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BWRX-300 UK Generic Design Assessment (GDA) Chapter 15.4 – Safety Analysis – Human Actions

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EXECUTIVE SUMMARY

The BWRX-300 Generic Design Assessment (GDA) Preliminary Safety Report (PSR) Chapter 15.4 presents the approach to the important human actions credited within the safety analysis for the BWRX-300. It demonstrates the adequacy of the treatment of these important human actions within the safety analysis.

The BWRX-300 Safety Strategy in conjunction with the BWRX-300 Human Factors Engineering (HFE) program aim to reduce the risks and consequences related to human interactions with the plant throughout all phases of the lifecycle. Important human actions are defined as human-machine interactions that are credited in the BWRX-300 Deterministic Safety Analysis, (DSA) Probabilistic Safety Analysis or Severe Accident Analysis. PSR Ch. 15.4 describes the approach to identification, modelling, and substantiation of these important human actions. However, it is not the intention for the PSR to provide detailed substantive analysis of the important human actions. That analysis will be developed later in the safety analysis program. Other human-machine interactions are addressed via the HFE Program as described in PSR Ch. 18.

The content for this PSR chapter reflects the level of maturity of the HFE Program, plant design, and safety analyses at the time of submission.

Claims and arguments relevant to GDA Step 2 objectives and scope are summarised in Appendix A, along with a statement regarding the reduction of risk such that it is As Low As Reasonably Practicable (ALARP). Appendix B provides a Forward Action Plan.

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ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ALARP	As Low As Reasonably Practicable
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrences
ASME	American Society of Mechanical Engineers
BL-DSA	Baseline Deterministic Safety Analysis
CAE	Claims, Arguments and Evidence
CBDTM	Cause-Based Decision Tree Method
CCA	Coping Capability Analysis
CN-DSA	Conservative Deterministic Safety Analysis
DBA	Design Basis Accident
DEC	Design Extension Condition
DL	Defence Line
DL1	Defence Line 1
DL2	Defence Line 2
DL3	Defence Line 3
DL4a	Defence Line 4a
DL4b	Defence Line 4b
DSA	Deterministic Safety Analysis
EX-DSA	Extended Deterministic Safety Analysis
EPRI	Electrical Power Research Institute
FFA	Functional Failure Analysis
FSF	Fundamental Safety Function
GDA	Generic Design Assessment
GEH	GE Hitachi Nuclear Energy
HBSC	Human Based Safety Claims
HCR	Human Cognitive Reliability
HEA	Human Error Analysis
HEP	Human Error Probability
HF	Human Factors
HFE	Human Factors Engineering
HFEA	Human Failure Event Analysis
HFEITS	Human Factors Engineering Issues Tracking System
HFEPP	Human Factors Engineering Program Plan
HOHE	Human Operation Hazard Evaluation
HMI	Human-Machine Interface
HRA	Human Reliability Assessment

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Acronym	Explanation
HSRC	Human Safety and Reliability Claim
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
ISV	Integrated System Validation
LfE	Learning from Experience
OE	Operating Experience
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
ORE	Operator Reactor Experiments
PCSR	Pre-Construction Safety report
PIE	Postulated Initiating Event
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
RAW	Risk Achievement Worth
RGP	Relevant Good Practice
RIF	Risk Increase Factor
SAA	Severe Accident Analysis
SCDS	Safety Case Development Strategy
SSCs	Structures, Systems and Components
THERP	Technique for Human Error Rate Prediction
UK	United Kingdom
V&V	Verification and Validation

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SYMBOLS AND DEFINITIONS

Symbol	Definition
Not Applicable	Not Applicable

Term	Definition
Anticipated Operational Occurrences	A frequency category applied to Postulated Initiating Event (PIEs) or event sequences with frequencies of occurrence greater than or equal to 1.0E-02 per reactor-year.
Design Basis Accident	A frequency category applied to PIEs or event sequences that are expected to occur at a frequency between 1.0E-02 and 1.0E-05 per reactor year.
Design Extension Conditions	A frequency category applied to PIEs or event sequences with frequencies of occurrence less than 1.0E-05 per reactor-year.
Deterministic Safety Analysis	Safety analysis using, for key parameters, single numerical values (taken to have a probability of 1), leading to a single value for the result. Typically used with either best estimate or conservative values, based on expert judgement and knowledge of the phenomena being modelled.
Fundamental Safety Functions	The highest-level objectives that must be delivered during both normal operation and under accident conditions. Under accident conditions, the circumstances are likely to be such that control of one or more functions has been lost. However, the same fundamental objectives remain.
Important Human Action	An important human action is a human-machine interaction that is credited in the BWRX-300 DSA, Probabilistic Safety Analysis (PSA) or Severe Accident Analysis (SAA).
Human Factors Engineering	The application of knowledge about human capabilities and limitations to plant, system, and equipment design. Human Factors Engineering (HFE) ensures that the plant, system, or equipment design, tasks, and work environment are compatible with the sensory, perceptual, cognitive, and physical attributes of the personnel who operate, maintain, and support it.
Human Factors Engineering Verification and Validation	HFE Verification and Validation (V&V) evaluates completed design features including alarms, controls, indications, and their associated hardware. During HFE V&V, design features are compared with regulatory requirements and guidance, HFE requirements, and the requirements generated during analysis of operator tasks. HFE V&V consists of design verification, task support verification, and Integrated System Validation (ISV).
Human Factors Issue	A problem or finding that is known to the industry or is identified throughout the life cycle of the HFE aspects of design, development, and evaluation. Issues are items that need to be addressed later and are tracked to ensure they are not overlooked.
Human Safety and Reliability Claim	An explicit or implicit statement in the safety analysis regarding HFs and/or human performance which needs to be demonstrated to be supported by fact (i.e., substantiated) for assurance that the analysis conclusions are tenable.

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Term	Definition
Normal Operation	Operation within specified operational limits and conditions. This includes startup, power operation, shutting down, shutdown, maintenance, testing and refuelling.
Operational Experience (OPEX)	Operating experience is the collection and dissemination of knowledge gained via the operation of nuclear facilities. It often includes descriptions of actual events and near-misses and how they were identified and resolved with the objective of preventing future recurrence.
Postulated Initiating Event	A change in state of plant equipment, caused by hazards such as equipment failures and internal/external events, that impacts the performance of a Fundamental Safety Function (FSF) and requires mitigation by DL functions.
Probabilistic Safety Assessment	A comprehensive, structured approach to identifying failure sequences, constituting a conceptual and mathematical tool for deriving numerical estimates of risk.
Severe Accident Analysis	Safety analysis focused on mitigating the consequences of core damage events.

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LIST OF FIGURES

None.

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Revision

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15.4 HUMAN ACTIONS

Introduction

This Preliminary Safety Report (PSR) chapter presents the approach to the identification, assessment and substantiation of the important human actions credited within the safety analysis for the BWRX-300. It demonstrates the adequacy of the treatment of these important human actions within the safety analysis.

The content of this PSR chapter reflects the level of maturity of the HFE program, plant design, and safety analyses at the time of submission.

Chapter Structure

The chapter covers the following elements:

- The proportionate/graded approach to important human actions
- Identification of important human actions
- Important human actions relating to pre-initiators, initiating events and post-fault actions
- Human reliability assessment and the progressive substantiation of important human actions
- Important human actions in the DSA
- Important human actions in the Probabilistic Safety Assessment (PSA)

Interfaces with Other Chapters

Table 15.4-1 identifies the PSR Ch. 15.4 PSR interfacing chapters.

Volume Interfaces

Volume II – Preliminary Environmental Report includes aspects such as radioactive waste management arrangements, sampling, and monitoring. The scope of the activities described in PSR Ch. 15.4 includes any important human actions that are credited in the DSA or PSA in relation to radioactive waste management.

15.4.1 General Considerations

The overall goal of the BWRX-300 HFE program is to control the risks arising from human interactions with the plant (Reference 15.4-11 and PSR Ch. 18). In doing so, the program gives consideration to human-machine interactions that occur during construction, commissioning, decommissioning, normal operations, and outages (refuelling and maintenance outages, including extended refurbishments), as well as in abnormal, emergency, and accident conditions.

The subset of the human-machine interactions that relate to nuclear safety are referred to generally as “important human actions.” Important human actions are defined as human-machine interactions that are credited in the BWRX-300 DSA, PSA or SAA. Within the nuclear industry these human actions may also be referred to as Human Based Safety Claims (HBSC), usually in relation to claims on operator action made within the DSA or human failure events (usually in relation to the PSA). PSR Ch. 15.4 describes the approach to identification, modelling and substantiation of important human actions that are credited in the BWRX-300 DSA or PSA. Other human-machine interactions are addressed via 005N1716, “BWRX-300 Human Factors Engineering Program Plan,” (HFEPP) (Reference 15.4-11) as discussed in PSR Ch. 18.

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As human-machine interactions are a central focus of the HFE program, the methodologies, tools, and activities described in PSR Ch. 18 directly address important human actions in the following ways:

- The decisions regarding the allocation of functions provide the first step in applying a hierarchy of controls and eliminating important human actions that may not be feasible or performed reliably PSR Ch. 18, Section 18.2.3.
- 005N3747, “Human Factors Engineering Concept of Operation for BWRX-300,” (Reference 15.4-12) provides the overarching context for the important human actions.
- Learning from OPEX relating to predecessor designs is taken into account. This informs the design of similar important human actions identified for the BWRX-300 (PSR Ch. 18, Section 18.2.1). OPEX also informs the application of a hierarchy of controls and the decisions on whether important human actions should be eliminated.
- The use of task analysis (including link analysis, timeline analysis, and preliminary workload analysis) informs the development of the important human actions. It also provides evidence substantiating the important human actions and any associated human error probabilities (Section 15.4.1.5).
- The iterative HFE design activities address performance influencing factors that could undermine reliable completion of the important human actions. These design activities include the development of requirements (PSR Ch. 18, Section 18.2.2), the application of codes and standards (PSR Ch. 18, Section 18.2.2), and testing and evaluation (refer to PSR Ch. 18, Section 18.3.4).
- Issues that are identified in relation to the design and substantiation of important human actions are managed via the Human Factors Engineering Issues Tracking System (HFEITS), PSR Ch. 18, (refer to Section 18.1.5).
- Evidence supporting the substantiation of important human actions is provided by ISV. HFE Validation ensures that the design, particularly the HFE-specified aspects, accomplishes its intended goals for usability and reducing the risk of human error to as low as reasonably achievable.

15.4.1.1 Proportionate/Graded Approach

The HFE Program takes a proportionate approach to the design and substantiation of human actions based on their level of risk (PSR Ch. 18, Section 18.1.5 and 005N1716 (Reference 15.4-11)). The human actions will be assigned risk levels based on the following principles (Appendix B, Forward Action PSR15.4-190):

- Any important human actions that are credited in the DSA will be assigned a high-risk level
- Where the PSA identifies important human actions as being risk significant based on measures of risk importance these will also be assigned a high-risk level
- The remainder of the important human actions modelled in the PSA will be assigned a medium-risk level. The risk level determines the HFE application level that will be applied to the human action and the HFE application level defines the graded work scope

At present, no important human actions are credited in the DSA. With regard to important human actions credited in the PSA, the analysis is not mature enough to provide the required insights into the risk significance of the important human actions. The evaluation of important human actions will be conducted in an iterative manner throughout the system design lifecycle. If future iterations of the DSA do credit human actions, the Risk Levels will be assessed. They

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will then be re-assessed for changes with each subsequent revision of the DSA. Similarly, once the PSA is baselined, the important human action risk levels will be assessed and then re-assessed with each subsequent revision.

In relation to the design, additional reviews of the appropriateness of the HFE application level will be undertaken for each human action by considering:

- Complexity of the action
- Anticipated complexity and constraints of the Human-Machine Interface (HMI)
- Complexity of the system
- Frequency of the task
- Physical environment
- Cognitive environment
- Novelty of the action, system, or HMI technology
- Time sensitivity of the action

This ensures an appropriate and integrated treatment of the important human actions both in the safety analysis and in the design.

15.4.1.2 Human Actions and Postulated Initiating Events

The PIEs are inputs to both the DSA and PSA. The PIEs are identified from the plant level failure analysis. The design is subject to two types of plant-level failure analysis, 006N5064, "BWRX-300 Safety Strategy," (Reference 15.4-13):

- Functional Failure Analysis (FFA). The FFA identifies failures of plant systems or equipment with potential to cause a PIE that challenges an FSF
- Human Failure Event Analysis (HFEA)

The primary objective of the two failure analyses is to systematically and comprehensively identify Structures, Systems, and Components (SSCs) functional and human failures that have potential to initiate a PIE, or to initiate or worsen a hazard that leads to a PIE.

The HFEA is intended to identify failures that involve a single human failure event that could potentially lead to a PIE. The HFEA was previously referred to as the Human Operation Hazard Evaluation, (HOHE). Human actions that can lead to a PIE are referred to as Type B important human actions (or more generally within the nuclear industry, Type B HBSCs or human failure events).

Type B important human actions are those that could potentially initiate an abnormal or accident event sequence; they primarily involve:

- Errors made by personnel during normal operations while changing the state of plant equipment from a designated control location, where the change in state is performed incorrectly
- Errors by personnel during a planned maintenance activity resulting in an unintended change in the state of plant equipment

The purpose of the HFEA is to identify plausible and relevant Type B important human actions, including understanding their underlying error mechanisms. As failure of Type B important human actions result in incorrect and undesired plant equipment state changes, failure of many of these important human actions result in equipment failures that are already included in the FFA. Additionally, failure of some Type B important human actions, particularly the maintenance-related ones, may result in an internal hazard already included in the internal

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hazard evaluation. Specifically identifying the potential human “initiation” of these failures and hazards, ensures consideration of the human error causes, leads to more realistic frequency estimation, and informs other aspects of the design, specifically the HMIs. The HFEA also identifies unique “new” human-initiated failures that may not be included in or identified by other analyses.

The scope of the HFEA is limited to single human failure events. By definition, a single human failure event encompasses the multiple human error mechanisms required to impact the plant in the postulated manner, i.e., failure of an important human action includes the error(s) in decision-making and the error(s) in action of a set of grouped cognitive and physical activities that comprise the failure of a task in such a way it causes the incorrect and undesired plant equipment state change.

The HFEA scope does not include violations (intentional, but non-malicious, performance of actions in direct non-compliance with documented procedures), or malicious acts. These are addressed within the site licensing phase. However, the potential for the design to induce violations through inefficient layout of equipment is considered by the broader HFE program (see Appendix B, Forward Action PSR15.4-189).

The HFEA analysis to date has not identified any important human actions that would cause an initiating event that would not already be covered by the general transient occurrence data (see Appendix B, Forward Action PSR15.4-192). If any Type B important human actions are identified during future iterations of the HFEA, 007N3073, “BWRx-300 Human Operation Hazard Evaluation,” (Reference 15.4-14) they will result in:

- Incorporation of the relevant important human actions into the PSA model and/or deterministic PIE selection as appropriate
- Design issues being entered into the HFEITS for resolution

15.4.1.3 Identification of Pre-Initiator and Post-Fault Human Actions

The pre-initiator and post-fault important human actions are established by the safety analysis, either explicitly or implicitly (also referred to within the nuclear industry as Type A and Type C important human actions or HBSCs/human failure events respectively).

The important human actions will be identified through review of the following safety analyses (see Appendix B, Forward Action PSR15.4-191):

- Functional failure hazard evaluation (i.e., Failure Mode and Effects Analysis)
- External and internal hazard evaluation
- Fault evaluation
- PSA
 - Internal events at-power
 - Low power and shut down events
 - Spent fuel pool events
 - Fuel and heavy load movements PSA
 - Human reliability assessment
 - Level 2 PSA, SAA
- DSA
- Fire safe shutdown analysis

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- Reliability evaluation and modelling (i.e., consideration of reliability, availability, maintainability and inspectability).

15.4.1.4 Human Safety and Reliability Claims Database

The important human actions, along with the basis, context, and summary of substantiating evidence, are documented in the Human Safety and Reliability Claim (HSRC) database for tracking and future substantiation (References 15.4-11 and 15.4-15). The specific claims are reproduced verbatim wherever possible to limit the possibility for misinterpretation.

The HSRCs will be organised into three categories (these are related to the way in which the HSRCs are substantiated, not to their risk significance):

- Category 1 - Human performance claims for human action credited in the DSA for event mitigation (e.g., task performance time) and associated assumptions
- Category 2 – Human Error Probability (HEP) quantifications for human failure events modelled in the PSA and associated assumptions
- Category 3 - All other claims (e.g., assumptions regarding particular procedures or alarms, generally made to support qualitative human performance claims)

The database will capture all the important human actions claimed in the safety analyses, as well as the source of the claim, their key characteristics, the related assumptions, and any associated HMIs. This ensures visibility of the important human actions. It also enables clear links to the design activities established and managed.

15.4.1.5 Substantiation of Claims on Important Human Actions through Qualitative Human Error Analysis

The important human actions, along with the basis, context, and summary of substantiating evidence, are documented in the HSRC database for tracking and future substantiation (References 15.4-11 and 15.4-15). The specific claims are reproduced verbatim wherever possible to limit the possibility for misinterpretation.

The HSRCs will be organised into three categories (these are related to the way in which the HSRCs are substantiated, not to their risk significance):

- Category 1 - Human performance claims for human action credited in the DSA for event mitigation (e.g., task performance time) and associated assumptions
- Category 2 – HEP quantifications for human failure events modelled in the PSA and associated assumptions
- Category 3 - All other claims (e.g., assumptions regarding particular procedures or alarms, generally made to support qualitative human performance claims)

The database will capture all the important human actions claimed in the safety analyses, as well as the source of the claim, their key characteristics, the related assumptions, and any associated HMIs. This ensures visibility of the important human actions. It also enables clear links to the design activities established and managed.

15.4.2 Human Actions in Deterministic Safety Analysis

15.4.2.1 Overview

The BWRX-300 Safety Strategy defines the approach to DSA, 006N5064 (Reference 15.4-13). There are five 'layers' of DSA: Baseline Deterministic Safety Analysis (BL-DSA), Conservative Deterministic Safety Analysis (CN-DSA), Extended Deterministic Safety Analysis (EX-DSA), Coping Capability Analysis (CCA) and SAA. The mapping of the functional Defence Lines (DLs) to these analyses is summarised in Section 2.1.4 of 006N5064 (Reference 15.4-13).

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At present, there are no important human actions credited in the functional Defence Line 2 (DL2), Defence Line 3 (DL3), Defence Line 4a (DL4a), Defence Line 4b (DL4b) that are analysed in the five 'layers' of the DSA. The Licensing Topical Report (LTR) 006N5064 (Reference 15.4-13) places specific constraints on when important human actions may be credited within the DSA. These constraints are described in the following sections. The treatment of important human actions associated with PIEs that provide an input to the DSA as discussed in Section 15.4.1.2 above.

15.4.2.2 Baseline Deterministic Safety Analysis

The scope of BL-DSA includes Anticipated Operational Occurrences (AOOs) and Design Basis Accident (DBA) PIEs. These are selected through the Deterministic PIE Selection portion of the Fault Evaluation process in 006N5064 (Reference 15.4-13). This includes AOO PIEs caused by a single important human action failure.

The primary objective for the BL-DSA is to model the expected plant response to AOO and DBA PIEs assuming all functions, regardless of safety category, are available to respond as designed to mitigate the event (excepting those failed as part of PIE initiation or by consequence of the PIE). Best estimate or realistic analysis conditions are used, and the results are compared to deterministic acceptance criteria based on the event category determined for the PIE in the fault evaluation.

As the BL-DSA documents the expected response of the plant, it can reflect important human actions when it is reasonable (in terms of time to diagnose and respond, availability of indications supporting diagnosis, and availability of systems to carry out the action given the scenario) for them to be part of the expected response. This is particularly the case for relatively slow-moving plant transients, and for PIEs initiated when operators are already actively involved in manual control of the plant processes. If it is identified that a PIE could result from a human error during manual control actions, the analysis can reflect personnel correcting the error allowing the action to be successful. This can be reflected without need for extended time to perform the action and recovery, because the person performing the action is already actively engaged, subject to appropriate analysis of the recovery actions, (e.g., taking account of any dependency coupling mechanisms between the initiating error and the recovery action (see Appendix B, Forward Action PSR15.4-183)).

If any important human actions are captured in future iterations of the BL-DSA:

- Confirmation will be sought that if the actions were not taken, the acceptance criteria associated with the event category of the PIE would be satisfied
- The important human actions will be identified as Defence Line 1 (DL1) provisions to be included in the plant operating procedures
- A CN-DSA event sequence stemming from the same PIE will be analysed assuming no important human actions

Any important human actions captured in future iterations of the BL-DSA will support the development of operating procedures for off-normal conditions and the minimisation of avoidable duty-cycles on plant equipment that supports automatic DL function actuations. Both of these would contribute to the robustness of DL1 provisions for the plant design and its operation (refer to 006N5064 (Reference 15.4-13)).

15.4.2.3 Conservative Deterministic Safety Analysis

The scope of CN-DSA includes AOO and DBA PIEs selected through the deterministic PIE selection portion of the fault evaluation process, with additional mitigation failures assumed (to form event sequences) compared to the BL-DSA analysis of the same PIEs. This could include AOO PIEs caused by failure of a single important human action.

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The primary objective for the CN-DSA is to demonstrate capability to mitigate AOO and DBA PIEs and event sequences crediting only DL3 functions and inherent or passive safety features. It provides the formal demonstration of the plant's capability to maintain performance of the FSFs for a 72-hour period crediting only passive functionality and Safety Class 1 equipment. Therefore, crediting of human actions to perform a FSF is not permitted in CN-DSA.

15.4.2.4 Extended Deterministic Safety Analysis

The scope of EX-DSA includes all PIEs and event sequences assigned to the Design Extension Condition (DEC) event category selected through the deterministic PIE selection and complex sequence selection processes.

The primary objectives of the EX-DSA are to:

- Demonstrate an effective means of motive force for control rod insertion, diverse from hydraulic action
- Demonstrate a second functional DL against DBA PIEs for which DL3 functions were credited in both the BL-DSA and CN-DSA
- Demonstrate mitigation of PIEs in the DEC event category
- Demonstrate the capability of the plant to avoid core damage (severe accident conditions) in unlikely event scenarios involving combinations or types of mitigation failures that are beyond those deterministically postulated ('complex sequences')

Crediting of human actions to perform a FSF is not permitted in those DEC event sequences identified through the deterministic PIE selection process. Human actions are not credited in those EX-DSA event sequences identified through the complex sequence selection process. However, it may be allowable in exceptional cases to credit a human action in these sequences; in such cases a justification of impracticability for implementing an automatic function will be provided (refer to 006N5064 (Reference 15.4-13)).

15.4.2.5 Coping Capability Analysis

The scope of the CCA includes the event sequences selected through the coping capability sequence selection process.

The objective of the CCA is to model time-extended plant response to the selected scenarios. It is the formal demonstration that the design supports seven days of coping capability using only installed systems with no reliance on significant human actions or external resources. In this context, "coping capability" refers to the ability to ensure the FSFs are maintained for seven days after an event resulting in reactor shutdown without the necessity of human action.

It is allowable to reflect simple human actions that are rule-based and require no complex cognitive or physical activity. Specifically, the following constraints apply:

- Simple monitoring of key parameters without decision-making or further actions required in response can be claimed
- Simple "automatic" rule-based actions that are not part of mitigating the sequence can be claimed. Such actions include:
 - Assessing the radiation environment in and around the plant in line with routine radiation protection assessment processes
 - Response to personnel hazard alarms, including relocation to a protected, habitable area if personnel are located in an area subject to conditions that cause it to become uninhabitable

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- Communicating plant conditions routinely with required parties, including plant public address announcements and interfacing with security and external parties as required

Examples of human actions that cannot be credited include:

- Actions to provide back-up to failed automatic DL functions
- Actions to reconfigure a process system (start/stop pumps or fans, open/close valves or dampers)
- Actions to reconfigure electrical systems (load shedding, reconfiguring of Instrumentation and Control (I&C) equipment)

15.4.2.6 Severe Accident Analysis

The scope of the SAA is defined by those event sequences selected through the SAA selection process.

Accident progression analyses are performed to establish plant thermal-hydraulic behaviour, chronology of accident progression (the timing of the core damage and containment failure), and containment loads due to complex severe accident phenomena. This analysis includes models for the important accident phenomena that might occur in the reactor pressure vessel, in the containment, and in the reactor building. There is a strong relationship between the deterministic SAA modelling of severe accident sequence progression and the Level 2 PSA. Deterministic modelling is used to confirm that the Level 2 containment event categories and release categories are valid. As best-estimate analysis conditions are used in the SAA, human actions may be credited. If any human actions are credited, then the output of the SAA provides insights into the important human actions considered during the development of accident management procedures (refer to 006N5064 (Reference 15.4-13)).

15.4.2.7 Deterministic Hazard Analysis

Internal and external hazards are handled differently to PIEs that arise from SSC failures or human failure events. As any number of possible PIEs might result from a hazard, the evaluations do not attempt to postulate specific PIEs caused by the identified hazards. Instead, the outputs from the internal hazard evaluation and external hazard evaluation (i.e., expected frequencies versus the intensity) are fed directly into appropriate deterministic hazard analyses. The analyses will demonstrate that the plant design can withstand the hazards while maintaining performance of the FSFs.

The objective of these analyses is to demonstrate that protection is provided against all selected credible hazards and hazard sources/sub-sources through DL1 provisions within the design of the plant SSCs. DL1 provisions include operational programs that ensure the plant is operated within its analysed safety profile, and the operating procedures in place to support this. Administrative controls such as these are discussed within PSR Ch. 18, Section 18.2.5. They are not the focus of PSR Ch. 15.4.

A control room habitability analysis will be performed to identify those scenarios that could challenge the ability of the operators to remain in the MCR and to confirm that the secondary control room will be habitable in the context of those scenarios. It also confirms that the equipment supporting requisite monitoring and operator control actions in the habitable location remain functional under the conditions associated with the scenario. This addresses a subset of the performance influencing factors that could impact important human actions in hazard scenarios.

The CCA imposes specific constraints regarding operator actions that can or cannot be credited to establish or extend habitability and operability of control/monitoring locations for

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the scenarios within its scope. These constraints will be applied during the control room habitability and operability analysis.

15.4.2.8 Operator Actions Not Credited in the Deterministic Safety Analysis

The BWRX-300 design is being developed in a manner that reduces the risk of human actions leading to PIEs and minimises reliance on important human actions following a PIE, consistent with both regulatory expectations for modern nuclear power plant designs and current industry good practice.

Where credit is not taken for human actions in the DSA, this does not mean that operators should not take action in such scenarios. Proper operator responses can lessen the severity of a scenario compared to the analysed demonstration cases and can, in relatively slow developing scenarios, prevent unnecessary challenges to the equipment ultimately relied on for safety. Such operator responses, and the operating procedures that guide them, are part of DL1 and are therefore integral to the BWRX-300 Defence-in-Depth (D-in-D) concept.

For these reasons, operator actions will be reflected in certain safety analyses, both deterministic and probabilistic, to gain understanding of when the actions can be beneficial and to inform operating procedure development. Additionally, in certain types of very low likelihood event sequences operator actions may be credited to support performance of FSFs. Examples include complex sequences involving failure to shut down the reactor and external hazards probabilistically combined with un-related, yet simultaneous common cause failures of systems protected from, and qualified for the hazard.

It is recognised that errors of commission could occur when undertaking operator actions that are not credited in the DSA. Errors such as these could aggravate fault conditions. These operator actions will be addressed by the PSA and through the general activities in the HFE program. If operator actions are reflected or credited in an analysis, they will be identified as HSRCs and will be demonstrated to be achievable to the required performance standards, see 006N5064 (Reference 15.4-13). The approach to elicitation and management of the HSRCs and how their achievability will be demonstrated is described in 007N3447, "Human Factors Engineering Safety Analysis Report," (Reference 15.4-15).

15.4.2.9 Substantiation of Human Actions Claimed in the Deterministic Safety Analysis

If any human actions are credited in the DSA, their substantiation occurs in an iterative and progressive manner as the maturity of the design increases during the system lifecycle. This culminates in validation of the human actions once the design and safety case have reached maturity. Substantiation uses a risk-proportionate level of task and Human Error Analysis (HEA) (Section 15.4.1.5 and PSR Ch. 18, Section 18.2.4 and in 005N1716 (Reference 15.4-11)).

15.4.3 Human Actions in Probabilistic Safety Analysis

15.4.3.1 Overview

Two PSA levels are applied that provide estimates of overall risk to the surrounding population and environment:

- Level 1 estimates the first measure of risk (core damage frequency):
 - The scope of Level 1 PSA includes all plant operational modes (i.e., full power, low power, and shutdown) and considers events affecting both the reactor core and the spent fuel pool. It includes consideration of:
 - Internal events at-power
 - Low power and shut down events

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- Spent fuel pool events
- Fuel and heavy load movements
- Level 2 estimates the second measure of risk (radioactivity release):
 - The scope of sequences evaluated in the Level 2 PSA corresponds to the core damage sequences developed in the Level 1 PSA
 - The primary objective of Level 2 PSA is to characterise the frequency, magnitude, timing, and other relevant characteristics of potential radioactive releases resulting from the core damage sequences
 - The Level 2 PSA interfaces with the SAA

A fundamental requirement of the Safety Strategy is to ensure that any claimed human actions are achievable and meet the performance requirements for the event sequences and scenarios in which they are claimed; (refer to 006N5064 (Reference 15.4-13)). As such, suitably scoped qualitative HEA and Human Reliability Assessment (HRA) will be performed in support of the safety strategy implementation and to inform the design through design-to-analysis requirements.

The objectives of the probabilistic HRA are to derive HEPs for selected important human actions determined by the PSA screening techniques to be risk-important to event sequences within the PSA. The HRA will also inform design improvements required to support the derived probabilities. Three types of important human action are defined for modelling in the PSA:

- Type A important human actions: Pre-initiators (refer to Section 15.4.1.3)
- Type B important human actions: Initiators (Section 15.4.1.2)
- Type C important human actions: Post-fault actions (Section 15.4.1.3)

Substantiation of the important human actions and associated HEPs occurs in an iterative and progressive manner (Section 15.4.1.5).

15.4.3.2 HRA Process

In relation to HRA, the aim is to conform with the guidance of the Relevant Good Practice (RGP) presented in:

- The American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-S-1.1-2022 “Standard for Level 1 /Large Early Release Probabilistic Risk Assessment for Nuclear Power plants” (Reference 15.4-16)
- International Atomic Energy Agency (IAEA)-TECDOC-1804, “Attributes of Fuell Scope Level 1 Probabilistic Safety Assessment for Application in Nuclear Power Plants,” (Reference 15.4-17)
- UK Office for Nuclear Regulation (ONR) NS-TAST-GD-063, “Technical Assessment Guide: Human Reliability Analysis,” (Reference 15.4-18)

The HRA process uses resources from the Electrical Power Research Institute (EPRI) suite of HRA tools. The general HRA process follows the steps outlined in EPRI-NP-3583, “Systematic Human Action Reliability Procedure,” (Reference 15.4-19). For the Level 1 PSA this involves:

- Identification of human-interactions
- Capturing key assumptions
- Focusing on the key interactions through screening

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- Describing the human actions in detail
- Incorporating performance influencing factors
- Quantifying the HEPs

In relation to the Type A human actions, screening criteria from ASME HR-B1 of the ASME Level 1 PSA Standard, RA-S-1.1-2022 (Reference 15.4-16) will be applied.

For the Level 2 PSA the HRA methodology is similar to that for the Level 1 PSA, with the following specific considerations for the severe accident conditions:

- Dependency between the Level 1 PSA and the Level 2 PSA
- Stress for operators in severe accident conditions
- The environmental effect of radiation, especially on the field operators
- Evacuation from MCR to an alternate location
- Instrument failure affecting HEPs

All the important human actions from the PSA will be captured in the HSRC database, (refer to Section 5.4.1.4 above).

Additional information on the HRA methodology can be found in the “BWRX-300 Standard Plant Probabilistic Safety Assessment Methodology” (Reference 15.4-20) and “Probabilistic Safety Assessment Summary” (Reference 15.4-21).

15.4.3.3 Quantification of Human Error Probabilities

For the Type A important human actions, the EPRI “HRA Calculator Software Manual,” (Reference 15.4-22) will be used to derive an HEP based on the Accident Sequence Evaluation Program method.

The Type C important human actions will be modelled as being composed of two elements, a cognitive element, and an execution/action element. The cognitive element involves the act of recognising the need to perform an action, or procedure step. It may also encompass the operations staff briefing on an evolution. Execution involves the actual tasks taken by the operator to bring the plant to a safe stable state.

The EPRI HRA Calculator uses the Cause-Based Decision Tree Method (CBDTM) and/or the Human Cognitive Reliability/Operator Reactor Experiments (HCR/ORE) methods to derive an HEP for the cognitive element of the Type C important human actions (Reference 15.4-21), see Appendix B, Forward Action PSR15.4-188). The Technique for Human Error Rate Prediction (THERP) is then used to derive the HEP for the execution/action element of the HEP (Reference 15.4-21).

As the development of the PSA models are in their early stages, screening HEPs are currently used. The EPRI HRA Calculator approach described above has been used to derive screening HEPs for important human actions when at-power. For the low power and shut down events and spent fuel pool events PSAs, the screening HEPs have been based on engineering judgment, with suitable justification being provided.

The quantification of any HEPs associated with Type B important human actions will be undertaken on a case-by-case basis, depending on the nature of the action and error. As discussed in Section 15.4.1.2, Type B important human actions may involve errors made by personnel during planned maintenance activities, or errors made by personnel during normal operations while changing the state of equipment. The latter may also include modelling of operator actions to recover the error before it escalates into an initiating event. Therefore, the HRA Calculator will be used to apply the most appropriate human error quantification technique for the error being quantified.

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The HRA Calculator will be used for Operator Action Dependency Analysis (Reference 15.4-22). The analysis is performed with the HEPs set to 1.0 or a value close to 1.0 to drive the human actions higher in risk and keep them from being truncated out in the quantification by the frequency cut-off. Instances where there are multiple human actions in a single cut-set are then identified. A decision tree will be applied to consider the dependency coupling mechanisms and the potential level of dependency. The levels of dependency utilised and the treatment of the HEPs will be based on application of THERP (Reference 15.4-21). Additionally, where there is potential for common cognitive failure between important human actions this will be addressed through the modelling.

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15.4.4 References

- 15.4-1 NEDC-34190P, "BWRX-300 UK GDA Ch. 18: Human Factors Engineering," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.
- 15.4-2 NEDC-34165P, "Ch. 3: Safety Objectives and Design Rules for SSCs" Rev A, GE-Hitachi Nuclear Energy Americas, LLC.
- 15.4-3 NEDC-34179P, "Ch 15.1: General Considerations of the BWRX-300 Safety Analysis," GE Hitachi Nuclear Energy Americas, LLC.
- 15.4-4 NEDC-34183P, "BWRX-300 UK GDA Ch. 15.5: Deterministic Safety Analyses," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.
- 15.4-5 NEDC-34184P, "BWRX-300 UK GDA Ch. 15.6: Probabilistic Safety Assessment," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.
- 15.4-6 NEDC-34185P, "BWRX-300 UK GDA Ch. 15.7: Internal Hazards," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.
- 15.4-7 NEDC-34186P, "BWRX-300 UK GDA Ch. 15.8: External Hazards," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.
- 15.4-8 NEDC-34176P, "BWRX-300 UK GDA Ch 13: Conduct of Operations," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.
- 15.4-9 NEDC-34191P, "BWRX-300 UK GDA Ch 19: Emergency Preparedness and Response, Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.
- 15.4-10 NEDC-34199P, "BWRX-300 UK GDA Ch 27: ALARP Evaluation," Rev A, GE-Hitachi Nuclear Energy Americas, LLC.
- 15.4-11 005N1716 "BWRX-300 Human Factors Engineering Program Plan," GE-Hitachi Nuclear Energy Americas, LLC.
- 15.4-12 005N3747, "Human Factors Engineering Concept of Operation for BWRX-300, " Rev 1, GE-Hitachi Nuclear Energy Americas, LLC, 2024.
- 15.4-13 006N5064, "BWRX-300 Safety Strategy," GE-Hitachi Nuclear Energy Americas, LLC.
- 15.4-14 007N3073, "BWRX-300 Human Operation Hazard Evaluation," Rev 0, GE-Hitachi Nuclear Energy Americas, LLC, 2024.
- 15.4-15 007N3447, "Human Factors Engineering Safety Analysis," Rev 0, GE Hitachi Nuclear Energy Americas, LLC, 2024
- 15.4-16 RA-S-1.1-2022, "Standard for Level 1 - Large Early Release Probabilistic Risk Assessment for Nuclear Power Plants," American Society of Mechanical Engineers, 2022.
- 15.4-17 IAEA-TECDOC-1804, "Attributes of Full Scope Level 1 Probabilistic Safety Assessment (PSA) for Applications in Nuclear Power Plants," International Atomic Energy Agency, 2016.
- 15.4-18 ONR NS-TAST-GD-063, "Technical Assessment Guide Human Reliability Analysis," Office for Nuclear Regulation, 2022.
- 15.4-19 EPRI NP-3583, "Systematic Human Action Reliability Procedure (SHARP), Electric Power Research Institute," Electric Power Research Institute, 1984.
- 15.4-20 006N2915, "BWRX-300 Standard Plant Probabilistic Safety Assessment Methodology," GE-Hitachi Nuclear Energy Americas, LLC.

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- 15.4-21 "Probabilistic Safety Assessment Summary," GE-Hitachi Nuclear Energy Americas, LLC.
- 15.4-22 EPRI 3002010680, "The HRA Calculator Software Manual," Electric Power Research Institute, 2017.
- 15.4-23 "Safety Assessment Principles for Nuclear Facilities," Office for Nuclear Regulation.
- 15.4-24 NEDC-34140P, "BWRX-300 Safety Case Development Strategy," GE-Hitachi Nuclear Energy Americas, LLC.

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Table 15.4-1: Interfacing Chapters

PSR Chapter	Summary of Interface with Chapter 15.4
Main Interfaces	
NEDC-34190P, "BWRX-300 UK GDA Ch. 18: Human Factors Engineering," (Reference 15.4-1)	PSR Ch. 18 discusses the HFE program for the BWRX-300 and demonstrates the adequacy of integration of HFE requirements and analysis results into the plant design. Section 18.2.5 addresses the treatment of important human interactions with the plant in general. It highlights how human actions are addressed within the HFE program in general and provides the link to PSR Ch. 15.4.
NEDC-34165P, "BWRX-300 UK GDA Ch. 3: Safety Objectives and Design Rules for SSCs," (Reference 15.4-2)	Safety objectives and design rules for SSCs describes the BWRX-300 general design principles and processes. It summarises measures and assessments to ensure safety, including human factors. This chapter provides the radiological acceptance principles and criteria. PSR Ch. 15.4 describes how important human actions are incorporated into the safety analysis for the BWRX-300.
NEDC-34179P, "BWRX-300 UK GDA Ch. 15.1: General Considerations of the BWRX-300 Safety Analysis (Reference 15.4-3)	PSR Ch. 15.1 defines of the scope of the safety analysis and the approach adopted (i.e., conservative, or realistic, as appropriate) for each plant state, from normal operation to DEC with core melting. It defines the scope of the analysis for PSR Ch. 15.4.
NEDC-34183P, "BWRX-300 UK GDA Ch. 15.5: Deterministic Safety Analyses," (Reference 15.4-4)	PSR Ch. 15.5 defines initiating faults and hazards that are reasonably foreseeable, conservatively justifies accident sequences that follow those faults and hazards and assesses the design against engineering principles. It defines where important human actions are credited within the DSA. In doing so it provides a key input to PSR Ch. 15.4, which deals with the analysis of those important human actions.
NEDC-34184P, "BWRX-300 UK GDA Ch. 15.6: Probabilistic Safety," Assessment," (Reference 15.4-5)	PSR Ch. 15.6 defines the approach to implementing a PSA that supports risk-informed design development. This enables an understanding of the overall risk and any dominant contributors. The PSA also provides essential understanding of strengths and weaknesses of the design with complex systems and interdependencies. The important human actions and human reliability assessment described in PSR Ch. 15.4 are integral parts of the PSA.
NEDC-34185P, "BWRX-300 UK GDA Ch. 15.7: Internal Hazards," (Reference 15.4-6)	PSR Ch. 15.7 defines the approach to internal hazards and where important human actions may be credited in the safety analysis.
NEDC-34186P, "BWRX-300 UK GDA Ch. 15.8: External Hazards," (Reference 15.4-7)	PSR Ch. 15.8 defines the approach to external hazards and where important human actions may be credited in the safety analysis.
Other Chapter Interfaces	
NEDC-34176P, "BWRX-300 UK GDA Ch. 13: Conduct of Operations," (Reference 15.4-8)	Conduct of operations describes the BWRX-300 organizational structure, staffing and procedures, including the use of Human Factors (HF) methods and guidance in their future development. These operational aspects provide the context for the important human actions considered in PSR Ch. 15.4.

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PSR Chapter	Summary of Interface with Chapter 15.4
NEDC-34191P, "BWRX-300 UK GDA Ch. 19: Emergency Preparedness," (Reference 15.4-9)	Emergency preparedness describes the use of functional requirement analysis, allocation of functions, and task analysis to identify the necessary parameters for accident monitoring using human factors principles. PSR Ch. 15.4 deals with the analysis of important human actions claimed in severe accidents and informs emergency preparations.
NEDC-34199P, "BWRX-300 UK GDA Ch 27: ALARP Evaluation," (Reference 15.4-10)	PSR Ch. 27 provides the ALARP demonstration for the BWRX-300. PSR Ch. 15.4 contributes to ALARP arguments in relation to the potential for human error.

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APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

A.1 Claims, Arguments and Evidence

The ONR SAPs 2014, "Safety Assessment Principles for Nuclear Facilities," (Reference 15.4-23) identify the expectation of the ONR that a safety case should clearly set out the trail from safety claims, through arguments to evidence. The Claims, Arguments and Evidence (CAE) approach is explained as follows:

- Claims (assertions) are statements that indicate why a facility is safe
- Arguments (reasoning) explain the approaches to satisfying the claims
- Evidence (facts) supports and forms the basis (justification) of the arguments

The GDA CAE structure is defined within NEDC-34140P, "BWRX-300 Safety Case Development Strategy," (SCDS) (Reference 15.4-24) 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."

This overall claim is broken down into Level 1 claims relating to environment, safety, security, and safeguards, which are then broken down again into Level 2 area related sub-claims and then finally into Level 3 (chapter level) sub-claims.

The Level 3 sub-claim identified within the SCDS (Reference 15.4-25) that this chapter demonstrates compliance against is as follows:

2.3.5 Human Factors assessments have been integrated into the design, safety assessments and management arrangements, to meet the relevant safety requirements.

Important human actions are a subset of the HMIs addressed by human factors that relate to nuclear safety. As such, the treatment of important human actions also contributes to the demonstration of compliance for other chapter level sub-claims (Table 15.4-2).

This PSR chapter has derived a suite of arguments that summarise how the applicable Level 3 sub-claims are met (Table 15.4-2).

It is not the intention to generate a comprehensive suite of evidence to support the derived arguments, as this is beyond the scope of GDA Step 2. However, where evidence sources are available, examples are provided in the section of the chapter referenced in Table 15.4-2.

A.2 Risk Reduction As Low As Reasonably Practicable

It is important to note that nuclear safety risks cannot be demonstrated to have been reduced ALARP within the scope of a 2-Step GDA. In relation to important human actions, understanding the human contribution to risk and achieving an ALARP position requires information that is not available at GDA Step 2 such as the full suite of tasks to be performed (tasks claimed in the safety studies and other important human actions, for example relating to maintenance and refuelling), as well as details on conduct of operations. It is considered that the most that can be realistically achieved is to provide a reasoned justification that the BWRX-300 design aspects will effectively contribute to the development of a future ALARP statement. In this respect, this chapter contributes to the overall future ALARP case by demonstrating that the chapter-specific arguments derived may be supported by existing and future planned evidence for the arguments in Table 15.4-2.

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Probabilistic safety aspects of the ALARP argument are addressed within PSR
Ch. 15.6 - Probabilistic Safety Assessment.

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Table A-1: Human Actions and Related Claims and Arguments

Chapter 15.4 Claim	Chapter 15.4 Argument	Sections and/or Reports that Evidence the Arguments
<p>2.1 All functions have been derived and substantiated taking into account RGP and OPEX, and processes are in place to maintain these through-life.</p>		
<p>2.1.2 The design of the system has been substantiated to achieve the safety functions in all relevant operating modes.</p>	<p>HEA contributes to design substantiation by confirming that important human actions that support safety functions are feasible and can be reliably performed.</p>	<p>15.4.1.5 Substantiation of Claims on Important Human Actions through Qualitative HEA</p>
	<p>The HFE V&V program evaluates the plant design (in parts and as an integrated whole) against safety case requirements, HFE design principles and requirements, user task requirements, job design and staff complement, procedural accuracy and usability, and effectiveness of training. In addition, HFE V&V activities provide the evidence that supports the substantiation of important human actions credited within the DSA and PSA.</p>	<p>15.4.1 General Considerations (in particular HFE V&V) 15.4.1.5 Substantiation of Claims on Important Human Actions through Qualitative HEA 15.4.2.9 Substantiation of Human Actions Claimed in the DSA</p>
<p>2.3 A suitable and sufficient safety analysis has been undertaken which presents a comprehensive fault and hazard analysis that specifies the requirements on the safety measures and informs emergency arrangements (Safety Analysis)</p>		
<p>2.3.5 Human Factors assessments have been appropriately integrated into the design, safety assessments and management arrangements, to meet the relevant safety requirements.</p>	<p>A graded (or proportionate) approach is applied to the conduct of activities within the HFE Program. This provides an appropriate level of analysis to substantiate important human actions.</p>	<p>15.4.1.1 Proportionate/Graded Approach 15.4.1.4 Human Safety and Reliability Claims Database 15.4.1.5 Substantiation of Claims on Important Human Actions through Qualitative HEA</p>

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Chapter 15.4 Claim	Chapter 15.4 Argument	Sections and/or Reports that Evidence the Arguments
	<p>The results from the HEA, including human factors issues, assumptions and requirements feed into the design and the development of organisational arrangements such as staffing, training, and procedures.</p>	<p>15.4.1.4 Human Safety and Reliability Claims Database</p>
	<p>Risk-proportionate task analysis is carried out to analyse tasks allocated to human or shared during the Allocation of Function process. The overall objective is to identify design requirements to ensure that tasks, including any important human actions claimed in the BWRX-300 safety analyses, are feasible and can be reliably performed.</p>	<p>15.4.1 General Considerations</p>
<p>2.4 Safety risks have been reduced as low as reasonably practicable</p>		
<p>2.4.1 RGP has been taken into account across all disciplines.</p>	<p>The elements of the HFE program and methodologies relating to important human actions are based on international standards, guidance, RGP and multiple nuclear regulatory requirements.</p>	<p>15.4.1 General Considerations</p>
<p>2.4.2 Operating Experience (OE) and Learning from Experience (LfE) has been taken into account across all disciplines.</p>	<p>The HFE program includes the identification, review, use and application of Operating Experience to ensure that HF issues (lessons learned and good practice) are incorporated into the design and safety analyses.</p>	<p>15.4.1 General Considerations</p>

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APPENDIX B FORWARD ACTION PLAN

Table B-1: Human Actions: Forward Actions

Unique Code	Finding	Forward Actions	Delivery Phase
PSR15.4-183	<p>Dependency between Type B and Type C human actions has been dismissed by the argument that: Emergency Operating Procedures are considered to decouple any dependency between operator induced Type B human failure events and Type C human failure events.”</p> <p>The argument is insufficient as elsewhere it is acknowledged that the same operator may be involved in both actions.</p>	<p>Methodologies to be revised to address human error dependency between Type B and Type C human actions.</p>	<p>During Pre-Construction Safety Report (PCSR) development</p>
PSR15.4-185	<p>In defining a proportionate approach to substantiation of human actions, only FV has been identified as a measure of risk significance for Category 2 HSRCs. It would be usual to use additional measures e.g., Risk Increase Factor (RIF)/Risk Achievement Worth (RAW) or consider the sensitivity to a shift in HEP of a number of orders of magnitude.</p>	<p>The approach to determining the risk significance of Category 2 HSRCs should include measures of risk significance that are not influenced by the HEP.</p>	<p>During PCSR development</p>
PSR15.4-186	<p>ONR’s position is that no human error quantification methods have been fully validated for modelling human-computer interaction.</p>	<p>The approach to quantification of any HEPs associated with software-based HMIs should be reviewed and justified.</p>	<p>During site-specific work</p>
PSR15.4-187	<p>Giving consideration to errors of commission is excluded in the approach described in Section 8.2.3 of 006N2915 “BWRX-300 Standard Plant Probabilistic Safety Assessment Methodology.” This would not be an acceptable position to ONR.</p>	<p>Methodologies for analysing human actions to be revised to include consideration of errors of commission.</p>	<p>During PCSR development/before Site License Application</p>

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Unique Code	Finding	Forward Actions	Delivery Phase
PSR15.4-188	The EPRI HRA Calculator uses time-reliability curves to quantify the cognitive element of human actions. THERP is used for the execution element. ONR have concerns over the use of time-reliability curves, especially where they are used to screen human actions.	Methodologies for analysing human actions to be revised to explain and justify the use of the cognitive models within the EPRI HRA Calculator in relation to screening and task analysis.	During PCSR development
PSR15.4-189	Violations in general are not within GEH's scope for Step 2. However, ONR's expectation is that consideration should be given to the potential for design induced violations, e.g., where plant layout or task design may result in the operator perceiving inefficiencies that they subsequently work around.	The task analysis methodology and approach to HFE V&V should be revised to give consideration to the potential for design induced violations, for instance, whether task design is inefficient, such that operators could be motivated to seek more efficient ways of completing tasks.	During PCSR development
PSR15.4-191	The majority of the Type A Human Actions: Pre-Initiating Event (Type A) Human Actions are not currently modelled in the PSA due to the incomplete design of the plant.	Some Type A events have already been identified and these will be incorporated into the model. Further analysis will be conducted as the design evolves and procedures are developed to ensure all Type A events are identified.	During PCSR development/before site license application
PSR15.4-192	Type B Human Actions: Currently no Initiating Events Induced by Human Actions (Type B events) have been identified in the PSA.	Further analysis will be conducted as the design evolves and procedures are developed to ensure all Type B events are identified.	Before site license application