this event is included in the PI group)

The IR bounding events selection only includes non-LOCA events. In this fault group, maintaining inventory above the Top of Active Fuel (TAF) ensures fuel cladding integrity. For non-LOCA events, there is no significant challenge to the RCPB or containment.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

Bounding Anticipated Operational Occurrence Event Selection

The event effect on coolant inventory is the parameter used in selecting the bounding AOO events in this group. There are two BL-AOO events in this group:

- Feedwater Pump Trip (FWPT) - one pump
- Inadvertent opening of one TBV

The FWPT event is selected as potentially limiting because the TBV opening represents less inventory loss than the FWPT and FW makes up inventory loss during the TBV opening.

Bounding Design Basis Accident Event Selection

The event effect on the coolant inventory are the parameters used in selecting the bounding DBA events for this group.

There are several PIEs events in the IR group:

- Loss of all FW flow
- Opening of all TCVs and TBVs
- RPV Pressure Control Open

The selected bounding event is Loss of Feedwater Flow (LOFW) CN-DBA event. FW may also be lost by a FW isolation valve closure DL4a CCF BL-DBA but is not selected because this is less severe than LOFW. This is also less frequent and is categorised as a DEC when combined with a CCF of DL2 mitigation. The RPV Pressure Control Open (CCF-RPCO) is selected as potentially limiting for events where inventory is lost via the MSL.

LOCA events bound these events because they result in a more significant challenge to the RPV inventory (fuel cooling and long-term cooling) and challenge containment temperature and pressure. The non-LOCA DBA events are not bounding for this IR fault group.

Bounding Design Extension Condition Without Core Damage Event Selection

The event effect on PCT and coolant inventory are the parameters used in selecting the bounding DEC events for this group.

The FW Isolation EX-DEC event is selected as a potentially limiting event. However, it is not expected to be a significantly different response than the CN-DBA event. LOCA events bound this event, and the non-LOCA events are not bounding for this IR fault group.

15.2.4.6 Bounding Event Selection for Loss-of-Coolant Accident Scenarios

The initiating events involving pipe breaks, scram, and trip initiation are identified in the fault list for evaluation in CN-DSA and EX-DSA. The evaluation scenarios are then selected to bound groups of pipe breaks. The consequences of postulated LOCAs are analysed for fuel cladding and containment integrity and shown to meet the design basis acceptance criteria in Table 15.3-2. CN-DSA LOCA analyses demonstrate that the reactor level does not decrease below the TAF, or the fuel cladding does not exceed the normal operating temperatures. Meeting these acceptance criteria assures the fuel cooling acceptance criteria in Table 15.3-2 are met.

The bounding scenarios for pipe breaks fall into two categories:

- Large breaks inside or outside containment
- Small breaks inside or outside containment

The breaks in each category may be in steam pipes or liquid pipes. These LOCA PIEs are analysed as DBAs even if their frequency of occurrence falls in the range of DEC events. CN-DSA sequences are mitigated by DL3 functions alone, assuming a CCF of DL2 functions. EX-DSA event sequences DECs are mitigated by DL2 and DL4a functions (see Subsection 15.2.4.6.4 for more details), assuming a CCF of DL3 functions.

Pipe breaks attached to the RPV may be as large as the complete rupture of the largest steam or feedwater pipes, or as small as leaks in smaller pipes attached to the RPV. Further, the largest breaks are instantaneous double-ended guillotine ruptures of the large pipes. This assumption is made to bound the thermal hydraulic response of the fuel, RPV and containment for all isolatable pipe breaks.

The bounding LOCA events provided in Table 15.2 2 and evaluated in Subsection 15.5.4.5 of NEDC-34183P, (Reference 15.2-1) (LOCA inside containment) and 15.5.9.1 (LOCA outside containment) NEDC-34183P (Reference 15.2-1) provide the assumptions and DL components used in mitigating the event with the corresponding signals, times, and other design parameters.

As described in NEDC-33910P-A "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection Licensing Topical Report," (Reference 15.2-12), a break between the RPV and the Reactor Isolation Valves (RIVs) is not evaluated as a DBA. Breaks inside containment are postulated to occur at any arbitrary location between the outer RIV, or the flow limiter for MS pipes, and the containment boundary.

Large Breaks

Pipes that are larger than 19 mm inside diameter have two isolation valves attached directly to the RPV. The largest postulated pipe breaks are in the MS, FW, and ICS lines. These pipes have RIVs, which close in less than 5 seconds once they start closing. Another 5-second delay is assumed before a RIV starts closing to account for delays in break detection and signal development.

MS pipes are equipped with a flow limiter to prevent very large break flow prior to MSRIV closure. The flow limiters are placed inside MS RPV nozzle. Breaks on the MS pipes are postulated to occur downstream of the outer RIV.

The analysis assumes the high containment pressure setpoint for break isolation is reached within 1 second for large breaks (steam and liquid). Peak containment pressure occurs at approximately the time RIVs are fully closed and the break is isolated. The largest steam pipe breaks, and the largest liquid pipe breaks are the most limiting for containment response because they have the highest mass and energy release until the break is isolated. For medium size breaks that are isolated on high containment pressure, there may be a delay in reaching the containment high pressure setpoint for isolation. However, containment pressure is increasing at a slower rate than it would for a larger break while the RIVs are closing. Because the RIV closure time is the same for all breaks, the containment peak pressure is smaller for a medium size break than it is for a large break.

Because the largest breaks are more limiting for the mass and energy release, fuel integrity and containment integrity, a break spectrum analysis is not required for isolatable breaks. This is also the case for breaks outside containment because they are isolated by the leak detection system in a similar manner.

The leakage detection system is designed to detect breaks in large pipes connected to the RPV. If the break is not detected, a conservative assumption is that the break remains un-isolated for 72 hours for CN-DBA sequences because no operator action is credited.

For large break LOCAs, RIVs close rapidly and prevent significant loss of RPV inventory. The core remains covered throughout. The remainder of the event after the RIVs are closed is an isolation event during which the ICS has ample capacity to remove the decay heat and depressurize the RPV and maintain fuel cooling for at least 72 hours. Fuel integrity is not a concern for large breaks. In the long-term, the Passive Containment Cooling System (PCCS) reduces containment pressure.

Small Breaks

The smaller pipes connected to the RPV have an inside diameter ≤ 19 mm, do not have automatic RIVs, and are considered un-isolated breaks. Small un-isolated breaks may also occur in larger pipes. The leakage detection system detects breaks in large pipes connected to the RPV but may not be capable of detecting breaks that are smaller than the area of a circle with a 19 mm diameter. If the break is not detected, it is assumed to remain un-isolated for 72 hours for CN-DBA sequences because no operator action is credited for this duration.

Small breaks conservatively credit only two of the three isolation condensers even though a break of less than 19 mm equivalent diameter on an isolation condenser does not cause degradation in the isolation condenser heat removal rate. There is sufficient steam in the RPV to feed the condensation in the isolation condenser. Not having sufficient steam in the RPV to feed the isolation condenser can only occur if the RPV is depressurized so far that almost all of the steam is escaping the break. This would be the case if the RPV pressure is even lower than that calculated for an instrument pipe break. However, in this case the break flow is less than the break flow calculated for an instrument pipe break, making the isolation condenser small break less limiting than the instrument pipe break. It can be concluded that whether the break flow rates are the same for a small steam pipe break regardless of the break location, or the RPV pressure is too low to feed the isolation condenser, the breaks on isolation condenser steam pipes are no more limiting than a break on an instrument steam pipe break.

For un-isolated breaks, the RPV inventory is depleted faster as the break area becomes larger. The largest of the un-isolated breaks is most limiting with respect to RPV inventory. Peak containment pressure is also higher as the break size is increased for an un-isolated break.

Large and Small Pipe Break Summary

The largest break sizes are the most limiting for isolatable (i.e., in the large break category) and un-isolatable (i.e., in the small break category) breaks.

In selecting the scenarios to evaluate for the pipe breaks, each of the sequences in the fault list is assessed with respect to the largest mass and energy release to containment and RPV inventory. Additional conservative assumptions that are described in Subsections 15.5.4.5 and 15.5.9.1 of NEDC-34183P (Reference 15.2-1) were made in constructing the bounding scenarios to reduce the number of analysis cases. Therefore, the bounding scenarios analysed for pipe breaks in large and small break categories are scenarios bounding all scenarios in that category. However, the liquid and steam pipe breaks are analysed separately.

The following common features are used in selecting the bounding sequences for pipe breaks:

- A pipe break does not cause loss of FW or loss of normal containment cooling unless a direct or indirect effect of the pipe break causes the pump(s) to trip. FW and normal containment cooling are lost concurrent with the break for the sequences involving a pipe break concurrent with LOPP. This observation is used to determine whether the breaks are more limiting with or without LOPP.
- All scrams credited in the LOCA analyses are direct scrams. A scram signal is initiated when the setpoint of the first scram function is reached for the scram functions that are available in the credited DL. Scram functions and setpoints in each DL and the trip parameters are provided.
- In the CN-DBA sequences, a CCF of DL2 functions concurrent with the pipe break are assumed and credit only DL3 functions.
- In the EX-DEC sequences, CCF of DL3 functions are assumed in the analysis and credit only DL2 and DL4a mitigation functions (see Section 15.2.4.6.4for more details).

The guidelines for considering CCFs in the design and safety analyses are discussed in NEDC-33934P (Reference 15.2-4).

The system responses, including the direct and indirect effects of large and small pipe breaks, are described in Subsections 15.5.4.5 and 15.5.9.1 of NEDC-34183P (Reference 15.2-1).

15.2.4.7 Large Steam Pipe Breaks

Large steam pipe breaks are postulated to occur in the following systems:

- MS pipes
- ICS steam supply pipes

For breaks inside containment, scram is initiated on high containment pressure (DL3-07). For breaks outside containment, scram is initiated on detection of a steam pipe break (DL3-09).

MSRIV closes on high containment pressure for breaks inside containment (DL3-22), and on MS pipe break detection for breaks outside containment (DL3-20). ICS RIVs close for an ICS train when an ICS break inside or outside containment is detected in the respective ICS train (DL3-27, 28 or 29).

Because the isolation valves close rapidly, the effect on isolation condenser availability is not significant for large breaks mass and energy releases. After break isolation, one ICS train is sufficient to remove decay heat and depressurize the RPV. As a bounding assumption, only one ICS train is credited for MS and ICS pipe breaks. This assumption is made so that an ICS steam supply break is bounded by an analysed MS pipe break. One ICS train initiates on high containment pressure (DL3-15) for breaks inside containment. One ICS train initiates on pipe break indication detection (DL3-16) in MS or ICS pipe breaks outside containment.

For a MS pipe break, the total break flow is the sum of break flows from both ends of the break. To bound all break locations, the break location is assumed as close to the RPV as possible, right outside the second or outboard RIV. Because two MS lines are connected through a header, the intact steam pipe also supplies the break location from the turbine side of the break. Break flow from the turbine side of the break is contributed by the flow from the RPV into the intact loop and the initial inventory in the piping. To maximize flow from the turbine side of the break, the isolation valves outside the containment are assumed to remain open, and TSVs and TCVs close rapidly. Because closure of CIVs outside containment are not credited, the calculated mass and energy release is applicable to breaks inside and outside containment.

If the break occurs when the plant is at very low power or hot shutdown, break flow from an MS pipe break may be higher due to carryover. At low power, there is more saturated water in the RPV to flash. This may increase the two-phase downcomer level much higher than that in the rated initial conditions case, contributing to the break flow due to increased liquid content although the break flow enthalpy is lower. In calculating the radiological consequences for breaks outside containment, the break mass flow rate for hot shutdown initial conditions may be more limiting. Both the rated initial conditions and the hot shutdown initial conditions are included in the bounding scenarios for MS pipe breaks.

The above scenario assumes LOPP concurrent with the break is bounding for the scenario when preferred power is available. If the preferred power is available, FW will continue to be injected into the RPV, which may increase mass release to the containment due to carryover. However, if preferred power is available, TSVs and TCVs also remain open, discharging much of the steam in the intact loop to the turbine rather than to the break from the turbine side of the break location. As a result, the break scenario with LOPP is the more bounding scenario.

Bounding CN Scenario for Large Steam Pipe Breaks

The bounding scenario analysed MS pipe breaks inside and outside containment concurrent with LOPP.

The case is analysed for rated initial power and hot shutdown conditions. The limiting break scenario is analysed in Subsections 15.5.4.5.1 and 15.5.9.1.1 of NEDC-34183P (Reference 15.2-1).

15.2.4.8 Large Liquid Pipe Breaks

Large liquid breaks may occur in:

- FW pipe
- ICS condensate return pipe
- CUW pipe

The largest pipe from this list is the FW pipe and is analysed in Subsection 15.5.4.5.2 of NEDC-34183P (Reference 15.2-1).

A large break in the ICS condensate return pipe is listed as a liquid break because the condensate return pipe is filled with water initially. However, this water is highly subcooled. The ICS condensate return valves are normally closed and the steam supply pipe is in communication with the RPV during normal operation. If a pipe break occurs at the condensate return pipe, the highly subcooled water in the condensate pipe is purged. The energy release to containment for a break inside containment or to the ICS pool for a break outside containment from the purged highly subcooled water is insignificant. After the liquid inventory steam supply pipe until the RIVs close for the broken ICS unit.

The ICS steam supply pipe is fitted with a 70 mm inside diameter orifice at the steam distribution pipes. Therefore, flow from a break in the condensate pipe is only steam flow through a 70 mm orifice after the subcooled water is purged. This is less limiting than a break in the ICS steam supply pipe included in the large steam pipe break cases. This pipe break requires no further analysis because it is included in the large steam pipe break category for CN-DBA sequences.

CUW pipe breaks may occur inside or outside the containment. CUW pipe is a smaller-bore pipe and CUW breaks are isolated the same as the FW pipe breaks. Therefore, the CUW pipe breaks inside containment are bounded by the FW pipe breaks. CUW pipe breaks are routed through the same compartments that house the FW pipes in the Reactor Building (RB). Therefore, CUW pipe breaks outside containment are also bounded by the FW pipe breaks for RB pressure and temperatures.

The limiting liquid pipe breaks inside and outside the containment are the FW pipe breaks. The break isolation, scram, and ICS initiating are discussed below.

Bounding CN Scenario for Large Liquid Pipe Breaks

For a pipe break inside containment, scram is initiated on high containment pressure (DL3-07). For breaks outside containment, scram is initiated on pipe break indication in FW or ICS pipes (DL3-09). Scram does not occur for the CUW pipe breaks outside containment.

In accordance with the fault list, preferred power is available for CUW breaks outside containment and FW injection continues.

LOPP concurrent with a break is assumed for the FW pipe breaks inside the containment. The TSVs and TCVs are conservatively assumed to close rapidly with LOPP. In the LOPP cases, there is flow from the pump side of the break before the CIVs close outside containment at 10 seconds due to pump coast down, but more importantly, due to flashing of the water in the FW piping. If the FW pumps continue running despite the break when preferred power is available, there is some increase in flow from the pump side of the break. However, normal containment cooling also continues to run when preferred power is available, compensating for the effect on containment pressure from an increase in the flow from the pump side of the break.

For breaks outside containment in the RB pressure and temperature calculations, the cases with and without FW running are included.

The radiological consequences for breaks outside containment uses the LOPP case that is more limiting. FW pump flow has no consequence because flow coming from the FW pump is decontaminated water and can only retard the break flow coming from the RPV through the intact loop. Only the water leaving the RPV is important in the radiological analyses that is higher if the pump is not running.

For breaks inside containment, RIVs close on high containment pressure (DL3-22). CIVs outside containment also close on high containment pressure (DL3-22).

For FW pipe breaks outside containment, FW RIVs close on break detection. MSRIVs also close on break detection (DL3-21).

For CUW pipe breaks outside containment, only CUW RIVs close on break detection (DL3-26).

For breaks inside containment, ICS is initiated on high containment pressure (DL3-15).

For FW and ICS pipe breaks outside containment, ICS is initiated on-line break detection (DL3-16). For CUW breaks outside containment, ICS initiation is not needed because preferred power is available.

Bounding Scenario Summary for Large Liquid Pipe Breaks

The bounding liquid pipe break for containment response is analysed for a FW pipe break inside containment:

- Double-ended guillotine FW pipe break inside containment concurrent with LOPP
- TSVs and TCVs are conservatively assumed to close rapidly retaining more energy
- FWPTs and coast down
- Scram initiation within 1 second after the break

- ICs initiate on high containment pressure
- MS and FW RIVs start closing in 5 seconds and are fully closed in 10 seconds
- FW CIVs outside containment start closing in 5 seconds and are fully closed in 10 seconds
- FW conservatively assumed to trip at time zero and coasts down with a 5 second time constant

The bounding liquid pipe break scenario outside containment is similar to breaks inside containment with the following differences:

- CIVs are conservatively assumed not to close
- FWPT may or may not occur. Area pressure and temperature calculations consider both cases. The radiological analyses conservatively assume a FWPT.

The above scenario does not rely on any scram or isolation function that is a result of the LOPP. Therefore, the same scenario bounds both preferred power available and LOPP cases.

15.2.4.9 Small Breaks

Small, un-isolated steam or liquid pipe breaks may occur in instrument lines. Small breaks in the large pipes may also remain un-isolated if they are below the leak detection system threshold, (i.e., less than 19 mm inside diameter). The lowest location for a liquid pipe break is 4 m above the TAF.

Small breaks are analysed using conservative assumptions demonstrating that fuel and containment integrity are maintained for at least 72 hours using only passive systems after which the event is terminated. The fault list includes un-isolable small breaks inside and outside the containment, with and without concurrent LOPP.

The bounding CN sequences in the fault list for a small break concurrent with LOPP are evaluated with respect to the fuel cladding and containment. For a break inside containment concurrent with LOPP, normal containment cooling system is also assumed to be lost. The energy discharged from the break to containment is removed by the PCCS, discussed in Subsection 6.5.4 of NEDC-34168P, “BWRX-300 UK GDA Ch. 6: Engineered Safety Features,” (Reference 15.2-13)) and through the containment dome. PCCS does not require actuation because it is always in service.

When the preferred power is available, FW continues to run. Because the FW pump can make up for the break flow, fuel integrity is not a concern. If preferred power is available, normal containment cooling also continues to run. If the normal containment cooling and PCCS cannot keep up with the break flow and containment pressure increases, the reactor scrams, isolation condensers initiate, and RPV depressurizes, reducing the break flow. Containment cooling continues to operate, maintaining containment pressure at a lower value than PCCS alone maintains in the LOPP case.

The bounding small liquid and steam pipe break scenarios are the same, the only difference is the discharge from the small liquid pipe break is initially from the liquid water space. It becomes steam flow after level falls below the RPV nozzle elevation of the broken pipe.

Following an un-isolated break in an instrument pipe concurrent with LOPP, turbine pressure decreases rapidly resulting in a decrease in the steam pipe pressure and an increase in steam flow. Although a consequential closure of TSVs and TCVs may occur on LOPP, this is not credited in the analysis. TSVs and TCVs are assumed to remain in their initial position. Scram is initiated when the steam pipe pressure decreases to low steam pipe pressure setpoint (DL3-02) with a 1.7 second delay. Power is assumed to remain at the initial value for an additional 2 seconds to account for the time elapsed until prompt fission power is diminished after the control rods start inserting. MSRIVs also start closing on low steam pipe pressure (DL3-17) over 5 seconds after a delay of 5 seconds.

FW pumps trip concurrent with LOPP and coast down with a 3 second time constant.

All available ICS train condensate return valves start opening with a delay of 1 second when the level falls to L2 level setpoint (DL3-14). Only two isolation condenser trains are assumed available. ICS condensate return valves are fully open 10 seconds after they start opening. There are no further actuations assumed for the remainder of the event.

Mass and energy releases from small pipe breaks do not credit containment back pressure for breaks inside containment. In addition, reactor scram and isolations initiated by high containment pressure are also not credited. Therefore, the above scenario applies to breaks outside containment as well as breaks inside containment. Because a break outside containment occurs from a longer pipe, break mass and energy releases calculated for a break inside containment bounds a small break outside containment.

Bounding Small Pipe Break

The bounding scenario is summarized as follows:

- Small steam or liquid pipe break concurrent with LOPP
- FWPTs and FW flow coasts down with a time constant of 3 seconds
- Pressure controller failure, TSVs and TCVs position remain open at their initial position
- Reactor scrams when steam pipe pressure decreases to low steam pipe pressure setpoint with a 1.7 second delay
- MSRIV closure when steam pipe pressure decreases to low steam pipe pressure setpoint with a 5-second delay
- ICS initiation when level is less than L2

15.2.4.10 Bounding Scenarios for Design Extension Condition Pipe Breaks

DEC pipe break event sequences assume DL3 CCF in addition to the pipe break. These event sequences are analysed via EX-DSA, crediting only DL2 and DL4a functions. There is either a DL2 or DL4a function for all credited DL3 functions in, Table 15.5-44, NEDC-34183P (Reference 15.2-1), except the isolation condenser pipe breaks. Because the heat removal by the ICS is a higher-class safety function than isolation by a DL4a function, no DL4a associated function exists. An un-isolated isolation condenser pipe break is like an un-isolated MS pipe break and will be evaluated via EX-DSA and will be addressed as part of the Pre-Construction Safety Report (PCSR).

DL2 and DL4a functions performing the same function for the credited DL3 functions are listed in NEDC-34183P (Reference 15.2-1) Table 15.5-45. The setpoints and timing of these functions are the same as or close to the DL3 functions. Therefore, except for the isolation condenser pipe breaks, the analysed design basis LOCA analyses bound the DEC pipe breaks for the isolatable large pipe breaks and un-isolatable small breaks.

15.2.5 List of Internal and External Hazards

The list and description of Internal and External Hazards is discussed in NEDC-34185P (Reference 15.2-7) and NEDC-34186P (Reference 15.2-8).

Table 15.2-1: Fault Groups and Explanation of Core Reactivity Response Basis

Name (ID)	Description	Discussion of Reactivity Effects
Temperature Decrease	Decrease in Core Coolant Temperature	Void reactivity is key. The decrease in temperature results in an increase in the core inlet subcooling. More core thermal power goes into heating up the water and less into void production. The core void fraction decreases and causes the power to increase. This is a relatively slow increase in core power due to the thermal inertia of the coolant.
Pressure Increase	Increase in Reactor Pressure	Void reactivity is key. Pressurization results in decrease of core voids and increase in core power. For rapid pressurization, control rod scram is needed to mitigate. Core exposure effects on void reactivity and control rod position are important aspects that are modelled and included in the analysis.
Reactivity Increase (RI)	Reactivity and Power Distribution Anomalies	Control rod reactivity is key. Errors or failure in control rod movement are expected event initiators as these events add reactivity and change the local and core wide power.
Inventory Increase (II)	Increase in Reactor Coolant Inventory	Void reactivity is key. An increase in coolant inventory results in a reactor water level increase. The increase in reactor water level normally has the following effects: – Core flow increase (small reactivity effect at rated power conditions) – Core inlet subcooling increase (because the additional inventory is expected to be from lower temperature coolant, larger reactivity effect than the core flow increase) These effects tend to increase void reactivity.
Inventory Reduction (IR)	Decrease in Reactor Coolant Inventory	Void reactivity is key. The decrease in reactor water level has the following potential results: – Core flow decrease (small reactivity effect at rated power conditions) – Core pressure decrease (if break in coolant pressure boundary cannot be compensated for by pressure control) – Core inlet subcooling decrease (comes with pressure decrease) These effects tend to insert negative reactivity, resulting in core power decrease, due to void reactivity. It is typical in LOCA analysis to ignore these effects as the protection systems typically act quickly to insert control rods and shutdown the core before the negative void reactivity feedback has significant effect.

Name (ID)	Description	Discussion of Reactivity Effects
Non-Reactor Faults	Event specific. These events are not core related.	Fuel Handling Accident

Table 15.2-2: Bounding Events Transient (Non-LOCA) and Loss-of-Coolant Accident

DSA Layer / Event Category	Event Sequence ID	DSA Section
Decrease in Core Coolant Temperature Bounding Event Summary		
BL-AOO	**Loss of Feedwater Heating (LFWH)** TD-LFWH_BL-AOO	15.5.3.1.1
CN-DBA	**Common Cause Failure – Loss of All Feedwater Heating** CCF- LFWH, Passive CCF DL2 Digital Technology Platform; TD-CCF-LFWH_CCF-DL2_CN-DBA	15.5.4.1.1
EX-DEC	None	N/A
Increase in Reactor Pressure Bounding Event Summary		
BL-AOO	**Generator Load Rejection or Turbine Trip** (LR-TT); PI-LR-TT_BL-AOO **Closure of One MSRIV** 1MSRIVC; PI-1MSRIVC_BL-AOO **Loss of Condenser Vacuum (LOCV)** PI-LOCV_BL-AOO **Loss-of-Preferred Power (LOPP)** PI-LOPP_BL-AOO	15.5.3.2.1 15.5.3.2.2 15.5.3.2.3 15.5.3.2.4
CN-DBA	**Generator Load Rejection or Turbine Trip** LR-TT, Passive CCF DL2 Technology Platform (CCF-DL2); PI-LR TT_CCF-DL2_CN-DBA **Loss-of-Preferred Power** LOPP, Passive CCF DL2 Technology Platform (CCF-DL2); PI-LOPP_CCF-DL2_CN-DBA **RPV Pressure Control Downscale** CCF - RPV Pressure Control Downscale (CCF-RPCD), Passive CCF DL2 Technology Platform (CCF-DL2); PI-CCF-RPCD_CCF-DL2_CN-DBA **Closure of All MSRIVs and FW Isolation Valves**	15.5.4.2.1 15.5.4.2.2 15.5.4.2.3 15.5.4.2.4

DSA Layer / Event Category	Event Sequence ID	DSA Section
	CCF - Closure of All MSRIVs and FW Isolation Valves (CCF-DL4a-MSRIVC-FWIV); PI-CCF-DL4a-MSRIVC-FWIV_CN-DBA	
EX-DEC	**Closure of One MSRIV** PI-1MSRIVC_CCF-Hydraulic Scram_EX-DEC **Complex Sequence of Generator Load Rejection or Turbine Trip** Complex Sequence of LR-TT + CCF-Mechanical-CRD; CSS-LR-TT_CCF-Mechanical-CRD_EX-DEC **Loss of Condenser Vacuum** LOCV, CCF-Hydraulic-Scram; PI-LOCV_CCF-Hydraulic-Scram_EX-DEC **Loss-of-Preferred Power** LOPP, CCF-Hydraulic-Scram; PI-LOPP_CCF-Hydraulic-Scram_EX-DEC	15.5.5.2.1 15.5.5.2.2 15.5.5.2.3 15.5.5.2.4
Reactivity and Power Distribution Anomalies Bounding Event Summary		
BL-AOO	None	N/A
CN-DBA	None	N/A
EX-DEC	**CCF- All Control Rod Withdrawal at Power – All Rods (CCF-ACRW)** Passive CCF DL2 Technology Platform (CCF-DL2); RI-CCF-ACRW_CCF-DL2_EX-DEC **Inadvertent Control Rod Withdrawal at Power – Single rod (ICRW)** RI-ICRW_DL2-CCF_EX-DEC	15.5.5.3.1 15.5.5.3.2
Increase in Reactor Coolant Inventory Bounding Event Summary		
BL-AOO	**Inadvertent Isolation Condenser Initiation – One Train (IICI-1)** II-IICI-1_BL-AOO	15.5.3.4.1
CN-DBA	**Feedwater Flow Increase – All Pumps** CCF-FWFI with Passive CCF DL2 Digital Technology Platform (CCF-DL2); II-CCF_FWFI_CCF-DL2_CN-DBA **Inadvertent Isolation Condenser Initiation – All Trains** (CCF-DL4a-IICI), Passive CCF DL2 Technology Platform (CCF-DL2); II-CCF-IICI_CCF-DL2_CN-DBA	15.5.4.3.1 15.5.4.3.2
EX-DEC	None	N/A

DSA Layer / Event Category	Event Sequence ID	DSA Section
Decrease in Reactor Coolant Inventory Bounding Event Summary (non-LOCA)		
BL-AOO	**Feedwater Pump Trip – One Pump** FWPT; IR-FWPT_BL-AOO	15.5.3.3.1
CN-DBA	**CCF Loss of Feedwater Flow** Passive CCF DL2 Technology Platform (CCF-DL2); IR-CCF-LOFW_CCF-DL2_CN-DBA **Reactor Pressure Vessel Pressure Control Open** CCF-RPCO, Passive CCF DL2 Technology Platform (CCF-DL2); IR-CCF-RPCO_CCF-DL2_CN-DBA	15.5.4.4.1 15.5.4.4.2
EX-DEC	**Feedwater Isolation** FW Isolation (CCF-FWI-DL3); IR-CCF-FWI-DL3_EX-DEC	15.5.5.4.1
Decrease in Reactor Coolant Inventory Bounding Event Summary (LOCA)		
CN-DBA	**Main Steam Pipe Breaks Inside the Containment, Conservative Case**	15.5.4.5.1
CN-DBA	**Feedwater Pipe Break Inside the Containment, Conservative Case**	15.5.4.5.2
CN-DBA	**Large Isolation Condenser Pipe Breaks Inside the Containment**	15.5.4.5.3
CN-DBA	**Small Steam and Liquid Pipe Breaks Inside the Containment**	15.5.4.5.4
CN-DBA	**Main Steam Line Break Outside Containment**	15.5.9.1.1
CN-DBA	**Large Feedwater Line Break Outside Containment**	15.5.9.1.2
CN-DBA	**Shutdown Cooling System Line Break Outside Containment**	15.5.9.1.3
CN-DBA	**Large Isolation Condenser Line Break Outside Containment**	15.5.9.1.4
CN-DBA	**Small Breaks Outside Containment**	15.5.9.1.5

15.2.6 References

15.2-1 NEDC-34183P, “BWRX-300 UK GDA Ch. 15.5: Deterministic Safety Analyses,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.2-2 NEDC-34184P, “BWRX-300 UK GDA Ch. 15.6: Probabilistic Safety Assessment,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.2-3 NEDC-34179P, “BWRX-300 UK GDA Ch. 15.1: Safety Analysis - General Considerations,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.2-4 NEDC-33934P, “BWRX-300 Safety Strategy,” Revision 1, GE-Hitachi Nuclear Energy.

15.2-5 NEDC-34181P, “BWRX-300 UK GDA Ch 15.3: Safety Analysis - Objectives and Acceptance Criteria,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.2-6 NEDC-34182P, “BWRX-300 UK GDA Ch. 15.4: Safety Analysis - Human Actions,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.2-7 NEDC-34185P, “BWRX-300 UK GDA Ch. 15.7: Deterministic Safety Analyses - Internal Hazards,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.2-8 NEDC-34186P, “BWRX-300 UK GDA Ch.15.8: Safety Analysis - External Hazards,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.2-9 NEDC-34187P, “BWRX-300 UK GDA Ch. 15.9: Summary of Results of the Safety Analyses,” (including Fault Schedule), Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.2-10 IAEA, SSG-61, “Format and Content of the Safety Analysis Report for Nuclear Power Plants. Specific Safety Guide series,” September 2021.

15.2-11 NEDC-34169P, “BWRX-300 UK GDA Ch. 7: Instrumentation and Control,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

15.2-12 NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection Licensing Topical Report,” GE-Hitachi Nuclear Energy, Revision 2.

15.2-13 NEDC-34168P, “BWRX-300 UK GDA Ch.6: Engineered Safety Features,” Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

The GDA CAE structure is defined within the Safety Case Development Strategy (SCDS) and is a logical breakdown of an overall claim that:

"The BWRx-300 is capable of being constructed, operated and decommissioned in accordance with the standards of environmental, safety, security and safeguard protection required in the UK."

APPENDIX B FORWARD ACTION PLAN

This chapter does not directly support any forward actions, as defined within the Forward Action Plan.