Revision	Approved by	Number of Pages
001		29
Approval Date	06/02-20	

General Nuclear System Ltd.

UK HPR1000 GDA Project

Document Reference:	HPR/GDA/PCSR/0033

Title:

Pre-Construction Safety Report

Chapter 33

ALARP Evaluation

DISTRIBUTION LIST

Recipients	**Cross Box**
General Nuclear System Executive	☐
General Nuclear System all staff	☐
General Nuclear System and BRB all staff	☒
CGN	☒
EDF	☒
Regulators	☒
Public	☒

TABLE OF CONTENTS

33.1 List of Abbreviations and Acronyms

ACP	Auxiliary Control Panel
ACPR1000	Advanced Chinese Pressurised Reactor
ALARP	As Low As Reasonably Practicable
ASG	Emergency Feedwater System [EFWS]
ASP	Secondary Passive Heat Removal System [SPHRS]
BAT	Best Available Technique
BSL	Basic Safety Level
BSO	Basic Safety Objective
CGN	China General Nuclear Power Corporation
CPR1000	Chinese Pressurised Reactor
CPR1000$^+$	Chinese Improved Pressurised Reactor
DiD	Defence in Depth
DR	Design Reference
ECS	Extra Cooling System [ECS]
EDG	Emergency Diesel Generator
EHR	Containment Heat Removal System [CHRS]
GDA	Generic Design Assessment
HPR1000	Hua-long Pressurised Reactor
HPR1000 (FCG3)	Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3
I&C	Instrumentation and Control
IVR	In-Vessel Retention
KDA	Severe Accident I&C System [SA I&C]
KDS	Diverse Actuation System [DAS]
KIC	Plant Computer Information and Control System [PCICS]
MCR	Main Control Room
NNSA	National Nuclear Safety Administration

NPP	Nuclear Power Plant
OPEX	Operating Experience
PCER	Pre-Construction Environment Report
PCSR	Pre-Construction Safety Report
PSA	Probabilistic Safety Assessment
PZR	Pressuriser
RBS	Emergency Boration System [EBS]
RCP	Reactor Coolant System [RCS]
RGP	Relevant Good Practice
RIC	In-core Instrumentation System [IIS]
RIS	Safety Injection System [SIS]
RPT	Radiation Protection Target
RPV	Reactor Pressure Vessel
RRI	Component Cooling Water System [CCWS]
SAPs	Safety Assessment Principles
SBO	Station Black Out
SEC	Essential Service Water System [ESWS]
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
UK HPR1000	UK version of the Hua-long Pressurised Reactor
UPS	Uninterruptible Power Supply

System codes (XXX) and system abbreviations (YYY) are provided for completeness in the format (XXX [YYY]), e.g. Containment Heat Removal System (EHR [CHRS]).

33.2 Introduction

Based on the requirements of the *Health and Safety at Work etc. Act 1974*, Reference [1], as expressed in *Reducing Risks, Protecting People*, Reference [2], and the nuclear industry specific application in *The Tolerability of Risk from Nuclear Power Stations*, Reference [3], it is necessary to show that the nuclear safety risks to workers and the public caused by a Nuclear Power Plant (NPP) are As Low As Reasonably Practicable (ALARP).

The objective of this chapter is to summarise the ALARP evaluation for the UK version of the Hua-long Pressurised Reactor (UK HPR1000) design.

The ALARP assessment presented in this chapter is based on the current ALARP assessment in other related PCSR chapters, and the basis of Design Reference (DR) is consistent with that of other related chapters.

33.2.1 Chapter Route Map

Demonstrating that the nuclear safety risks are reduced to a level that is ALARP is essential to support the ***Fundamental Objective*** of UK HPR1000:

The ***Fundamental Objective*** of the UK HPR1000 is that: *The Generic UK HPR1000 could be constructed, operated, and decommissioned in the UK on a site bounded by the generic site envelope in a way that is safe, secure and that protects people and the environment.*

To underpin this objective, five high level claims (Level 1 claims) and a number of Level 2 claims are developed and presented in Chapter 1. This chapter supports ***Claim 3.4*** derived from the high level ***Claim 3***.

Claim 3: *The design and intended construction and operation of the UK HPR1000 will protect the workers and the public by providing multiple levels of defence to fulfil the fundamental safety functions, reducing the nuclear safety risks to a level that is as low as reasonably practicable.*

Claim 3.4: *The safety assessment shows that the nuclear safety risks are ALARP.*

To support the Claim 3.4, this chapter develops five arguments and a number of relevant evidences:

a) ***Argument 3.4.SC33-A1:*** *An ALARP demonstration methodology has been developed to review the design. It is being applied to determine whether there are practicable options to further reduce risk. The methodology is outlined in Sub-chapter 33.4;*

   1) ***Evidence 3.4.SC33-A1-E1:*** *The ALARP Methodology document, Reference [4], has been issued to guide the ALARP demonstration for UK HPR1000;*

   2) ***Evidence 3.4.SC33-A1-E2:*** *The methodology has been applied in each topic area to support the production of the ALARP demonstration reports for the topic areas, where the identification and review against Relevant Good Practice (RGP), identification of potential improvements, analysis of potential improvements, optioneering, etc., will be presented.*

b) ***Argument 3.4.SC33-A2:*** *A historic review of the Hua-long Pressurised Reactor (HPR1000) design shows that the HPR1000 is an optimised design. A summary of the historic review of the HPR1000 design is presented in Sub-chapter 33.5;*

Evidence 3.4.SC33-A2-E1: *The document HPR1000 R&D History, Reference [5], summarises the main design modifications from Chinese Pressurised Reactor (CPR1000) to HPR1000, demonstrating that the evolutionary design process of HPR1000 has led to an optimised and balanced design.*

c) ***Argument 3.4.SC33-A3:*** *All reasonably practicable options to improve nuclear safety have been adopted, demonstrating that the risk is ALARP. The system design of UK HPR1000 will be reviewed to demonstrate that all reasonably practicable measures have been adopted to improve the design. Potential improvements and potential design changes that have been identified and considered will be discussed in Sub-chapter 33.6;*

Evidence 3.4.SC33-A3-E1: *Optioneering reports (if any) for the topic areas describe the optioneering process when the case is complicated. The ALARP demonstration reports for the topic areas summarise the ALARP assessments within the topic areas to analyse the potential improvements (if any) and to demonstrate that all reasonably practicable options have been implemented. Some important ALARP demonstrations for the topic areas are summarised in the document Holistic ALARP Demonstration Report, Reference [6].*

d) ***Argument 3.4.SC33-A4:*** *The radiological protection assessment demonstrates that the risk to workers and members of the public from the potential harmful effects of ionising radiation during normal operation is ALARP. A summary of the assessment is presented in Sub-Chapter 33.7, which forms an important part of the ALARP assessment for UK HPR1000;*

Evidence 3.4.SC33-A4-E1: *Radiation protection targets have been set out for normal operation, fault and accident conditions in the PCSR Chapter 4 Sub-chapter 4.4.1.3. Argument 4 is supported by the assessment of radiation protection targets for normal operation which is addressed in Pre-Construction Safety Report (PCSR) Chapter 22 and Pre-Construction Environment Report (PCER) Chapter 7 Radiological Assessment, Reference [7]. Preliminary results of the assessment of radiation protection targets for normal operation are presented in Sub-chapter 33.7.*

e) ***Argument 3.4.SC33-A5:*** *The radiological protection assessment demonstrates that the risk to workers and members of the public from the potential harmful effects of ionising radiation resulting from fault and accident conditions complies with UK legal requirements and is ALARP. A summary of the assessment will be presented in Sub-chapter 33.7, which forms an important part of the ALARP assessment for UK HPR1000.*

Evidence 3.4.SC33-A5-E1: *This argument will be supported by the assessment of radiation protection targets for fault and accident condition* *which will be addressed in Chapter 12 and Chapter 14. The assessments of the radiation protection targets for fault and accident condition will also rely on the analysis presented in Chapter 13 and Chapter 22.*

33.2.2 Chapter Structure

The general structure of this chapter is presented as follows:

a) Sub-chapter 33.1 List of Abbreviations and Acronyms:

This sub-chapter lists all the Abbreviations and Acronyms which are used in this chapter.

b) Sub-chapter 33.2 Introduction:

This sub-chapter gives a brief introduction to the chapter route map, structure and interfaces of this chapter.

c) Sub-chapter 33.3 Applicable Codes and Standards:

This sub-chapter presents the applicable codes and standards considered in the ALARP evaluation for the UK HPR1000 design.

d) Sub-chapter 33.4 ALARP Methodology for UK HPR1000:

This sub-chapter outlines the ALARP methodology used for the UK HPR1000, which supports Argument 1 for this chapter.

e) Sub-chapter 33.5 Historical Design Process:

This sub-chapter reviews the historical development process for the HPR1000 showing the rationale for major design characteristics, which supports Argument 2 for this chapter.

f) Sub-chapter 33.6 ALARP Review of UK HPR1000:

This sub-chapter presents a summary of the process that has been developed to identify potential improvements and to consider reasonably practicable options to reduce the risk.

g) Sub-chapter 33.7 Assessments of Radiation Protection Targets:

This sub-chapter presents a summary of the findings from the evaluation of the Radiation Protection Targets (RPTs) to confirm that the risks have been reduced to ALARP.

h) Sub-chapter 33.8 Concluding Remarks:

This sub-chapter summarises the claims and arguments and presents the concluding remarks.

i) Sub-chapter 33.9 References:

This sub-chapter lists all the references supporting this chapter.

33.2.3 Interfaces with Other Chapters

The interfaces with other PCSR Chapters are listed in the following table.

T-33.2-1 Interfaces between Chapter 33 and Other Chapters

PCSR Chapter	Interface
Chapter 1 Introduction	Chapter 33 provides arguments and evidence to support the relevant nuclear safety claims presented in Chapter 1.
Chapter 4 General Safety and Design Principles	Chapter 4 presents general safety and design principles which are consistent with RGP and provides the RPTs addressed in Chapter 33.
Chapter 5 Reactor Core	The ALARP approach presented in Chapter 33 has been applied in Chapters 5 to 11 to perform the ALARP demonstration for the structure, system and component designs, which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 6 Reactor Coolant System	
Chapter 7 Safety Systems	
Chapter 8 Instrumentation and Control	
Chapter 9 Electric Power	
Chapter 10 Auxiliary Systems	
Chapter 11 Steam and Power Conversion System	
Chapter 12 Design Basis Condition Analysis	Chapter 12 provides the assessment of RPTs for design basis conditions which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 13 Design Extension Conditions and Severe Accident Analysis	Chapter 13 provides the analysis of design extension conditions and severe accidents which support the assessment of radiation protection targets for fault and accident condition, which will eventually support the overall ALARP demonstration addressed in Chapter 33.
Chapter 14 Probabilistic Safety Assessment	Chapter 14 provides the analysis of risk and the evaluation of RPTs for fault and accident conditions which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 15 Human Factors	Chapter 15 provides the assessment of human action impacting safety which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 16 Civil Works & Structures	ALARP analysis for civil structures by applying the ALARP methodology is provided by Chapter 16 to support the overall ALARP demonstration addressed in Chapter 33.

PCSR Chapter	Interface
Chapter 17 Structural Integrity	ALARP analysis for structural integrity by applying the ALARP methodology is provided by Chapter 17 to support overall ALARP demonstration addressed in Chapter 33.
Chapter 18 External Hazards	Chapter 18 provides the ALARP demonstration for external hazards protection by applying the ALARP methodology, which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 19 Internal Hazards	Chapter 19 provides the ALARP demonstration for internal hazards protection by applying the ALARP methodology, which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 21 Reactor Chemistry	Chapter 21 demonstrates that the chemistry aspects of the plant design have been developed to reduce the risk ALARP, which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 22 Radiological Protection	Chapter 22 provides the risk assessment related to radiological protection and the evaluation of RPTs 1 and 2 which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 23 Radioactive Waste Management	The ALARP approach presented in Chapter 33 has been applied in Chapter 23 to perform the ALARP demonstration for radioactive waste management, which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 24 Decommissioning	The ALARP approach is presented in Chapter 33. And Chapter 24 presents the ALARP assessment for decommissioning based on these principles and the approach
Chapter 25 Conventional Safety and Fire Safety	Chapter 25 demonstrates that the conventional health and safety risks and fire safety risks are ALARP, which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 28 Fuel Route and Storage	The ALARP approach presented in Chapter 33 has been applied in Chapters 28 to perform the ALARP demonstration for fuel handling and storage, which supports the overall ALARP demonstration addressed in Chapter 33.

PCSR Chapter	Interface
Chapter 29 Interim Storage of Spent Fuel	The ALARP approach presented in Chapter 33 covers the ALARP approach adopted for the work included in PCSR Chapter 29, which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 30 Commissioning	Chapter 30 demonstrates that the safety risks are reduced to the level that is ALARP by in the consideration of commissioning, which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 31 Operational Management	Chapter 31 demonstrates that the safety risks are reduced to the level that is ALARP by applying operational management, which supports the overall ALARP demonstration addressed in Chapter 33.
Chapter 32 Emergency Preparedness	Chapter 32 demonstrates that the safety risks are reduced to the level that is ALARP by in the consideration of emergency preparedness, which supports the overall ALARP demonstration addressed in Chapter 33.

33.3 Applicable Codes and Standards

For Generic Design Assessment (GDA), there is a fundamental requirement for the Requesting Party to set out the process to reduce risk to a level that is ALARP. Demonstrating that risk has been reduced to a level that is ALARP is also a mandatory UK requirement. The following have been considered during the development of ALARP methodology and the ALARP demonstration of the UK HPR1000:

a) The Health and Safety at Work etc. Act, 1974.

b) The Ionising Radiation Regulation, 1999.

c) Management of Health and Safety at Work Regulations, 1999.

d) Reducing Risks, Protecting People, 2001.

e) The Tolerability of Risk from Nuclear Power Stations, 1992.

In addition, *Safety Assessment Principles* (SAPs), Reference [8] and the *Guidance on the Demonstration of ALARP*, Reference [9], provide a useful insight for the Requesting Party to understand how a safety case is evaluated and specifically, how a proposed design is considered to have demonstrated the risk is ALARP. Therefore, the ALARP methodology is developed against the background of requirements from Reference [9], which gives detailed information about the requirements on ALARP demonstration.

33.4 ALARP Methodology for the UK HPR1000

This sub-chapter outlines the methodology for the assessment of the generic design of the UK HPR1000 to determine whether the nuclear safety risks of the construction, operation and decommissioning phases are ALARP. The approach is developed against the background of the legal requirements and *in SAPs*, Reference [8].

The ALARP process includes consideration of the four areas that are common to the demonstration of ALARP in the UK, Reference [9]:

a) Comparison with RGP.

b) Identification and evaluation of options (optioneering).

c) Risk assessment, as a way of understanding the significance of the issue to the overall demonstration of ALARP.

d) Implementation of all reasonably practicable improvements.

An overview of the ALARP approach is presented in F-33.4-1. There are three parts for the assessment:

a) Part 1 is a holistic review, where the plant as a whole is reviewed to identify the areas for potential improvement.

b) Part 2 is a specific review of each area for potential improvement to implement measures that are considered to be reasonably practicable;

c) Part 3 is the holistic evaluation of whether the implementation of these improvements has reduced risks to levels that are ALARP. This requires iterations.

Part 1 is a holistic process aimed at identifying potential gaps along with their significance, and then identifying items for improvement (hereafter it is simply referred to as “potential improvements”). By looking at the design of the whole plant, the following steps are designed to identify potential improvements:

a) Step 1: A review of the evolution of the HPR1000 design is carried out to demonstrate that safety improvements have been incorporated into the design, and that relevant Operating Experience (OPEX) has been considered. The historical review of the HPR1000 is presented in Sub-chapter 33.5.

b) Step 2: Systematic review of UK HPR1000 design is the step where the design of the UK HPR1000 is systematically reviewed against RGP and OPEX to identify potential improvements. Once available, insights from Probabilistic Safety Assessment (PSA) provide additional information on potential improvements. Step 2 is therefore divided into 2 sub-steps to address the holistic review from these two points of view:

1) Step 2.1: Systematic review of design against RGP and OPEX, where RGP and OPEX is used as a basis for undertaking a review of the holistic design to identify potential improvements. The summary of the systematic review is presented in Sub-chapter 33.6, including the preliminary reviews against RPTs and engineering principles.

2) Step 2.2: Insights from the PSA of the Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3 (HPR1000 (FCG3)), and the subsequent development of the PSA model for the UK HPR1000, identify aspects of the plant where potential improvements could help reduce the overall plant risk.

c) Step 3: Collate the potential improvements is the step where all of the identified potential improvements following the Step 2 review are collected and managed according to the GDA configuration management procedures, References [10] and [11]. Potential improvements that have been identified are presented in Sub-chapter 33.6.

When potential improvements are identified, these may be grouped for convenience and then processed through Steps 4 to 6 (Part 2), with an individual assessment produced for each potential improvement (or group of potential improvements):

a) Step 4: Optioneering is the process of generation and evaluation of options which implement the specific potential improvement. The following steps can be followed when performing the optioneering process:

1) Define and characterise the specific potential improvement, where the fundamental issue, including its safety significance and potential implications (i.e. risk profile), should be understood and established.

2) Develop the potential options to address the problem, where a broad range of options should be considered, by applying the Defence in Depth (DiD) principle, to take account of multiple factors to prevent, protect and mitigate the risk identified in the previous step.

3) Assess the options in terms of their benefits and dis-benefits, where each option should be evaluated systematically and its relative merits identified. It is possible that this step may require a number of iterations as each option may require further information to support judgments made.

b) Step 5: Implementation of reasonably practicable options, where the option(s) carried forward from Step 4 are considered for implementation / further analysis or rejected until a suitable solution is reached.

c) Step 6: ALARP position justified for specific potential improvement, where a justification to explain that the associated risk is reduced to ALARP at the time of assessment is provided.

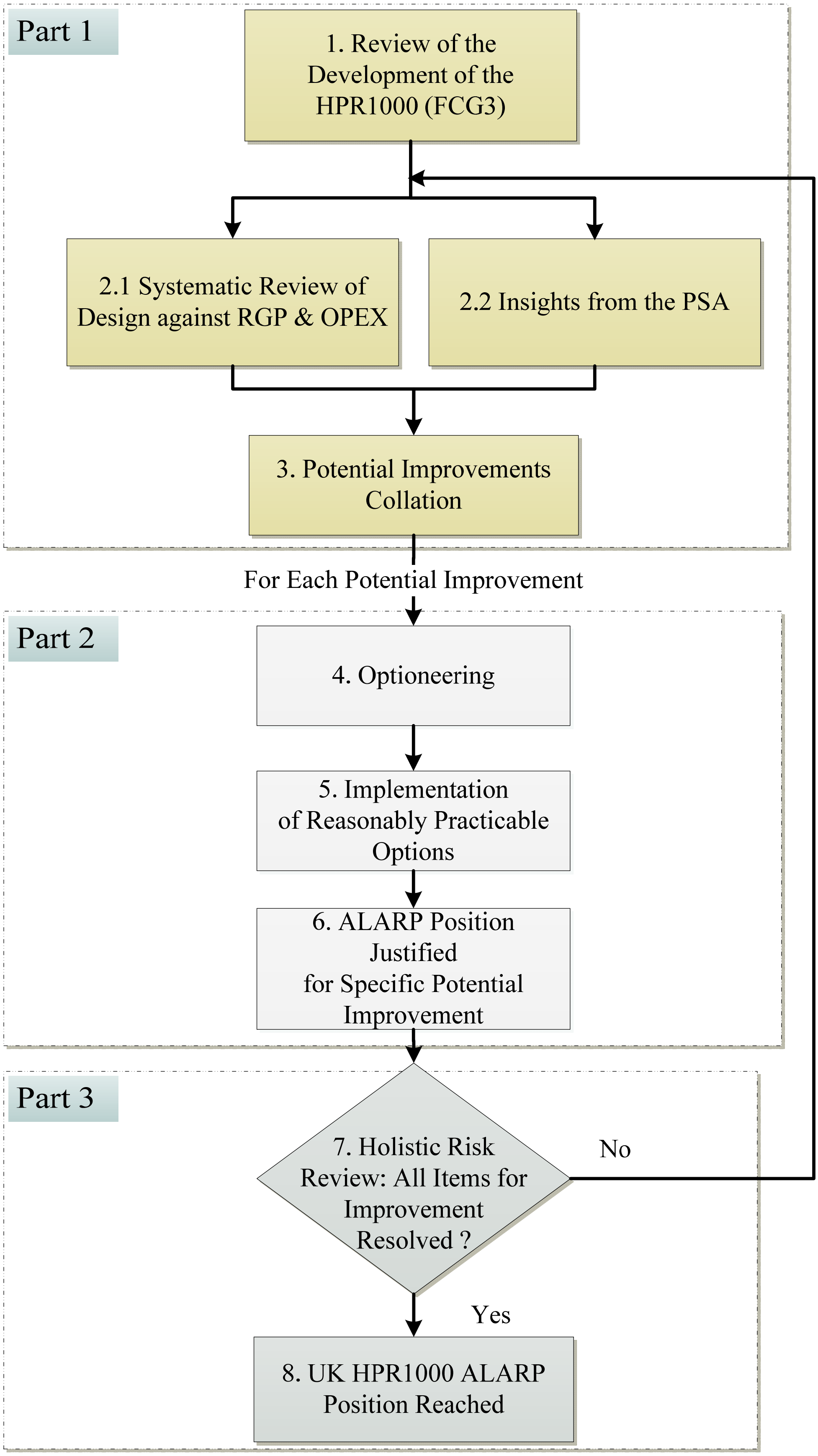
The GDA configuration management procedures should include the verification of whether the design option chosen can reduce the risk to ALARP. The procedures should be followed when performing the potential design modifications.

The process of optioneering and implementation of the practicable options shall continue whilst there are potential areas for improvement (Part 3).

a) Step 7: Holistic risk review. Once all of the potential improvements have been assessed, and a suitable (ALARP) solution has been implemented for each potential improvement, the design is subject to a further holistic review (Step 2). The design goes through the whole process to identify any further potential improvements, unless it is demonstrated that all options for improvement are resolved.

b) Step 8: UK HPR1000 ALARP position reached. Where there are no further reasonably practicable options to implement, and no further identified areas for potential improvement, the design is considered as optimised, reflecting UK expectations, and the safety risks from the UK HPR1000 design are considered ALARP.

Further guidance is included in the *ALARP Methodology* document, Reference [4].



F-33.4-1 Overview of the UK HPR1000 ALARP Methodology

33.5 Historical Design Process

Following the ALARP methodology presented in Sub-chapter 33.4, the historical design process of the HPR1000 has been reviewed. Similar to the concept of ALARP, the idea of reducing risk and optimising the design was incorporated in the HPR1000 historical development, focussing notably on the design of the HPR1000 (FCG3). This sub-chapter summarises the historical development process, from the CPR1000 to Chinese Improved Pressurised Reactor (CPR1000$^+$), Advanced Chinese Pressurised Reactor (ACPR1000) and finally the HPR1000. The historical process has been sub-divided into two phases:

a) Continuous improvement of the second generation design, from CPR1000 to ACPR1000;

b) Development of the Generation III HPR1000 from the ACPR1000.

33.5.1 Historical Development from CPR1000 to ACPR1000

In the 1980s, M310 nuclear reactor technology was imported in China from Framatome, France, and the first large commercial NPP was built in China, Daya Bay. Derived from the CPY three-loop design of Framatome, the M310 design has been modified in both safety and economic aspects to address the operational experience feedback from the Three Mile Island accident. The M310 design was considered as an advanced nuclear reactor design at that time.

Since then, continuous improvement and design modifications were performed. By considering operational experience feedback from similar nuclear power plants and applying new proven technologies and new codes and standards, a series of major modifications were determined and implemented to form the design of CPR1000. CPR1000 demonstrated a safer design. For example, the following safety related modifications were implemented, which are also considered in the HPR1000 design:

a) Adoption of one-piece forging for Reactor Pressure Vessel (RPV) core shell to improve the mechanical properties by eliminating the circumferential welding in active area.

b) Application of digital Instrumentation and Control (I&C) systems and an advanced Main Control Room (MCR) to provide an improved human machine interface which reduces the risk of human errors.

c) Addition of a combustible gas control system to reduce the risk of hydrogen explosion.

In 2009, the CPR1000$^+$ design was established based on the CPR1000 design by implementing a number of major modifications with due consideration of improving the nuclear safety (especially in the aspects of severe accident mitigation), optimising the waste treatment design, improving the reliability and economics and facilitating the operation and maintenance.

The following examples of modifications are also inherited by the HPR1000 design:

a) Application of In-Vessel Retention (IVR) strategy to prevent most ex-vessel phenomena which could challenge the integrity of containment under severe accident conditions.

b) Optimisation of the hydrogen recombiner configuration to improve the reliability by applying the passive autocatalytic recombiner. The hydrogen concentration can be controlled under design basis conditions and design extension conditions to prevent hydrogen explosion and to ensure the containment integrity.

c) Modifications to the radioactive waste treatment systems by applying new proven techniques to reduce the production of radioactive wastes.

d) By optimising the habitable area and modifying the ventilation systems, the MCR habitability was improved, which eventually allowed for improved control of the reactor under severe accident conditions.

e) Addition of safety classified and seismically qualified isolation valves for each train of the main feedwater line to ensure the isolation of main feedwater, which reduces the release of mass and energy in the containment and helps to prevent the introduction of positive reactivity in the reactor core due to the overcooling of the primary circuit under fault and accident conditions.

On March 11, 2011, the well-known Great East Japan Earthquake and the subsequent Tsunami struck the Japan Fukushima Daiichi site, causing severe damage to the reactors. To sufficiently consider the lessons learnt from the Fukushima accident and to meet the new national requirements, China General Nuclear Power Corporation (CGN) proposed a number of major improvements based on the design of CPR1000^{+}. The improvements were identified to address the lessons learnt from the Fukushima accident, such as the consistency review against the latest nuclear safety requirements, and insights from the full scope PSA. By implementing a number of modifications identified, the ACPR1000 design demonstrated higher reliability, safety and the ability to manage beyond design basis conditions similar to the Fukushima accident, which possessed the major safety characteristics of the Generation III nuclear power plant.

Notable examples of the major improvements are presented below:

a) Addition of extra heat removal measures to cool the containment and the spent fuel pool under loss of ultimate heat sink condition, which corresponds to the Extra Cooling System (ECS [ECS]) of the UK HPR1000.

b) Addition of a Station Black Out (SBO) diesel generator to protect against the SBO condition. One SBO diesel generator was implemented for each unit, which provides further DiD measures to ensure the reliability of the power supply under

SBO.

c) Addition of a severe accident dedicated valve performing the depressurisation of the primary circuit to prevent core melt under high pressure.

d) Modification on the reactor coolant pump seal to ensure the integrity of the Reactor Coolant System (RCP [RCS]) in the event of SBO.

e) Addition of the Diverse Actuation System (KDS [DAS]) to cope with the frequent faults combined with common cause failure in RPS [PS].

f) The Severe Accident I&C (KDA [SA I&C]) and the Uninterruptible Power Supply (UPS) were implemented to provide dedicated monitoring measures for severe accidents, which demonstrated the improved monitoring and control of the NPP.

Details regarding the modifications during the development from CPR1000 to ACPR1000 are presented in the *HPR1000 R&D History*, Reference [5].

33.5.2 Historical Development of HPR1000 from ACPR1000

Based on the ACPR1000 design, a balanced Generation III HPR1000 was developed with due consideration of the development needs in internationalisation and the future reactor type upgrading. The HPR1000 design has successfully completed the review held by National Nuclear Safety Administration (NNSA) and the Generic Reactor Safety Review held by IAEA.

Compared with the ACPR1000 design, the single-reactor scheme HPR1000 has better site foundation adaptability and higher safety and operating performance. It incorporates the concept of DiD more comprehensively, and in particular, further strengthens severe accidents prevention and mitigation. It integrates advanced design ideas of Generation III nuclear technologies, the experience of the design, construction, commissioning and operation regarding pressurised water reactors in China, and the achievements of nuclear power development and research in recent years. The following major design modifications have been implemented in the HPR1000 (FCG3):

a) Reactor Core

The reactor core of the HPR1000 design is composed of 177 advanced fuel assemblies, compared to 157 fuel assemblies for the ACPR1000. The HPR1000 reactor core is provided with a lower average linear power density and higher thermal and hydraulic margin. In addition, benefiting from the higher thermal and hydraulic margin, the HPR1000 reactor core is capable of providing a potential rise of output power in the future, which presents significant economic advantages.

b) Reactor Coolant System

The design of the RCP [RCS] ensures the reactivity control provided by the reactor coolant water, the heat removal from the core to the secondary cooling side via reactor coolant, and the confinement of radioactive material. The design of the RCP [RCS] has been improved based on the ACPR1000 design with due consideration of proven techniques and the wide experience from the design, manufacture, construction, commissioning and operating feedback of existing NPPs. The following examples are the major modifications implemented during the development of the HPR1000 design:

1) Increase in the volume of the Pressuriser (PZR). The PZR controls the pressure of the RCP [RCS] during normal operation and transient operation of the NPP. By increasing the volume of the PZR, the pressure control ability is improved, which helps reduce the risk of overpressure and maintain the integrity of the primary pressure boundary.

2) Increase in the volume of secondary side Steam Generator (SG). SGs serve as the first means for heat removal from the reactor. By increasing the secondary side volume in the SGs, the inherent safety of the HPR1000 can be improved. The larger volume of the SGs contributes to improving the ability of temperature control and improving the autonomy of the SGs. For example, in Steam Generator Tube Rupture (SGTR) conditions, the larger secondary side steam volume of the SGs can prevent the overfill of the affected SG, which improves the resilience to transients and accidents.

3) Eliminating the penetrations in the RPV lower head. The RPV is the highest reliability pressure boundary, and contains the reactor core, core support structure and water coolant. In the ACPR1000 RPV design, measuring instrumentation adapters penetrate the lower head of the RPV, which may challenge the integrity of the pressure boundary and may increase the risk of leakage from the lower head. By considering the international research feedback, lower head penetrations are eliminated in the HPR1000 design and the measuring instrumentation adapters are implemented on the closure head on top of the RPV, which improves the integrity of the highest reliability pressure boundary.

c) Safety Systems

In the ACPR1000 design, the safety systems consist of two independent trains, but these two trains are connected with a common header. To improve the reliability of the safety systems, their configuration has been improved. Considering the case where one train of a safety system fails due to an initiating fault and the second train fails taking the single failure criterion into account, a third redundant train is needed to ensure that the required safety functions are performed reliably. As a result, three independent, segregated safety divisions are designed for design basis conditions with a capacity of 3×100%. The design of the three independent, segregated safety divisions meets the single failure criterion, demonstrates higher reliability of safety systems and reduces the risks related to common cause failures and internal hazards. The modifications involve the Safety Injection System (RIS [SIS]), Emergency Boration System (RBS [EBS]), Emergency Feedwater System (ASG [EFWS]) and the related supporting systems, such as Component Cooling Water System (RRI [CCWS]), Essential Service Water System (SEC [ESWS]) and Emergency Diesel Generators (EDGs).

For the safety features designed for design extension conditions, one train of safety features was designed for the ACPR1000. To improve resilience to design extension conditions, two independent and redundant trains of safety features have been implemented in the HPR1000 design. The safety features include Containment Heat Removal System (EHR [CHRS]), Extra Cooling System (ECS [ECS]) and the Pressuriser Safety Valve. In addition, for the ACPR1000 design, the Secondary Passive Heat Removal System (ASP [SPHRS]) was added to cope with the total loss of feedwater and other design extension conditions. In HPR1000 design, the large capacity water tank of the ASP [SPHRS] was implemented to improve the autonomy of the nuclear power plant for 72 hours.

Details about the modifications to the safety systems are provided in *HPR1000 R&D History*, Reference [5].

d) Electric Power

To adapt to the modifications implemented in the mechanical and process systems, and to ensure the reliability of electric power supplies, adaptive optimisations have been performed in electrical power systems.

Based on the ACPR1000 design, the two-train emergency power supply is modified to the independent three-train emergency power supply in the HPR1000 design. The modifications in the electrical power systems improve the reliability of the power supply for the three-train safety systems.

To further improve the reliability of the power supply, an additional SBO diesel generator has been added to the HPR1000, which further improves the autonomy of the power supply for design extension conditions and the ability to respond to a SBO.

Considering the principle of DiD, the current electrical power system of the HPR1000 (FCG3) consists of the main power supply, auxiliary power supply, 3 emergency diesel generators, 2 SBO diesel generators and a UPS for 2 hours and another for 12 hours. This improves the reliability of the electrical power supply.

With the objective of further reducing the risk from human errors, the MCR design was optimised based on the ACPR1000 design. An Auxiliary Control Panel (ACP) is utilised in HPR1000 MCR design. The ACP works as the auxiliary monitoring and control means in case of unavailability of Plant Computer Information and Control System (KIC [PCICS]). The digital design of ACP provides a reduced size of the system compared with the conventional hardwired backup means provided for the ACPR1000, which facilitates the optimisation of the MCR layout design and provides advanced human-machine interface.

Aligning with the improvement made to the RPV through eliminating the lower head penetrations, the HPR1000 In-core Instrumentation System (RIC [IIS]) uses integrated instrumentation assemblies inserted from the top of the RPV to achieve the measurement of neutron flux, temperature and reactor coolant level. In ACPR1000 design, the measurement of the parameters via RIC [IIS] was performed periodically and manually by inserting instrumentation assemblies from the lower head into the core. In the HPR1000 design, the modified RIC [IIS] can perform real time measurement of core parameters. The improvements made to the RIC [IIS] help the operators to be aware of plant parameters and the safety margin of the reactor core.

e) Hazard Protection

In addition to the system modifications implemented, the HPR1000 demonstrates a more comprehensive, detailed and systematic hazard analysis and hazard protection design compared with the ACPR1000, especially for the commercial aircraft crash, earthquake and external flooding hazards. For example, to prevent the effects of common cause failure on the safety systems due to internal hazards, the safety systems are implemented in segregated buildings. Double-walled (internal and external) containment is designed to protect against external hazards, including large commercial aircraft crash.

Detailed descriptions regarding the modifications are provided in *HPR1000 R&D History*, Reference [5].

33.5.3 Conclusion on the Design Evolution

With the objective of further enhancing the design of each level of DiD and continuously improving the design reliability and nuclear safety, the historical development of HPR1000 represents a process of continuous improvement by considering RGP, OPEX and international research outcomes. As presented above, the continuous development of the HPR1000 and the continuous risk reduction process have led to an optimised design with nuclear safety significantly increased, which forms the basis of UK HPR1000 design.

33.6 ALARP Review of the UK HPR1000

The ALARP review of the UK HPR1000 follows the ALARP methodology presented in Sub-chapter 33.4. A summary of Step 1 of the methodology used for the review of the historical design process has been presented in Sub-chapter 33.5. Specific application of the ALARP demonstration for different disciplines of UK HPR1000 is presented in ALARP demonstration reports for each topic area, according to the guidance of *ALARP Demonstration Instruction*, Reference [12]. This sub-chapter provides the ALARP review of the UK HPR1000 following completion of Step 2 to Step 8. A summary of the outcomes from each step and the strategy for the steps that are not completed are provided.

33.6.1 Holistic Review: Systematic Review of UK HPR1000 Design

The main objective of the first part of the ALARP methodology, the holistic review, is to identify the areas for potential improvement by looking at the design of the whole plant. The review of the UK HPR1000 design includes the consistency review against RGP and OPEX (Step 2.1 of the ALARP methodology) and the insights of risk analyses (Step 2.2 of the ALARP methodology). Identification of the potential improvements is the basis to show that no further reasonably practicable improvements can be implemented.

33.6.1.1 Systematic Review of Design against RGP and OPEX

In each topic area of the UK HPR1000, RGP and relevant OPEX have been identified. The sub-chapters on Applicable Codes and Standards summarise the identification and analysis of applicable codes and standards that can be considered as RGP in the PCSR Chapters.

The systematic review against RGP/OPEX and identification of potential improvements in each topic area has been preliminarily completed and potential improvements have been identified through this process. Detailed information is provided in the ALARP demonstration reports for each topic area and is summarised in Reference [6]. During the GDA process, RGP identification and compliance analysis will be carried out continually. In addition, any lessons learned during the next steps of GDA will be taken into account and relevant information will be supplemented accordingly.

33.6.1.2 Insights from Risk Assessment

Following the progress in fault analysis, especially the progress in risk assessment, the areas for potential improvements can be identified by considering the insights from risk analyses, including fault analysis, PSA, hazard analysis, human factors analysis, structural integrity analysis, chemistry analysis, radiological protection analysis and conventional safety analysis, as explained in Reference [12]. Insights from risk analyses for which potential improvements could be put forward have been obtained and transferred to relevant downstream disciplines according to *Risk Control Specifications for GDA Project*, Reference [13]. The consideration of risk assessment insights has been taken into account and a number of potential improvements have been identified through this process. Detailed information is provided in the ALARP demonstration reports for each topic area and is summarised in Reference [6]. The identification and transfer of insights will continue as the risk assessment process develops.

33.6.1.3 Collation of Potential Improvements

Potential improvements are managed by the project risk management system. Potential improvements identified in the previous steps of the ALARP methodology have been collated at the chief engineer review meeting and registered in the project risk register according to Reference [13]. Specific assessment results are described in ALARP demonstration reports for different topic areas and are summarised in Reference [6].

33.6.2 Specific Review of Potential Improvements

To determine further reasonably practicable options to reduce risks and improve safety, the identified potential improvements are investigated and the corresponding optioneering performed. If appropriate, the potential improvements are incorporated into the design change process following the process proposed in the ALARP methodology (Part 2: specific review) and the GDA configuration management procedures, References [10] and [11].

During the GDA process, a batch of optioneering has been performed, and further options have been considered across all topic areas and have resulted in a number of important design changes. Decisions taken to implement these design changes have contributed to the ALARP demonstration for the UK HPR1000.

Further information on potential improvements and design changes is presented in Reference [6].

33.6.3 Strategy for Iterative Holistic Evaluation

Once all of the potential improvements have been assessed, and a suitable (ALARP) option implemented for each potential improvement, the design is subject to a further holistic review (Part 3 of the ALARP methodology).

33.7 Assessments of Radiation Protection Targets

Radiation Protection Targets 1-9 (RPTs 1-9) are set out in PCSR Chapter 4 Sub-chapter 4.4.1.3. Assessments against radiation protection targets provide a quantitative assessment of the risk levels. This review against these targets will also serve as an important source to indicate the areas of potential improvements. Demonstrating that the radiation protection targets are achievable is an essential element to support the claim that the nuclear safety risks are ALARP.

The radiation protection targets are in the form of dose levels, frequencies of occurrence, or risk of death. Each target is set in terms of a Basic Safety Level (BSL) and a Basic Safety Objective (BSO). The performance of the UK HPR1000 against the targets is summarised in this sub-chapter.

33.7.1 Assessment of Radiation Protection Targets for Normal Operation

a) Assessment of RPT 1

RPT 1 sets the targets and limits for effective dose from sources of ionising radiation in a calendar year for any person on the site. The assessment progress is presented in PCSR Chapter 22 Sub-chapter 22.9.

The preliminary evaluation result of UK HPR1000 for employees working with ionising radiation is between the BSL and BSO, and the preliminary evaluation result of the UK HPR1000 for other employees on the site is below the BSO value, presented in T-33.7-1

T-33.7-1 RPT 1: Normal Operation-Any Person on the Site

Employees working with ionising radiation	
BSL	20 mSv
BSO	1 mSv
UK HPR1000 value	Between BSL and BSO
Other employees on the site	
BSL	2 mSv
BSO	0.1 mSv
UK HPR1000 value	<BSO

b) Assessment of RPT 2

RPT 2 sets the targets for average effective dose in a calendar year to defined groups of employees on the site working with ionising radiation. The assessment progress is presented in PCSR Chapter 22 Sub-chapter 22.9.

The preliminary evaluation result of UK HPR1000 for normal operation in any group on the site is between the BSL and BSO, which are presented in T-33.7-2.

T-33.7-2 RPT 2: Normal Operation-Any Group on the Site

BSL	10 mSv
BSO	0.5 mSv
UK HPR1000 value	Between BSL and BSO

c) Assessment of RPT 3

RPT 3 sets the target and the limit for effective dose from sources of ionising radiation originating on the site in a calendar year for any person off the site. RPT 3 has been interpreted and analysed using the Best Available Technique (BAT) approach which is similar to the ALARP approach.

The preliminary evaluation result of the UK HPR1000 for normal operation-any person off the site is between the BSL and BSO, which are presented in T-33.7-3 with the detailed supporting analysis presented in Sub-chapter 7.6 of PCER Chapter 7 *Radiological Assessment*, Reference [7].

T-33.7-3 RPT 3: Normal Operation-Any Person off the Site

BSL	1 mSv
BSO	0.02 mSv
UK HPR1000 value	Between BSL and BSO

33.7.2 Assessment of Radiation Protection Targets for Fault and Accident Condition

The radiation protection targets for fault and accident conditions correspond to numerical targets 4 to 9 in the *Safety Assessment Principles*:

a) Assessment of RPT 4

RPT 4 sets the targets for the effective dose received by any person arising from design basis fault sequences. The targets are presented in T-33.7-4. The assessment against RPT 4 is still in progress and will be presented in Chapter 12 Sub-chapter 12.14.

T-33.7-4 RPT 4: Design Basis Fault Sequences-Any Person

On site	
BSL	20 mSv for initiating fault frequencies exceeding 1×10^{-3} pa
	200 mSv for initiating fault frequencies between 1×10^{-3} pa and 1×10^{-4} pa
	500 mSv for initiating fault frequencies between 1×10^{-4} pa and 1×10^{-5} pa
BSO	0.1 mSv
Off site	
BSL	1 mSv for initiating fault frequencies exceeding 1×10^{-3} pa
	10 mSv for initiating fault frequencies between 1×10^{-3} pa and 1×10^{-4} pa
	100 mSv for initiating fault frequencies between1×10^{-4} pa and1×10^{-5} pa

BSO	0.01 mSv

b) Assessment of RPT 5

RPT 5 gives the individual risk of death from accidents to any person on the site. The targets are presented in T-33.7-5. The ALARP assessment against RPT 5 will be carried out and presented in Chapter 14 Sub-chapter 14.10.

T-33.7-5 RPT 5: Individual Risk of Death from Accidents-Any Person on the Site

BSL	1×10^{-4} pa
BSO	1×10^{-6} pa

c) Assessment of RPT 6

RPT 6 gives the predicted frequency of any single accident in the facility which could give a dose to a person on the site. The targets are presented in the table T-33.7-6. RPT 6 is under evaluation for the UK HPR1000 with the progress and working plan presented in PCSR Chapter 14 Sub-chapter 14.10.

T-33.7-6 RPT 6: Frequency Dose Targets for Any Single Accident-Any Person on the Site

Effective dose (mSv)	Predicted frequency per annum	
	BSL	BSO
2-20	1×10^{-1}	1×10^{-3}
20-200	1×10^{-2}	1×10^{-4}
200-2000	1×10^{-3}	1×10^{-5}
>2000	1×10^{-4}	1×10^{-6}

d) Assessment of RPT 7

RPT 7 sets the targets for the individual risk of death to a person off the site from accidents at the site resulting in exposure to ionising radiation. The targets are presented in T-33.7-7. RPT 7 is supported by RPT 8 in the form of a dose-frequency staircase derived from RPT 7. The assessment against RPT 7 is still in progress as the assessment against RPT 8 is not accomplished. The assessment against RPT 7 will be presented in Chapter 14 Sub-chapter 14.10.

T-33.7-7 RPT 7: Individual Risk of Death to People off the Site from Accidents

BSL	1×10^{-4} pa

BSO	1×10^{-6} pa

e) Assessment of RPT 8

RPT 8 sets the targets for the total predicted frequencies of accidents on an individual facility which could give a dose to a person off the site. The targets are presented in T-33.7-8. Numerical RPT 8 is addressed in Chapter 14 Sub-chapter 14.10.

T-33.7-8 RPT 8: Frequency Dose Targets for Accidents on an Individual Facility – Any Person off the Site

Effective Dose (mSv)	Total Predicted Frequency per Annum	
	BSL	BSO
0.1-1	1	1×10^{-2}
1-10	1×10^{-1}	1×10^{-3}
10-100	1×10^{-2}	1×10^{-4}
100-1000	1×10^{-3}	1×10^{-5}
>1000	1×10^{-4}	1×10^{-6}

f) Assessment of RPT 9

RPT 9 gives the targets for the total risk of 100 or more fatalities, either immediate or eventual, from accidents at the site resulting in exposure to ionising radiation. The targets are presented in T-33.7-9. RPT 9 assesses the societal risk which may be caused by a severe accident at the facility. The assessment against RPT 9 will be presented in Chapter 14 Sub-chapter 14.10.

T-33.7-9 RPT 9: Total Risk of 100 or More Fatalities

BSL	1×10^{-5} pa
BSO	1×10^{-7} pa

The assessments of the RPTs for fault and accident condition are on-going.

33.8 Concluding Remarks

Following the UK requirements, reducing risks ALARP should be performed continuously for the UK HPR1000. To achieve this objective, ALARP assessment is performed in the PCSR Chapters listed in T-33.2-1. The sub-chapter ‘ALARP Assessment’ in these PCSR Chapters summarises the progress and further work related to the ALARP demonstration process.

To demonstrate that nuclear safety risks are reduced to ALARP, a suitable methodology has been established to support the review of the design and the identification and implementation of reasonably practicable options to reduce risks. By applying the ALARP methodology, a historical review of the HPR1000 design has been provided to demonstrate that the development process of HPR1000 can lead to an optimised and balanced design. The review of UK HPR1000 design against RGP/OPEX together with insights from risk assessments is being performed and a number of potential improvements have been identified and collated. Specific reviews for the identified potential improvements have been carried out and the optioneering process for some potential improvements is on-going. The assessment of the RPTs for normal operation is completed and the assessment of the RPTs for fault and accident conditions is on-going.

Other than the progress mentioned above, there may be more that can reasonably be done to reduce the risks further. Following the GDA stage, the ALARP methodology will be applied to further perform a holistic review of the plant to ensure that all reasonably practicable measures have been adopted and no further optimisation is appropriate for the design. The assessment of the RPTs will be summarised to demonstrate that the nuclear safety risks are reduced to the level that is ALARP.

33.9 References

[1] The Stationery Office, Health and Safety at Work etc. Act 1974, 1974.

[2] Health and Safety Executive, Reducing Risks, Protecting People, 2001.

[3] Health and Safety Executive, The Tolerability of Risk from Nuclear Power Stations, 1992.

[4] CGN, ALARP Methodology, GHX00100051DOZJ03GN, Revision B, April 2018.

[5] CGN, HPR1000 R&D History, GHX99980001DXZJ01MD, Revision B, August 2019.

[6] CGN, Holistic ALARP Demonstration Report, GHX00100071KPGB03GN, Revision A, December 2019.

[7] General Nuclear System Limited, Pre-Construction Environment Report Chapter 7 Radiological Assessment, HPR/GDA/PCER/0007, Revision 001, January 2020.

[8] Office for Nuclear Regulation (UK), Safety Assessment Principles for Nuclear Facilities, Revision 0, November 2014.

[9] Office for Nuclear Regulation (UK), Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable), NS-TAST-GD-005, Revision 9, March 2018.

[10] CGN, Provisions on Configuration Management for UK HPR1000 Generic Design Assessment (GDA) Project, GH-40M-009, Revision A, September 2018.

[11] CGN, Provisions on Configuration Change Management for UK HPR1000 Generic Design Assessment (GDA) Project, GH-40M-012, Revision C, November 2019.

[12] CGN, ALARP Demonstration Instruction, GHX00100119DOZJ03GN, Revision A, March 2019.

[13] CGN, Risk Control Specifications for GDA Project, GH-40M-002, Revision B, July 2019.