New Reactor Division – Generic Design Assessment

Step 2 Assessment of the Fuel & Core Design of the UK HPR1000 Reactor

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EXECUTIVE SUMMARY

This report presents the results of my Fuel & Core assessment of the UK HPR1000 undertaken as part of Step 2 of the Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA).

The GDA process calls for a step-wise assessment of the Requesting Party's (RP) safety submission with the assessments increasing in detail as the project progresses. Step 2 of GDA is an overview of the acceptability, in accordance with the regulatory regime of Great Britain, of the design fundamentals, including ONR's review of key nuclear safety and nuclear security claims (or assertions). The aim is to identify any fundamental safety shortfalls that could prevent ONR from permitting the construction of a power station based on the design.

During GDA Step 2 my work has focused on the assessment of the Fuel & Core aspects within the UK HPR1000 Preliminary Safety Report (PSR), and a number of supporting references and supplementary documents submitted by the RP, focusing on design concepts and claims.

The standards I have used to judge the adequacy of the RP's submissions in the area of Fuel & Core have been primarily ONR's Safety Assessment Principles (SAPs), in particular SAPs ERC.1 to 4 and EKP.1 to 5, and ONR's Technical Assessment Guides NS-TAST-GD-005 Guidance on the demonstration of ALARP, NS-TAST-GD-051, The purpose, scope and contents of nuclear safety cases and NS-TAST-GD-075 Safety of nuclear fuel in power reactors. I have also made use of the other relevant standards and guidance issued by IAEA.

My GDA Step 2 assessment work has involved continuous engagement with the RP in the form of technical meetings, including meetings with the plant designers.

The UK HPR1000 PSR is primarily based on the Reference Design, Fangchenggang Nuclear Power Plant Unit 3 (FCG3), which is currently under construction in China. Key aspects of the UK HPR1000 preliminary safety case related to Fuel & Core, as presented in the PSR, its supporting references and the supplementary documents submitted by the RP, can be summarised as follows:

- Description of the major design features of the Fuel & Core design for FCG3;
- Definition of the most important functional requirements and design criteria applied in the FCG3 design; and
- Identification of the critical degradation mechanisms for the fuel and core structures, systems and components.

It should be noted that the although the FCG3 design features provide initial visibility, the actual assessment in GDA Step 3 can only progress when the actual details of the UK HPR1000 fuel and core design and safety justification have become available to ONR.

During my GDA Step 2 assessment of the UK HPR1000 aspects of the safety case related to Fuel & Core I have identified the following main areas of strength:

- The safety functions, functional requirements and design criteria are detailed in the PSR to a level sufficient for GDA Step 2.
- The main degradation mechanisms are appropriately identified.
- The safety functions and main functional requirements related to the spent fuel interim storage (SFIS) outside the reactor building are correctly outlined.

During my GDA Step 2 assessment of the UK HPR1000 aspects of the safety case related to Fuel & Core I have identified the following main areas that require follow-up:

- The design criteria do not meet ONR's requirements to maintain fuel integrity in Frequent Faults and protections against PCI & PCMI need to be demonstrated.
- The design of the fuel has not yet been finalised and therefore it is not possible to plan my assessment for Step 3 in detail at this stage.
- The models, computer codes, uncertainty and safety margins of the Fuel & core analysis have not been presented.
- SFIS design concept and safety justification are not yet developed.

During my GDA Step 2 assessment, I have not identified any fundamental safety shortfalls in the area of Fuel & Core that might prevent the issue of a Design Acceptance Confirmation (DAC) for the UK HPR1000 design.

LIST OF ABBREVIATIONS

ALARP	As Low As Reasonably Practicable
BAT	Best Available Technique
BMS	Business Management System
BSL	Basic Safety Level (in SAPs)
BSO	Basic Safety Objective (in SAPs)
CGN	China General Nuclear Power Corporation
DAC	Design Acceptance Confirmation
DBC	Design Basis Condition
DEC	Design Extension Condition
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EA	Environment Agency
EDF	Électricité de France
FA	Fuel Assembly
FCG3	Fangchenggang Nuclear Power Plant Unit 3
FHSS	Fuel Handling and Storage System
FTT	Fuel Transfer Tube
GNS	Generic Nuclear System Ltd
IAEA	International Atomic Energy Agency
K_{eff}	Neutron transport equation principle Eigen Value for the domain
NPP	Nuclear Power Plant
ONR	Office for Nuclear Regulation
PCI	Pellet – Cladding Interaction
PCSR	Pre-construction Safety Report
PSR	Preliminary Safety Report
RAPFE	Radially-averaged Peak Fuel Enthalpy
RCCA	Rod Cluster Control Assembly
RGP	Relevant Good Practice

RIA	Reactivity Induced Accidents
RP	Requesting Party
RQ	Regulatory Query
SAP(s)	Safety Assessment Principle(s)
SFAIRP	So far as is reasonably practicable
SFA	Spent Fuel Assembly
SFIS	Spent Fuel Interim Storage
SFP	Spent Fuel Pool
SSE	Safe Shut-down Earthquake
TAG	Technical Assessment Guide(s)
TSC	Technical Support Contractor
WENRA	Western European Nuclear Regulators’ Association

TABLE OF CONTENTS

Tables

1 INTRODUCTION

1. The Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA) process calls for a step-wise assessment of the Requesting Party's (RP) safety submission with the assessments increasing in detail as the project progresses. General Nuclear System Ltd (GNS) has been established to act on behalf of the three joint requesting parties (China General Nuclear Power Corporation (CGN), Électricité de France (EDF) and General Nuclear International (GNI)) to implement the GDA of the UK HPR1000 reactor. For practical purposes GNS is referred to as the 'UK HPR1000 GDA Requesting Party'.

2. During Step 1 of GDA, which is the preparatory part of the design assessment process, the RP established its project management and technical teams and made arrangements for the GDA of the UK HPR1000 reactor. Also, during Step 1 the RP prepared submissions to be assessed by ONR and the Environment Agency (EA) during Step 2.

3. Step 2 commenced in November 2017. Step 2 of GDA is an overview of the acceptability, in accordance with the regulatory regime of Great Britain, of the design fundamentals, including ONR's assessment of key nuclear safety and nuclear security claims (or assertions). The aim is to identify any fundamental safety or security shortfalls that could prevent ONR permitting the construction of a power station based on the design.

4. My assessment has followed my GDA Step 2 Assessment Plan for Fuel & Core (Ref. 1) prepared in October 2017 and shared with GNS to maximise openness and transparency.

5. This report presents the results of my Fuel & Core assessment of the UK HPR1000 as presented in the UK HPR1000 Preliminary Safety Report (PSR) (Ref. 2).

2 ASSESSMENT STRATEGY

6. This section presents my strategy for the GDA Step 2 assessment of the Fuel & Core aspects of the UK HPR1000 (Ref. 2). It also includes the scope of the assessment and the standards and criteria I have applied.

2.1 Scope of the Step 2 Fuel & Core Assessment

7. The objective of my GDA Step 2 assessment was to assess relevant design concepts and claims made by the RP related to Fuel & Core. In particular, my assessment has focused on the following:

- The adequacy and completeness of the RPs safety claims related to Fuel & Core, and how they may be supported in future GDA Steps;
- The functional requirements and criteria applied by the RP in their design;
- How the RP justifies the safety of the design and how they intend to demonstrate that this reduces the risks to ALARP; and
- Any novel features of the UK HPR1000 Fuel & Core design (where, in accordance with Ref. 3, a novel feature is *"any major system, structure or component not previously licensed in a nuclear facility anywhere in the world")*.

8. During GDA Step 2 I have also evaluated whether the safety claims related to Fuel & Core are supported by a body of technical documentation sufficient to allow me to proceed with GDA work beyond Step 2.

9. Finally, during Step 2 I have undertaken the following preparatory work for my Step 3 assessment:

- Discussions with the RP regarding a schedule of submissions to support my assessment in later GDA Steps;
- Identification of the main data needed for confirmatory analyses of the Fuel & Core design in later GDA Steps; and
- Preliminary discussion of my Step 3 Assessment Plan with the RP.

2.2 Standards and Criteria

10. For ONR, the primary goal of the GDA Step 2 assessment is to reach an independent and informed judgment on the adequacy of a preliminary nuclear safety and security case for the reactor technology being assessed. Assessment was undertaken in accordance with the requirements of the Office for Nuclear Regulation (ONR) How2 Business Management System (BMS) guide NS-PER-GD-014 (Ref. 3).

11. In addition, the Safety Assessment Principles (SAPs) (Ref. 4) constitute the regulatory principles against which duty holders' and RP's safety cases are judged. Consequently the SAPs are the basis for ONR's nuclear safety assessment and have therefore been used for the GDA Step 2 assessment of the UK HPR1000. The SAPs 2014 Edition are aligned with the IAEA standards and guidance.

12. Furthermore, ONR is a member of the Western Regulators Nuclear Association (WENRA). WENRA has developed Reference Levels, which represent good practices for existing nuclear power plants, and Safety Objectives for new reactors.

13. The relevant SAPs, International Atomic Energy Agency (IAEA) standards and WENRA reference levels are embodied and expanded on in the ONR's Technical Assessment Guides (TAGs) which provide the principal means for assessing the Fuel & Core design and safety case in practice.

2.2.1 Safety Assessment Principles

14. The key SAPs (Ref. 4) applied within my assessment are SAPs which set expectations for the reactor core: ERC.1, ERC.2, ERC.3, and ERC.4. I have also applied and considered the SAPs related to inherent safety (EKP.1), fault tolerance (EKP.2), defence in depth (EKP.3), safety functions (EKP.4) and safety measures (EKP. 5) - (see also Table 1 for further details).

2.2.2 Technical Assessment Guides

15. The following Technical Assessment Guides have been used as part of this assessment (Ref. 5):

- NS-TAST-GD-005 - Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable);
- NS-TAST-GD-051 - The purpose, scope and content of nuclear safety cases; and
- NS-TAST-GD-075 - Safety of Nuclear Fuel in Power Reactors.

2.2.3 National and International Standards and Guidance

16. The following national and international standards and guidance have been considered as part of this assessment:

- Relevant IAEA standards (Ref. 6): Design of the Reactor Core for Nuclear Power Plants; and
- WENRA references (Ref. 7): Western European Nuclear Regulators' Association: Reactor Reference Safety Levels.

2.3 Use of Technical Support Contractors

17. During Step 2 I have not engaged Technical Support Contractors (TSCs) to support the assessment of the Fuel & Core for the UK HPR1000.

2.4 Integration with Other Assessment Topics

18. Early in GDA, I recognised the importance of working closely with other assessors (including Environment Agency's assessors) as part of the Fuel & Core assessment process. Similarly, other assessors sought input from my assessment of the Fuel & Core for the UK HPR1000. I consider these interactions are key to the success of the project in order to prevent or mitigate any gaps, duplications or inconsistencies in ONR's assessment. From the start of the project, I have endeavoured to identify potential interactions between the Fuel & Core and other technical areas, with the understanding that this position will evolve throughout the UK HPR1000 GDA.

19. The key interactions I have identified are:

- Fault Studies & PSA: These topics provide analytical input to the Fuel & Core assessment. This formal interaction has commenced during GDA Step 2.
- Rad-waste and Decommissioning: These topics provide input to the spent fuel safety aspects of the Fuel & Core assessment, in particular relating to longer term fuel storage proposals. This formal interaction has commenced during GDA Step 2.

3 REQUESTING PARTY'S SAFETY CASE

20. During Step 2 of GDA, GNS submitted a PSR and other supporting references, which together outline a Preliminary Nuclear Safety Case for the UK HPR1000. This section presents a summary of GNS's Preliminary Nuclear Safety Case in the area of Fuel & Core. It also identifies the documents submitted by GNS which have formed the basis of my Fuel & Core assessment of the UK HPR1000 during GDA Step 2.

3.1 Summary of the RP's Preliminary Safety Case in the Area of Fuel & Core

21. The aspects covered by the UK HPR1000 preliminary safety case in the area of Fuel & Core can be broadly grouped under the following headings:

- **Reactor Core**
  Main claim outlined in the preliminary safety case (Ref. 2, Chapter 5):
  "*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*."

  Specific claims are presented for different core features, for example:
  "- The core is designed so that the diametrical and azimuthal oscillations due to spatial xenon effects are self-damping and no operator action or control action is required to suppress them.
  - The stability to diametrical oscillations is so great that such phenomenon is deemed highly improbable.
  - Convergent azimuthal oscillations due to prohibited motion of individual control rods would be readily observable and alarmed.
  - Any axial xenon spatial power oscillations that occur in core life are controlled."

- **Fuel Assemblies (FA) and Fuel Rods (FR)**
  Main claim outlined in the preliminary safety case (Ref. 2, Chapter 5):
  "*The FA and fuel rods design ensures that the main safety functions will be supported in normal operation and under frequent faults*".

- **Fuel Handling and Storage System (FHSS)**
  Main claims outlined in the preliminary safety case (Ref. 2, Chapter 10):
  "- FHSS should maintain the handled or stored FA in a subcritical state.
  - FHSS should allow sufficient cooling of the irradiated fuel assemblies during handling and storage.
  - FHSS should maintain the fuel cladding integrity during fuel assemblies handling & storage".

- **Spent Fuel Interim Storage (SFIS)**
  Main claim outlined in the preliminary safety case (Ref. 2, Chapter 23):
  "SFAs from the UK HPR1000 will be managed safely until final disposal and the associated risk will be reduced to a level that is ALARP".

- **ALARP principles and methodology**
  Main claims outlined in the preliminary safety case (Refs. 10 and 11):
  "- The level of safety shall be no less than a comparable facility already operating or being constructed in the UK or elsewhere in the world.
  - The evolution of the design shall be shown to maintain or improve the design from a safety perspective taking into account the appropriate codes and standards, improved analyses and operating experience feedback."

22. Besides the safety claims, the preliminary safety case presents the base functional requirements and main design criteria for the Fuel & Core. The actual design is outlined in approximate terms by text, tables and drawings which are from FCG3 which is the reference design for UK HPR1000. The specific design of UK HPR1000 Fuel & Core as well as its supporting safety case – the Pre-Construction Safety Report (PCSR) will be submitted at the start of GDA Step 3.

23. Notably, the SFIS technology (wet/dry) and relevant equipment are not yet defined as the FCG3 project does not include such a facility. This is due to the specific Chinese arrangements for centralised spent fuel collection and reprocessing. The RP's decision on the subject and the conceptual framework for the SFIS of UK HPR1000, are expected in the early stage of GDA Step 3.

3.2 Basis of Assessment: RP's Documentation

24. The RP's documentation that has formed the basis for my GDA Step 2 assessment of the safety claims related to the Fuel & Core aspects of the UK HPR1000 is presented in in the following references:

- UK HPR1000 GDA Project Preliminary Safety Report (PSR)HPR/GDA/PSR/001 Chapters 4, 5, 10, 12, 13 & 23 (Ref. 2);

The PSR is the first major submission of the GDA Process. It has been developed and delivered to satisfy the requirements for Step 2 (Ref. 12) to provide confidence that UK safety standards could be met by the proposed reactor design and that the claimed principles and design criteria are likely to be achievable. The PSR outlines the reactor equipment and structures, the design and safety philosophy, the codes and standards applied in the design and the quality management systems applied by the designers. The PSR sets the fundamental GDA objective of the RP - to demonstrate 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"

- GNS GDA Project Scope for UK HPR1000 GDA Project (Ref. 9);

The main purpose of this document is to define the technical scope for the UK HPR1000 GDA in terms of specific features of the site, buildings, systems and components that need to be considered, as well as the appropriate level of design details and analyses for this stage of the project.

- GNS GDA Project – ALARP and BAT Principles and Requirements for UK HPR1000 GDA (Ref. 10);

This document presents the fundamental principles and processes applied by the RP to demonstrate that the risk from the UK HPR1000 is brought down to ALARP level by appropriate design, and that the Best Available Techniques (BAT) have been applied to protect the environment from any hazards related to the UK HPR1000 construction and operation.

- Generic Design Assessment for UK HPR1000 – ALARP Methodology (Ref. 11).

This document briefly presents the approach and methods applied by the RP to optimise the UK HPR1000 design so that the risk from operation of this power plant is reduced to ALARP level without compromising the plant economic efficiency.

25. The RP's response to my Regulatory Queries (Ref. 8) has provided clarifications and additional information that has facilitated my assessment.

26. In April 2018 GNS submitted to ONR, for information, an advance copy of the UK HPR1000 Pre-Construction Safety Report (PCSR) - in which Chapter 5 addresses the Fuel & Core design. Having early visibility of the PCSR scope and content has been useful in the planning and preparation of my GDA Step 3 assessment work.

4 ONR ASSESSMENT

27. This assessment has been carried out in accordance with HOW2 guide NS-PER-GD-014, “Purpose and Scope of Permissioning” (Ref. 3).

28. My Step 2 assessment work has involved continuous engagement with the RP’s Fuel & Core specialists. Four main technical meetings (Ref. 16) and a number of progress discussions have provided additional clarifications and information on RP’s work and plans for delivery of specific information to ONR.

29. During my GDA Step 2 assessment, I have identified some gaps in the documentation formally submitted to ONR. Consistent with ONR’s Guidance to Requesting Parties (Ref. 12), these normally lead to Regulatory Queries (RQs) being issued. At the time of writing my assessment report, I have raised six RQs to facilitate my assessment.

30. Details of my GDA Step 2 assessment of the UK HPR1000 preliminary safety case in the area of Fuel & Core, including the conclusions I have reached, are presented in the following sub-sections of this report. This includes the areas of strength I have identified, as well as the items that require follow-up during the next GDA Steps. The most serious difficulty with the assessment is that the PSR is based on FCG3, hence may be not completely representative of UK HPR1000. This has limited the extent to which I can draw conclusions from the material presented to date.

31. Throughout my assessment I have aimed to answer the following four questions:

    (1) Is the design presented to a sufficient level of detail?
    (2) Are the safety functions sufficiently defined and acceptable in principle?
    (3) Are the potential degradation mechanisms identified?
    (4) Are functional requirements and design criteria defined appropriately?

4.1 Reactor Core

32. The reactor core is composed of 177 FAs with fuel enrichment less than 5%, cooled and moderated by light water at a pressure of 15.5MPa. After a 12 to 18 months period of reactor operation (Fuel Cycle), the reactor is shut down. Then the fully irradiated (spent) FAs are replaced with fresh FAs, while the remaining (partly spent) FAs are re-located inside the core, so that the overall profile of energy generation and fuel use are optimised. Then the reactor is started up for the next fuel cycle.

33. Soluble boron in the primary coolant serves as a neutron absorber. The concentration of boron is varied to control reactivity changes that occur relatively slowly, including the effects of fuel burnup. Permitted design values are limited to maintain the desired negative reactivity coefficients.

34. Additional neutron absorber (gadolinium oxide), is added to the fuel in order to balance core reactivity and ensure even power distribution. Gadolinium is depleted (through neutron absorption) during the course of each fuel cycle, and is therefore known as a ‘burnable neutron absorber’ or ‘burnable absorber’.

35. The core reactivity and power distribution are also controlled by a number of Rod Cluster Control Assemblies (RCCA). Each RCCA consists of a group of individual neutron absorber rods (control rods) securely fastened at the top end to a common hub (spider). The RCCAs are moved vertically within special channels inside selected FAs to allow for reactor start-up/shutdown as well as for control of the overall core reactivity and power distribution. Fast shut down is provided by the de-energising the magnetic latch of the RCCA which releases the control rods to fall rapidly into the core.

4.1.1 Assessment

36. ONR SAP ERC.1 requires that "*The design and operation of the reactor should ensure the fundamental safety functions are delivered with an appropriate degree of confidence for permitted operating modes of the reactor."* I note that ensuring of the fundamental safety functions is claimed for each of the assessed aspects of Fuel & Core design presented in Ref. 2. This demonstrates compliance with the main purpose of GDA Step 2 – to assess the key claims and identify any fundamental safety or security shortfalls that could prevent ONR permitting the construction of a power station based on the design (Ref. 1).

37. My assessment of the RP's submissions on the subject has found that most of the information related to the reactor core and its components is presented in Chapter 5 of Ref. 2. Noting that Ref. 2 is based on the FCG3 design, I have focused my attention on the overall characteristics of the reactor core which are not expected to change significantly during the further stages of the UK HPR1000 project, such as number of fuel assemblies and fuel rods in each assembly, fuel enrichment limit, ways and means for reactivity control and for flattening of the core power profile. By applying such selective approach I have ensured that my assessment of FCG3 –based information will not undermine my judgement regarding the design of UK HPR1000.

Design presentation

38. My assessment of Chapter 5 (Ref. 2) has found that the principal (nuclear, thermal-hydraulic and mechanical) design parameters of the FCG3 reactor core are presented in a tabulated format.

39. I have not identified any design features that could comply with the definition of NS-TAST-GD-051 "The purpose, scope and contents of nuclear safety cases" for novel: "*any major system, structure or component not previously licensed in a nuclear facility anywhere in the world*". This is encouraging. However, since the final design has not been identified, I am not able to reach any firm conclusions.

40. I also note the statement in Ref. 2 that the UK HPR1000 core design will be presented in the GDA PCSR. Hence I expect the actual core design of UK HPR1000 to be submitted to ONR before the start of GDA Step 3.

41. Considering the above observations I judge that (Ref. 2) does not present the design of the reactor core and its components to a level of detail which is sufficient for GDA Step2. However, I have accepted an undertaking from the RP that this will be rectified to allow Step 3 to proceed in a meaningful way.

Definition of Safety Functions

42. My assessment of the core safety functions (reactivity control, heat removal and confinement of radioactive materials) has found that these are defined in compliance with the applicable ONR SAPs ERC.1 to ERC.3 (Ref. 4) and IAEA guidance (Ref. 6). For example, ERC 1 requires that "*The design and operation of the reactor should ensure the fundamental safety functions are delivered with an appropriate degree of confidence for permitted operating modes of the reactor*" and Chapter 5 of the PSR (Ref. 2) states: ''*The reactor core design (including nuclear core design, thermal hydraulic design, fuel rod and fuel assembly design) of the UK HPR1000 will set safety analysis bounding limits for the reactor core which together ensure that the fundamental safety functions are delivered during normal operation and following all Design Basis Condition events''*

43. I note that Ref. 9 describes two independent systems provided in the design to shut the reactor down. This demonstrates general compliance with the requirement of SAP ERC.2:"*At least two diverse systems should be provided for shutting down a civil reactor."* I find this sufficient for GDA Step 2. However the shutdown system efficiency in fault conditions remains to be substantiated by appropriate analyses which I plan to assess in Steps 3 and 4.

44. My assessment of chapter 5 of Ref. 2, has found reasonable claims and arguments for both the neutronic (xenon) stability and hydraulic stability of the core in normal operation. This indicates compliance with SAP ERC.3 "*The core should be stable in normal operation and should not undergo sudden changes of condition when operating parameters go outside their permitted range"*.

45. Regarding the core stability in terms of the moderator temperature coefficient of reactivity I have observed that according to Chapter 5 of Ref. 2 one of the main design requirements for is that: "*A negative moderator temperature coefficient is obtained for power operating conditions*". Chapter 5 of Ref.2 provides a brief discussion of the impact of fuel temperature and moderator temperature feedback on potential rapid increase of reactivity. It states: "*As a result, an increasing temperature leads to a more negative moderator temperature coefficient*." and "*The moderator temperature coefficient of reactivity is negative in the core from hot zero power to nominal power, which provides another slower negative feedback effect related to coolant temperature or void content*."

46. Considering the above observation I am content with the design provisions for overall core stability in FCG3 as a reference plant for UK HPR1000. However, in Step 3 and Step 4 I will assess in more detail the relevant design data, safety analyses and other supporting evidence to arrive at a distinctive conclusion on this subject.

47. My assessment of PSR Chapter 5 "Reactor Core" and Chapter 8 "Instrumentation & Control" has observed that the presented information on core monitoring is very brief and not sufficient to prove compliance with SAP ERC.4:"*The core should be designed so that parameters and conditions important to safety can be monitored in all operational and design basis fault conditions and appropriate recovery actions taken in the event of adverse conditions being detected*". However I note that the design includes three monitoring systems:

- Nuclear Instrumentation System based on neutron detectors which are installed outside the reactor pressure vessel and continuously monitor the changes of the reactor power level and power distribution.
- In-core instrumentation system – based on assemblies installed inside the reactor pressure vessel to measure reactor core neutron flux distribution, fuel assembly outlet coolant temperature, and pressure vessel water level.
- System for position indication and control of the RCCAs.

48. In my view the above information is acceptable at GDA Step 2 as it demonstrates appropriate core monitoring in the normal operation of FCG3. However in Step 3 I will examine the design provisions for core monitoring in UK HPR1000 under fault conditions and their capability to support adequate response of plant and operators.

Identification of Degradation Mechanisms

49. My assessment of Chapter 5 "Reactor Core" of the PSR has not found explicit identification of core degradation mechanisms. However I have found a description of the core thermal hydraulics and brief justification of the core resistance against hydraulic instability.

50. Considering that the risk of corrosion is covered by the chemical topic assessment, I see the hydraulic departure from nucleate boiling as a significant risk for core degradation. My assessment has found that the core pressure drop characteristics of FCG3 are claimed to have been determined by hydraulic tests (Ref. 2, Chapter 5 Section 5.5.4.2.1), which in my view is a good practice.

51. I am content with the brief presentation of the FCG3 core resistance to hydraulic departure from nucleate boiling and hydraulic instability. However I note that the substantiation of the actual UK HPR1000 core performance remains to be submitted by the RP and will be subject to my assessment in GDA Step 3.

Definition of Functional Requirements and Design Criteria

52. Having examined the presentation in Ref. 2 of the functional requirements (fuel integrity, shut-down margin and negative moderator temperature coefficient) and the relevant design basis criteria (fuel burnup, reactivity coefficients, control of power distribution, maximal controlled reactivity insertion rate, availability shutdown systems and stability of the core), I am satisfied that these aspects of the UKHPR1000 design are sufficiently defined for GDA Step 2, and acceptable in principle, noting that detailed evidence will need to be provided later in GDA to substantiate this.

53. My assessment has found that the core nuclear design is based on the cycle specific calculation that models reactor operation from start up to shut down and demonstrates that the core will meet the main requirements for safe performance:

- Fuel rod power is maintained in an optimised manner across the core to prevent DNB (hence damage of fuel cladding) in DBC-1 or DBC-2 events;
- There is sufficient shutdown margin to bring the core to a safe state under all operating modes and conditions; and
- The moderator temperature coefficient at power is negative.

54. In my view this approach is compliant with the applicable IAEA guidance (Ref. 6): "*The power distribution in the core and the fuel assemblies changes during the fuel cycle owing to the burnup of fuel. Accordingly, the excess reactivity of the core and the reactivity coefficients of the core also change. These phenomena should be taken into account in the design of the core and the fuel*".

55. Ref. 2 states that the computer codes applied in the development and justification of the core nuclear design of UK HPR1000 may be different from those used for FCG3 – depending on the fuel vendor selection which will be made before the start of GDA Step 3. This may cause some differences in the calculation results.

56. The principal subject of GDA Step 2 are the main claims and general design features, hence this situation is currently tolerable. Considering the statement in Ref. 2 that the selection of fuel vendor and consequently – of appropriate computer codes will be completed for Step 3, I intend to assess in that stage of GDA the computer codes, models, and data used for the core nuclear design of UK HPR1000 in terms of their verification and validation status, users' qualification, previous experience etc. I also expect the RP to submit supporting evidence (experimental results, benchmark calculations, etc.) for my assessment in Step 4.

57. I accept that the RP has defined the fundamental safety design criteria for the reactor core under normal operation and frequent design basis faults (control, cooling, containment) as well as for infrequent faults (diverse shut-down systems, adequate cooling and limitation of release) in line with the applicable ONR expectations, i.e.:

- No fuel failure for Frequent Fault sequences (P>1.E-3 1/y); and
- ALARP risk of fuel failure for Infrequent Faults (P<1.E-3).

58. I note however, that the RP intends to tolerate "small number of fuel rod cladding failures" in DBC-2 and "*fuel damage that precludes resumption of operation without considerable outage time*" in DBC-3 events. This is not in line with good practice in the UK. ONR expects that the design objective is to maintain fuel integrity in frequent faults (up to a return frequency of 1E-3/yr). Compliance with this criterion is in my view necessary to demonstrate the sound nature of the design and operating rules.

59. Another observation that resulted from my assessment of the PSR (Ref. 2) is that the definition of "Infrequent fault" presented there: "*a fault that may occur once during the lifetime of a fleet of operating plants*" is equal to ONR's definition of a "Frequent fault": "*event which may occur during the lifetime of a fleet of reactors*" (e.g. SAPs, para.727: "...frequent faults (i.e. those with an initiating fault frequency exceeding 1 x 10^{-3} pa …").

60. I will pursue the above anomalies early in Step 3 and raise a Regulatory Observation to address this shortfall if appropriate.

61. I note that the operational thermal limits applied to ensure delivery of the fundamental safety functions in the reactor core are defined as "*The power distribution is controlled so that the operational thermal limits (Maximum peak linear power density and departure from nucleate boiling ratio (DNBR)) are not exceeded either under DBC-1 and DBC-2 conditions*."
Note: DNBR is the ratio between the limiting heat flux at which departure from nuclear boiling occurs and the actual heat flux through the fuel cladding of the fuel rod with maximal linear power.

62. This approach aims to protect the fuel pellet (maximal linear power density) and fuel cladding (DNBR) from damage due to overheating. The use of a DNBR value for the lead fuel rod (set to ensure that the lead fuel rod has a low probability of DNB) is an established practice which ensures that (taking account of uncertainty) few if any pins will be damaged. While ONR regards this as necessary, ONR also expects a demonstration that the protection system set points are such that the expected (statistically defined) number of fuel rods experiencing DNB before a reactor trip is less than 1.0.

63. I note that the pellet-cladding interaction (PCI) is mentioned as a fuel degradation mechanism in the PSR (Ref. 2) but without discussion of the analyses needed to evaluate the PCI-related risk in normal operation and fault conditions. The means to protect the fuel against this failure mechanism needs to be defined and the potential impact on the reactor operating envelope – e.g. the rules limiting power increase rate. IAEA advise that protection measures should be based on empirical data from fuel ramp tests and this is ONR's initial expectation.

64. Design criteria relating to rapid reactivity faults are also inadequate. Many countries have established limits on the Radially-averaged Peak Fuel Enthalpy (RAPFE) and these have been subject to revision to take account of burnup effects. In previous GDA's ONR (Ref. 15) has acknowledged the international experimental evidence which shows that at burnup levels higher than 50 MWd/tU fuel cladding failures during RIA are more likely to be caused by pellet-cladding mechanical interaction than by high-temperature related to critical heat flux. The effects of burn-up appear to enhance the loading to the cladding and/or alter the failure mechanism and make the critical heat flux (as a stand-alone criterion) inappropriate. Hence I expect a fuel design criterion for rapid reactivity faults which takes all failure mechanisms into account to be defined.

Considering the average discharge burnup limit of 47 MWd/tU quoted in the draft PCSR, I am not concerned with the application of DNBR limit for RIA in GDA Step 2. However in Step 3 I will examine the operational burn-up limits proposed for UK HPR1000 and will pursue a detailed justification of fuel safety in RIA – in line with the internationally established good practices and available test data.

65. In my view the absence of detailed fuel analyses is tolerable for this stage of the GDA. Besides, in response to my RQ-UKHPR1000-0055 (Ref. 8) the RP has presented a list of challenges to the fuel design together with the relevant design criteria and design limits. I will assess these in GDA Steps 3 and 4.

66. Having considered the criteria and functional requirements outlined in Ref. 2 for the core design in order to ensure performance of the main safety functions of reactivity control, fuel cooling and prevention of radioactive release, I find these appropriate for this stage of the GDA. I expect more detailed definition of the design criteria and justification of the relevant operational limits in the in Step 3 and Step 4 submissions.

67. The above observations give me confidence that at the start of GDA Step 3 the RP will provide sufficient information to allow for a meaningful assessment of the core operating envelope and the relevant safety margins to the design limits.

4.1.2 Strengths

68. My assessment of the reactor core design has identified the following strengths:

- The description of the reference design provides sufficient level of detail for the current stage of the GDA.
- The safety claims are clearly defined and substantiated by fundamental safety design criteria and relevant functional requirements.
- The important degradation mechanisms are identified and considered by appropriate design provisions.

4.1.3 Items that Require Follow-up

69. During my GDA Step 2 assessment of “Reactor Core” I have identified the following additional potential shortfalls that I will follow-up during Step 3 of GDA:

- The computer codes applied in the core nuclear design of UK HPR1000 may be different from those used for FCG3 – depending on the fuel vendor selection which is due before the start of GDA Step 3. In Step 3 I intend to assess the computer codes, models, and data used for the UK HPR1000 in terms of their verification and validation status etc. I expect the RP to submit supporting evidence (experimental results, benchmark calculations, etc.) for my assessment in Step 4.
- The relevant good practice that protection will ensure the expected (statistically defined) number of fuel rod experiencing a sustained departure form nucleate boiling is less than one.
- The definition of a design criterion to protect against PCI failure in normal operation and frequent faults and the provision of suitable protection and operating rules to ensure that this is not breached.
- The definition of a design criterion for rapid reactivity insertion faults, which protects high-burnup fuel pins against cladding failure.

4.1.4 Conclusions

70. Based on the outcome of my Step 2 assessment of "Reactor Core", I have concluded that the main safety claims and design features of the reference core design of FCG 3 presented in the preliminary safety case for UK HPR1000 (Ref. 2) provide sufficient visibility and confidence in the design for this stage of GDA, but my assessment identified a number of significant shortfalls in this area, and these will be the subject of discussion during Step 3.

4.2 Fuel Assemblies and Fuel Rods

71. The RP is planning to carry out a formal fuel system procurement exercise in order to select an international manufacturer of high-quality fuel systems using relevant good practice, targeting zero fuel rod failures under DBC-1 and DBC-2 conditions in all cycles. The results of this activity are expected at the start of GDA Step 3.

72. Based on the FCG3 design the PSR outlines the following main design features:

73. Each fuel assembly (FA) consists of 264 fuel rods in a 17x17 square array. The fuel rods are located in the FA structure so that there is clearance between the fuel rod ends and the top and bottom FA nozzles to accommodate fuel rod expansion due to the high temperature and radiation in the core.

74. The fuel rod (fuel pin) consists of a fuel pellets stack, holding spring, top and bottom end plugs, and cladding which is the first barrier to the release of fission products. The fuel pellets are made from sintered uranium dioxide with or without burnable poison (gadolinium). The cladding is a tube made of a zirconium alloy that is closed with the top and bottom end plugs by welding. The fuel rods are installed in the FA and supported laterally by the grid assemblies.

4.2.1 Assessment

Design presentation

75. I note that the actual fuel characteristics of UK HPR1000 are currently not available and the relevant information in Ref. 9 is indicated as approximate – based on FCG3. As noted above, such situation is tolerable for GDA Step 2, but in my contacts with the RP (Ref. 16) I have explained that a meaningful assessment in GDA Step 3 needs to consider the actual fuel specification and have reached an agreement for timely provision of this information for my assessment in GDA Step 3.

76. The selection of fuel vendor and fuel specification is also important regarding the expenditure of ONR's resources in Step 3. Considering the significant experience of assessment of fuels already used in Sizewell B and proposed for other GDA projects, the depth and scope of assessment will depend on the level of difference between the UK HPR1000 fuel and those already familiar to ONR.

77. My review of Ref.9 Chapter 5 has observed that the reference (FCG3) FA and fuel rod designs have several aspects of similarity with those of other PWRs (e.g. the French 900 MW and Sizewell B). This gives me a reason to expect compliance of the UK HPR1000 design with the internationally established good practices.

78. I note the mixing vanes in the fuel assembly spacer grid design and the fuel assembly intermediate flow mixers which improve the coolant flow mixing – thus increasing the potential heat removal from the fuel by forced convection. On the other hand, mixing vanes and flow mixers are known to increase the FA flow resistance, which in turn requires application of higher hold-down force that tends to increase the risk of FA

bowing. In my view the absence of actual design details and justification of the mixing vanes and flow mixers is not a significant concern for GDA Step 2. However in GDA Step 3 I intend to consider these and other specific features of the FA in my detailed assessment of the actual UK HPR1000 fuel design.

Definition of Safety Functions

79. My review of Ref.9, Chapter 5, has found that the FA & fuel rod safety functions are outlined in the relevant design descriptions as follows:

- FA safety functions: appropriate positioning of the fuel elements inside the core and flow mixing by the grid vanes;
- Fuel rod safety functions: heat transfer from the fuel pellets to the coolant by the cladding, retention of the fission products inside the cladding and provision of even power distribution by the gadolinium doped pellets.

80. My assessment observed that the PSR (Ref. 2) does not indicate debris retention at the FA inlet. In discussion with the RP (Ref. 16-a) I was assured that such a device is included in the FA design and will be presented within the detailed fuel specification which is planned for submission at the start of GDA Step 3. I am content that some details of the FA design are not presented in the PSR which reflects FCG3. However, my Step 3 assessment will explore the UK HPR1000 fuel design to ensure that lower level safety functions like debris retention are appropriately provided and justified.

Identification of Degradation Mechanisms

81. My assessment of Ref. 9 has found a detailed list of the degradation mechanisms which challenge the FA and the Fuel Rod while in the reactor as well as during handling and storage. This demonstrates compliance with the relevant requirements of ONR's guide "NS-TAST-GD-075 Safety of nuclear fuel in power reactors" .I am therefore content with the RP's understanding of these mechanisms and their impact on the FA & fuel rod.

82. Considering the above observations I am content that in GDA Step 3 the UK HPR1000 design presentation and safety justification can be expected to provide sufficient information regarding the fuel degradation mechanisms and the relevant protective arrangements.

Definition of Functional Requirements and Design Criteria

83. My review of Ref.9, Ch.5, has not found specific functional requirements and design criteria for the FA and fuel rod. Considering the current stage of the GDA (PSR based on FCG3) I accept the RP's argument that "A set of nuclear safety analysis bounding limits will be set for the fuel rod performance analysis, which will be explained in future stages of GDA. These design limits will be set following a further review before a fuel and fuel assembly manufacturer is selected" (Ref.9, Chapter 5).

84. The RP's response to my RQ-UKHPR1000-0055 (Ref. 8) has indicated plans for delivery of detailed information on functional requirements, design criteria and safety analysis bounding limits for GDA Step 3. My discussions with the RP's specialists (Ref. 14) have provided me confidence in their understanding of this subject.

85. Considering the above observations I judge that the reference (FCG3) design of the FA and fuel rod is compatible with ONR's expectations for this stage of GDA and that the RP has established appropriate arrangements to produce the supporting analyses and data for a meaningful design assessment in Step 3.

4.2.2 Strengths

86. My assessment of the FA and Fuel Rod design has identified the following strengths:

- The main safety functions and degradation mechanisms are clearly identified.
- The RP has developed the initial plan for further work and main deliverables in GDA Steps 3 and 4.

4.2.3 Items that Require Follow-up

87. During my GDA Step 2 assessment of "FA and Fuel Rod" I have identified the following additional potential shortfalls that I will follow-up during Step 3 of GDA:

- Delivery of the actual fuel specification for UK HPR1000; and
- Analyses needed to evaluate the PCI-related risk in normal operation and accident conditions and the potential impact on the reactor operating envelope.

4.2.4 Conclusions

88. Based on the outcome of my Step 2 assessment of "Fuel Assembly and Fuel Rod", I have concluded that the main safety claims and design features presented in the preliminary safety case for UK HPR1000 (Ref. 2) – based on FCG3 - provide sufficient visibility of the FA & Fuel Rod design framework for this stage of GDA. I will carry out a detailed analysis of the actual FA & Fuel Rod design for UK HPR1000 in Step 3.

89. My assessment has not identified any fundamental safety shortfalls in this area.

4.3 Fuel Handling and Storage System

90. The Fuel Handling and Storage system (FHSS) is presented in Section 10.7 in Chapter 10 of Ref. 2. The system is located partly inside the reactor containment and partly in the fuel building and provides the following main functions:

- Admission, inspection and storage of the new assemblies;
- Fuel loading and unloading in reactor core;
- Transport of FAs between the buildings via the Fuel Transfer Tube (FTT);
- Post-irradiation FA storage and cooling in the Spent Fuel Pool (SFP) which is located inside the Fuel Building; and
- Loading of the spent fuel assemblies (SFA) into spent fuel casks for delivery to the spent fuel interim storage (SFIS) facility, which is located on site.

4.3.1 Assessment

Design presentation

91. My assessment of the PSR (Ref. 2, Chapter 23) has observed that the FHSS is arranged in a way which is common for most PWRs worldwide. The fuel route description explains the sequence and character of operations, and identifies the main equipment involved. Noting that this reflects FCG3 I find the presentation sufficiently detailed to conclude that the FHSS design is fit for the purpose of GDA Step 2.

92. I have also found that the PSR does not present a fuel route safety analysis. However I note that in response to RQ-UKHPR1000-0099 - Fault Studies - Safety Case for Fuel Route & Spent Fuel Storage (Ref. 8) – the RP commits to introduce in the PCSR analyses of the bounding fault events for the fuel handling and storage operations.

93. In GDA Step 3 I will assess these analyses to make sure that fuel safety is appropriately justified against correctly selected criteria.

Definition of Safety Functions

94. Ref. 2 defines the following safety functions of the FHSS:

- Reactivity control to ensure sub-critical conditions during fuel transport and storage;
- Removal of the residual (decay) heat generated in the fuel during fuel transport and storage; and
- Confinement of radioactive materials:
  - maintain the fuel cladding integrity
  - ensure containment isolation by the FTT valves during operation at power.

95. Having reviewed the presentation of FHSS safety functions in (Ref. 2) I agree that these are aligned to support the main safety functions: prevention of criticality, cooling of the fuel and isolation from the reactor containment during operation at power.

96. My assessment has not identified any significant gaps in the defined safety functions. However I note that the successful performance of these functions in normal and fault conditions remains to be justified by appropriate safety analyses.

Identification of Degradation Mechanisms

97. Two main groups of degradation mechanisms are identified within the FHSS design:

- Fuel: damage during transport;
- overheating/criticality/corrosion during storage in the SFP; and
- FHSS equipment: corrosion and ageing.

98. My assessment of Chapters 10 and 23 of the PSR (Ref.2) has outlined the following claims regarding the prevention of risks related to the identified degradation mechanisms:

99. Section 10.7.1.3 of Ref. 2 claims that the risk from fuel damage during transport is reduced by the FHSS design to minimize the risk of fuel assembly dropping or impact which might damage, even in case of earthquake or loss of electrical power. This claim is supported by the following design provisions:

- "*no heavy load can be handled above the underwater fuel storage area.*
- *the equipment used to transfer or lift fuel assemblies is provided with automatic braking or stopping devices to avoid inadvertent movement following loss of electrical power.*
- *both the spent fuel pool and underwater fuel storage racks are designed to maintain their integrity (including leak-tightness) when exposed to loads due to a safe shutdown earthquake (SSE).*
- *the design of the underwater fuel storage racks precludes:*
  - the placing of more than one fuel assembly in a single storage cell or putting or jamming an assembly between two storage cells.
  - geometry changes due to changes in ambient conditions or due to operating effects.
  - tipping or any other unplanned movement of the fuel or the racks."

100. The risks of fuel overheating and criticality of spent fuel is claimed to be prevented by the following design requirements to the SFP:

- *"The underwater fuel storage racks should be designed to allow free flow of the pool water so that assemblies are cooled and that criticality is precluded.*

- *The array pitches and the permanent neutron absorber shields should be such that Keff must not exceed 0.95"* (i.e. the risk of criticality inside the SFP must be prevented by the selection of materials and location of the SFA storage racks).

101. The risk of fuel corrosion is claimed to be prevented by the control of the chemical composition of SFP water.

102. The risk of new (fresh) fuel criticality is claimed to be prevented by the following design provisions (Ref. 2, Chapter 10):

- *"The centre-to-centre spacing of dry fuel storage racks is such that the K_{eff} does not exceed 0.98 for new fuel of the highest anticipated enrichment assuming the optimum physically envisaged moderation conditions"* (i.e. new FAs must be located at sufficiently large distance from each other so that criticality is prevented even under the hypothetical assumption that "*the fuel is submerged in pure water with normal density of 1.0g/cm^3*").
- *"During storage, new fuel assemblies are protected from falling objects, such as tools used during handling operations*."
- The design of dry fuel storage racks precludes:
  - the placement of more than one fuel assembly in a single storage cell or putting or jamming an assembly between two storage cells.
  - geometry changes due to changes in ambient conditions caused by operating effects.
  - tipping, and any unplanned movement of the fuel assemblies or the racks.
- *"Flammable products are not stored near new fuel assemblies, to avoid excessive fuel temperature in case of fire*."

103. The detailed assessment of risk from dropped load will require collaboration with the Internal Hazards and Mechanical Engineering disciplines – hence out of the Fuel & Core scope. My review of the above design provisions has not identified omission of any important risks. In Step 3 I expect the RP to demonstrate FHSS resistance to snags and other mechanical faults.

104. The criticality risk during storage of fresh or spent fuel is well known and I find the preventive measures applied in FCG3 appropriate considering the advice of SAP ECR.1, paragraph 571: "The principal means of passive engineering control of criticality should be geometrical constraint". However I note that the actual design and justification of these features of UK HPR1000 are currently missing and will only be able to form a judgement after their delivery and assessment in Step 3.

105. The risk of spent fuel overheating is correctly identified and the relevant arrangement of storage racks is reasonable. I have left the examination of further details in this area (e.g. cooling system capacity, fault conditions, etc.) for GDA Step 3 under the assumption that the actual design and safety justification for UK HPR1000 will be available for assessment.

106. The degradation of FHSS equipment is not generally within the scope of my assessment. However I note that the risk from corrosion of the FHSS equipment is considered in Ref. 2, Chapter 23, with regard to the selection of materials. I am also aware that the ONR Chemistry Inspector has initiated a discussion (Ref. 13) with the RP on the long term effectiveness of the solid (fixed) neutron absorbers located in the SFP. During GDA Steps 3 and 4 I will join the Chemistry Inspector to follow up this aspect of the SFP design.

107. Considering the above observations I find that the RP has identified the main degradation mechanisms of the fuel which is operated by the FHSS to a reasonable level of detail which is commensurate with the current stage of the GDA.

Definition of Functional Requirements and Design Criteria

108. The FHSS functional requirements comply with the overall NPP design principles: minimising the risk of FA drop during handling, protection against internal and external hazards, provision of biological shielding, requirements to the operations monitoring, equipment maintenance etc. Having examined the definitions of these requirements and considering SAPs EKP 1-5, I have found these appropriate for GDA Step 2.

109. I note however that the actual implementation of the functional requirements in the FHSS design remains to be demonstrated and justified in the next steps of GDA and will need to be assessed by the relevant ONR specialist inspectors in mechanical engineering, internal/external hazards, etc.

110. I have reviewed the design criteria suggested in Ref. 2 for the SFP, fresh fuel store and the fuel handling equipment and haven't found any significant gaps. Considering the key role of the SFP in the preparation of spent fuel for interim storage I have focused my attention on the criteria applied to the SFP design. My assessment resulted in the following observations and conclusions:

111. I have observed that the storage of failed fuel is not explicitly covered by the SFP design criteria. However, I note that Table T-10.7.1 identifies the availability of a sipping test facility for detection of failed irradiated fuel. Besides, according to section 23.7.5 of Ref.9, Chapter 23: "Any suspected failed IFAs (irradiated FAs) and SFAs (spent FAs) are transferred into dedicated failed fuel storage cells located in the SFP. In my assessment during GDA Steps 3 and 4 I will seek further details on the design provisions and safety justification of failed fuel handling and storage.

112. Considering the above observations I judge that the SFP design will allow for storage of failed fuel. However, I note that the functional requirement for SFP storage capacity needs to be presented and justified in more detail during the next steps of GDA.

113. I note that Chapter 10 of the PSR claims "*The spent fuel pool and the underwater fuel storage racks have sufficient capacity to contain all spent fuel assemblies produced by one reactor within at least 10 years, plus a full core in the event of a forced unloading*."

114. Based on my experience in similar projects for other PWRs I expect that 10 years cool-down in the SFP should provide for sufficient reduction of the spent fuel decay heat to facilitate safe interim storage. I am aware however, that a potential increase of the fuel burnup could increase significantly the spent fuel decay heat and consequently – the time needed for post-reactor cooling in the SFP. Therefore, my Step 3 assessment will examine the details of fuel and core design in terms of average and maximal burn up, as well as of decay heat profile to judge whether the SFP capacity is sufficient to provide appropriate level of cool down before the fuel is sentenced for interim storage.

115. I note that Ref. 2, Chapter 23, requires the FHSS lifting and handling equipment to be capable to stop and hold the transported fuel during a safe shutdown earthquake (SSE). I find this design provision useful for the prevention of load drop accidents. In collaboration with the mechanical engineering topic lead, I will examine the protection against snag events in Step 3.

116. In summary, I am content that the definition of functional requirements and design criteria presented in the PSR (Ref. 2) is sufficient for this stage of the GDA.

4.3.2 Strengths

117. My assessment of the FHSS design has outlined the following strengths:

- The design description is clear and sufficiently detailed for GDA Step 2.
- The design provisions against the identified degradation mechanisms are logically arranged and broadly comply with the engineering SAPs.

4.3.3 Items that Require Follow-up

118. During my GDA Step 2 assessment of the FHSS design, I have identified the following additional potential shortfalls that I will follow-up during Step 3 of GDA:

- Design provisions and safety justification for storage of failed fuel.

4.3.4 Conclusions

119. Based on the outcome of my Step 2 assessment of the FHSS design, I have concluded that the main safety claims and design features of the FCG 3 FHSS presented in the preliminary safety case for UK HPR1000 (Ref. 2) provide sufficient visibility and confidence in the design for this stage of GDA.

120. My assessment has not identified any fundamental safety shortfalls in this area.

4.4 Interim Spent Fuel Storage

121. China has specific arrangements of the spent fuel treatment - hence the reference plant FCG is not equipped with a SFIS facility. According to Chapter 23 of (Ref. 2) the SFIS operations planned for the UK HPR1000 will include transfer of the spent fuel assemblies from the SFP into transportation casks/canisters which are then moved to a SFIS building on site for storage prior to retrieval for final disposal off site.

122. The technological options considered for UK HPR1000 are presented in section 23.8.5: wet storage of the spent fuel assemblies (SFAs) and dry storage in separate modules or inside a vault.

123. According to Ref. 14, The RP has indicated it would complete the assessment of the two main technology options during Step 2, with a view to present the selected technology option in the PCSR during Step 3 and provide a preliminary assessment to demonstrate that relevant risks can be reduced to ALARP level.

4.4.1 Assessment

Design Information

124. I note that the absence of a SFIS design is a shortfall against ONR's expectations and could compromise the performance of a meaningful assessment during GDA Step 3. On the other hand, the RP's commitment to provide the design concept and preliminary safety justification provides assurance that the shortfall will be rectified. Hence I have come to opinion that the current situation is tolerable for GDA Step 2, but a Regulatory Observation should be raised if the Step 3 assessment is disturbed by late delivery or insufficient quality of the RP's submission on this subject.

125. My assessment in Step 3 will be focused on the adequacy of the design provisions and their safety justification, especially the interface by the fuel operation history (rated power, burnup, hydriding, decay heat), SFP storage, transfer to SFIS facility and storage in it, the options for spent fuel inspection and repackaging, as well as the post-SFIS preparation of the fuel for long term off-site storage.

Definition of Safety Functions

126. The SFIS safety functions are briefly described in the PSR (Ref. 2), as follows: “to safely prepare, handle, transfer, store, and retrieve the spent FAs.”

127. Considering that the actual SFIS technology has not yet been selected, I find this general definition tolerable for the current stage of GDA. However, it is obviously not sufficient for a meaningful assessment in Step 3 and I intend to follow up the timely delivery of the relevant information.

Identification of Degradation Mechanisms

128. The safe storage of spent fuel depends heavily on the fuel rods decay heat, levels of cladding oxidation/hydriding and on the internal gas pressure, which in turn is directly related to the fuel burnup. I note that cladding oxidation and hydriding are identified as fuel degradation mechanism in Ref. 9, Chapter 5. However, the specific degradation mechanisms related to SFIS are not identified in Chapter 23 (e.g. hydride reorientation in the cladding at certain levels of temperature and stress).

129. As noted above, this situation is tolerable for GDA Step 2. However, considering the inter-connection of safety limits for reactor operation, SFP cooling and SFIS, my assessment in Step 3 will include a detailed examination of the definition of degradation mechanisms, establishment of relevant limits and introduction of preventive measures which are supported by adequate analyses.

Definition of Functional Requirements and Design Criteria

130. Section 23.8.4 of Ref. 2 Chapter 23 outlines a number of functional requirements for the SFIS design, for example:

- *“100 years operation lifetime of the SFIS facility;*
- *Structural integrity of the SFAs during long term interim storage lifetime (i.e. storage duration of the SFAs in the SFIS building);*
- *Inspection requirements for the SFAs during the SFIS operations (including long term interim storage) to verify the integrity of the SFAs;*
- *Potential use of the SFIS facility to accommodate other High Level Waste;;*
- *Potential for the use of the SFIS facility for multiple reactor units; and*
- *Retrievability of the SFAs prior to and after the Reactor Building and Fuel Building have been decommissioned”.*

131. In my view the above functional requirements are described to a sufficient level of detail for this stage of the SFIS project. My assessment has not observed the absence of any important functional requirement that could compromise the safety of the facility.

4.4.2 Strengths

132. My assessment of the SFIS presentation has outlined the following strengths:

- The main functional requirements are clearly defined and sufficiently detailed for this stage of the project.

4.4.3 Items that Require Follow-up

133. During my GDA Step 2 assessment of the SFIS presentation I have identified the following additional potential shortfalls that I will follow-up during Step 3 of GDA:

- The available information is not sufficient for a meaningful assessment in Step 3 and I intend to follow up the timely delivery of the relevant design documents and safety analyses.

4.4.4 Conclusions

134. Based on the outcome of my Step 2 assessment of the SFIS presentation, I have concluded that the main safety claims and functional requirements outlined in the preliminary safety case for UK HPR1000 (Ref. 2) provide a tolerable level of visibility for this stage of GDA.

135. My assessment has not identified any fundamental safety shortfalls in this area.

4.5 Categorisation of Safety Functions and Classification of Systems, Structures and Components

136. The RP's approach to categorisation of safety functions and classification of equipment in FCG3 is presented in Chapter 4 of the PSR (Ref. 2). It is based on the IAEA guidance which reflects the internationally established good practices.

4.5.1 Assessment

137. My assessment of PSR Chapter 5 Reactor core has not found discussion of the categorisation of safety functions and classification of equipment.

138. I note that Section 4.7.1 states: "*A gap analysis is being undertaken between the approach used for HPR1000 (FCG3) and those recognized as relevant good practice in UK. The approach to be applied for UK HPR1000 will be defined after the gap analysis completes and will be described in UK HPR1000 Pre-Construction Safety Report (PCSR). It will be mainly based on the approach described in this sub-chapter and the gaps will be addressed properly.*"

139. Considering the above statement, I intend to assess the presentation on this subject within the PCSR which should be submitted at the beginning of GDA Step 3.

4.5.2 Strengths

140. My assessment of the Categorisation of Safety Functions and Classification of Systems, Structures and Components has outlined the following strengths:

- The RP has committed to carry out a gap analysis and align their categorisation and classification approach for UK HPR1000 with those recognized as relevant good practice in UK.

4.5.3 Items that Require Follow-up

141. During my GDA Step 2 assessment of the Categorisation of Safety Functions and Classification of Systems, Structures and Components I have identified the following additional potential shortfalls that I will follow-up during Step 3 of GDA:

- The actual categorisation and classification for UK HPR-1000 will be presented in the PCSR which I intend to assess during the Step 3 GDA – including the safety classification of the core monitoring system and the safety category of the functions it performs as a specific point of interest for the Fuel & Core topic.

4.5.4 Conclusions

142. Based on the outcome of my Step 2 assessment of the Categorisation of Safety Functions and Classification of Systems, Structures and Components, I have concluded that the approach outlined in the preliminary safety case for UK HPR1000 (Ref. 2) provides sufficient visibility for this stage of GDA.

143. My assessment has not identified any fundamental safety shortfalls in this area.

4.6 ALARP Considerations

144. Ref. 11 presents a brief review of ALARP legal base and interpretation in terms of nuclear safety and claims that consideration have been taken of ONR's approach - by reference to NS TAST-GD-005 "Guidance on the demonstration of ALARP" (Ref. 5).

145. The actual ALARP justification for UK HPR1000 should be presented in the PCSR and is claimed to demonstrate that:

- The level of safety shall be no less than a comparable facility already operating or being constructed in the UK or elsewhere in the world.
- The evolution of the design shall be shown to maintain or improve the design from a safety perspective taking into account the appropriate codes and standards, improved analyses and operating experience feedback.
- The relevant Key Engineering Principles in the ONR SAPs shall be addressed.

146. The following main steps for preparing the PCSR ALARP analysis are specified:

- Demonstration of compliance with relevant good practices for controlling risk;
- Examination of options for risk reduction;
- Assessment of the risk reduction achieved by the selected option; and
- Demonstration that all practicable measures have been adopted

147. Ref. 12 presents the flowchart and details of the ALARP justification process that will be applied by the RP in the PCSR preparation. The process aims to outline the safety improvements and risk reduction achieved during the evolution from the Chinese Pressurized Reactor (CPR1000) to the UK HPR1000 design.

4.6.1 Assessment

148. I note that the RP has adopted ONR's approach to ALARP (Ref. 11) and has established a logically structured process (Ref. 11) for risk assessment and justification of ALARP in all aspects of the UK HPR1000 design.

149. I am satisfied that the ALARP analysis begins after the design has demonstrated compliance with the relevant good practices for controlling risk and that where shortfalls against risk targets are identified an optioneering process is applied to identify reasonable mitigation measures.

150. In my view the three steps that follow (risk reduction optioneering, assessment of the selected option and demonstration that all practicable measures have been applied) are logical and fit within ONR's expectations for a robust ALARP justification – as reflected in the relevant TAG-005 "Guidance on the demonstration of ALARP" (Ref. 5).

151. Based on the above observations I judge that the framework for ALARP analysis set by the RP is appropriate. However, the effective implementation of this framework and the produced results remain subject of ONR’s assessment in GDA Step 3 and Step 4.

152. Considering that the actual design and safety analyses for the UK HPR1000 fuel, fuel handling and storage are not yet available, I have left the specifics of ALARP justification out of the scope of my the Fuel & Core GDA Step 2 assessment.

153. In GDA Step 3 I will focus my assessment on the UK HPR1000 Fuel & Core design and safety justification to judge whether the potential hazards are correctly identified and the selected design options have reduced the overall risk to a level that is ALARP.

4.6.2 Strengths

154. My assessment of the ALARP considerations has outlined the following strengths:

- The RP has adopted ONR’s approach to ALARP and has established a logically structured process for risk assessment and justification of ALARP in all aspects of the UK HPR1000 design.

4.6.3 Items that Require Follow-up

155. During my GDA Step 2 assessment of the ALARP considerations I have identified the following additional topics for consideration or potential shortfalls that I will follow-up during Step 3 of GDA:

- The actual design and safety analyses for the UK HPR1000 fuel, fuel handling and storage are not yet available.

4.6.4 Conclusions

156. Based on the outcome of my Step 2 assessment of the ALARP considerations, I have concluded that the process outlined in the preliminary safety case for UK HPR1000 (Ref. 2) provides sufficient visibility for this stage of GDA.

157. My assessment has not identified any fundamental safety shortfalls in this area.

4.7 Out of Scope Items

158. The following activities outlined in my GDA Step 2 Assessment Plan (Ref. 1), have been left outside the scope of my GDA Step 2 assessment of the UK HPR1000 Fuel & Core.

- Further familiarization with the UK HPR1000 design to provide a basis for planning subsequent, more detailed, assessment during Steps 3 and 4 of GDA.
- Commissioning and management of the work of Technical Support Contractors (TSCs), if / as required.

159. These activities were not carried out due to the unavailability of the actual UK HPR1000 design of fuel, core, fuel handling and fuel storage facilities – including descriptions, drawings, specifications, applied computer codes, etc. I have found the FCG3 information submitted by the RP on these subjects incomplete as it only represented the reference plant that is currently built in China and is referred to in the PSR as “approximate values”.

160. My contacts with the RP have provided a reasonable level of confidence the necessary information is currently under preparation and will be provided at the beginning or

during the early stages of GDA Step 3. I note however, the actual scope and timing of relevant submissions is still subject of discussion.

161. It should be noted that the above omissions do not invalidate the conclusions from my GDA Step 2 assessment. I will capture the follow-up of the above out-of-scope items in my GDA Step 3 assessment plan and will pursue their assessment in Step 3.

4.8 Comparison with Standards, Guidance and Relevant Good Practice

162. In Section 2.2, above, I have listed the standards and criteria I have used during my GDA Step 2 assessment of the UK UKHPR1000 Fuel & Core, to judge the adequacy of the preliminary safety case. In this regard, my overall conclusions can be summarised as follows:

- SAPs: a reasonable level of compliance with the applicable SAPs has been observed. Table 1 provides further details.
- TAGs: The expectations of ONR's TAGs: NS TAST-GD-005 Guidance on the demonstration of ALARP, NS-TAST-GD-075 Safety of nuclear fuel in power reactors and NS-TAST-GD-081 Safety aspects specific to storage of spent nuclear fuel have been satisfied to a level commensurate with the current stage of the GDA process.
- My assessment has not identified points of conflict with the applicable IAEA and WENRA guidance.

163. My assessment of the PSR Chapter 5 Reactor core has not identified any reference to the standards applied in the FCG3 Fuel & Core design. However I note that the RP's response to RQ-UKHPR1000-0055 (Ref. 8) refers to ASME codes in the definition of design criteria for fuel stress (ASME BPVC III Rules for Construction of Nuclear Facility Components) which is acknowledged as an example of good practice.

164. During my assessment in Step 3 I will follow up the delivery of the list of codes and standards applied by the RP in the Fuel & Core design of UK HPR10000.

4.9 Interactions with Other Regulators

165. I have not participated in interactions with other regulators in relation to UK HPR1000.

5 CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions

166. During Step 2 of GDA GNS submitted a PSR and other supporting references, which outline a preliminary nuclear safety case for the UK HPR1000. These documents have been assessed by ONR. The PSR together with its supporting references present at a high level the claims in the area of Fuel & Core that underpin the safety of the UK HPR1000.

167. During Step 2 of GDA I have targeted my assessment at the content of the PSR and its references that is of most relevance to the area of Fuel & Core; against the expectations of ONR's SAPs and TAGs and other guidance which ONR regards as Relevant Good Practice. From the UK HPR1000 assessment done so far, I conclude the following:

- The design criteria do not meet ONR's requirements to maintain fuel integrity in Frequent Faults and demonstration of protections against PCI and PCMI is needed.
- The design of the fuel has not yet been finalised and therefore it is not possible to plan my assessment for Step 3 in detail at this stage.
- The models, computer codes, uncertainty and safety margins of the Fuel & core analysis have not been presented.
- The Fuel & Core design information presented by the RP reflects the reference plant FCG3 to a level of detail which is broadly acceptable for GDA Step 2 and the claims made by the RP in the areas of core design, fuel assembly and fuel rod, fuel handling and storage are reasonable.
- While I have identified some gaps in the information provided, I note that these are recognised and the RP has committed to address them for GDA Step 3.
- The unavailability of the actual UK HPR1000 design of fuel, core, fuel handling and fuel storage facilities has not allowed for complete implementation of my Step 2 assessment plan. The provision of appropriate data in these areas is critical for a meaningful assessment in GDA Step 3.
- I have achieved sufficient level of familiarity with the Fuel & Core design of the reference plant (FCG3). I expect to develop this further in GDA Step 3 based on the specific data for UK HPR1000.

168. Overall, during my GDA Step 2 assessment, I have not identified any fundamental safety shortfalls in the area of Fuel & Core that might prevent the issue of a Design Acceptance Confirmation (DAC) for the UK HPR1000 design.

5.2 Recommendations

169. My recommendations are as follows.

- Recommendation 1: ONR should consider the findings of my assessment in deciding whether to proceed to Step 3 of GDA for the UK HPR1000.
- Recommendation 2: All the items identified in Step 2 as important to be followed up should be included in ONR's GDA Step 3 Fuel & Core Assessment Plan for the UK HPR1000.
- Recommendation 3: All the relevant out-of-scope items identified in sub-section 4.7 of this report should be included in ONR's GDA Step 3 Fuel & Core Assessment Plan for the UK HPR1000.

6 REFERENCES

1. Generic Design Assessment of GNS's UK HPR1000 - Step 2 Assessment Plan for Fuel & Core ONR-GDA-AP-XX-009 Revision 0.
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   Chapter 4 General Safety and Design Principles, TRIM: 2017/401351
   Chapter 5 Reactor Core, TRIM: 2017/401353
   Chapter 10 Auxiliary Systems, TRIM: 2017/401360
   Chapter 12 DB Conditions, TRIM: 2017/401363
   Chapter 13 DE&SA Conditions, TRIM: 2017/401365
   Chapter 23 Rad Waste Management & Fuel Storage, TRIM 2017/401394

3. ONR HOW2 Guide NS-PER-GD-014 Revision 4 - Purpose and Scope of Permissioning. July 2014. http://www.onr.org.uk/operational/assessment/index.htm

4. Safety Assessment Principles for Nuclear Facilities. 2014 Edition Revision 0. November 2014. http://www.onr.org.uk/saps/saps2014.pdf

5. Technical Assessment Guides
   http://www.onr.org.uk/operational/tech_asst_guides/index.htm

   NS TAST-GD-005 Guidance on the demonstration of ALARP
   NS-TAST-GD-051 The purpose, scope and contents of nuclear safety cases
   NS-TAST-GD-075 Safety of nuclear fuel in power reactors

6. IAEA guidance – www.pub.iaea.org/MTCD/publications/pdf
   Safety Guide: Design of the Reactor Core for Nuclear Power Plants No. NS-G-1.12, Vienna, 2005

7. Western European Nuclear Regulators' Association: Reactor Reference Safety Levels, WENRA, September 2014, www.wenra.org

8. UK HPR1000 Regulatory Query Tracking Sheet, 2 November 2017 - Reference for Step 2 Assessment Reports, TRIM 2018/315144

9. GNS GDA Project Scope for UK HPR1000 GDA Project,
   29/05/2018, TRIM: 2018/179809

10. GNS GDA Project – ALARP and BAT Principles and Requirements for UK HPR1000 GDA, 29/05/2018, TRIM: 2018/181393

11. Generic Design Assessment (GDA) for UK HPR1000 – ALARP Methodology.
    TRIM: 28/04/2018, TRIM: 2018/181415

12. ONR-GDA-GD-001 - New Nuclear Reactors: Generic Design Assessment Guidance to Requesting Parties, Revision 3, September 2016,
    http://www.onr.org.uk/new-reactors/guidance-assessment.htm

13. ONR-GDA-UKHPR1000-AR-18-015, GDA Step 2 Assessment of Chemistry of the UK HPR1000 Reactor, Sept. 2018, TRIM: 2018/244343

14. ONR-GDA UKHPR1000-AR-18-016, GDA Step 2 Assessment of Radioactive Waste Management, Decommissioning and Spent Fuel Management of the UK HPR1000 Reactor, Sept. 2018, TRIM: 2018/244383

15. ONR-NR-AR-17-019, GDA, Step 4 Fuel & Core Design Assessment of the UK ABWR, Nov. 2017, TRIM: 2016/492101.

16. Contact records from L4 Technical meetings TRIM: 4.5.9490

a) Meeting date: 21 December 2017, ONR-NR-CR-18-611, TRIM: 2018/5574
b) Meeting date: 09 February 2018, ONR-NR-CR-17-690, TRIM: 2018/53011
c) Meeting date: 23 March 2018, ONR-NR-CR-17-779, TRIM: 2018/105519
d) Meeting date: 26&27 April 2018, ONR-NR-CR-18-090, TRIM: 2018/15404
e) Meeting date: 18 – 20 April 2018, ONR-NR-CR-18-078, TRIM: 2018/157946

Table 1

Relevant Safety Assessment Principles considered during the Assessment

SAP No and Title	Description	Interpretation	Comment
ECR.1 Engineering principle: criticality safety paragraph 571	Safety measures The principal means of passive engineering control of criticality should be geometrical constraint.	Design measures to prevent criticality In fuel storage	Addressed in Section 4.3.1 of this report.
ERC.1 Design and operation of reactors	The design and operation of the reactor should ensure the fundamental safety functions are delivered with an appropriate degree of confidence for permitted operating modes of the reactor	I note that ensuring of the fundamental safety functions is claimed for each of the assessed aspects of fuel & core design presented in (Ref. 9).	Addressed in Section 4.1.1 of this report.
ERC.2 Shutdown systems	At least two diverse systems should be provided for shutting down a civil reactor.	I note that Ref.9 describes two independent systems provided in the design to shut the reactor down.	Addressed in Section 4.1.1 of this report. The adequacy of these systems remains to be substantiated by appropriate analyses in Steps 3 and 4.
ERC.3 Stability in normal operation	The core should be stable in normal operation and should not undergo sudden changes of condition when operating parameters go outside their permitted range.	My review of (Ref. 9, Chapter 5) has found reasonable claims and arguments for both the neutronic (xenon) stability and hydraulic stability of the core in normal operation.	Addressed in Section 4.1.1 of this report.
ERC.4 Monitoring of parameters important to safety	The core should be designed so that parameters and conditions important to safety can be monitored in all operational and design basis fault conditions and appropriate recovery actions taken in the event of adverse conditions being detected.	My assessment of PSR Chapter 5 “Reactor Core” and Chapter 8 “Instrumentation & Control” has observed that the presented information on core monitoring is very brief and not sufficient to prove compliance with SAP ERC.4:”The core should be designed so that parameters and conditions important to safety can be monitored in all operational and design basis fault conditions and appropriate recovery actions taken in the event of adverse conditions being detected”.	Addressed in Section 4.1.1 of this report.

SAP No and Title	Description	Interpretation	Comment
EKP.1 Inherent Safety	The underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility.	The FHSS and SFIS functional requirements outline compliance with the overall NPP design principles: minimising the risk of FA drop during handling, protection against internal and external hazards, provision of biological shielding, requirements to the operations monitoring, equipment maintenance etc. Having checked the definitions of these requirements against SAPS EKP.1-5 I have observed sufficient level of compliance for this stage of the project.	Addressed in Section 4.1.3 of this report. I note that the implementation of these functional requirements remains to be demonstrated and justified in the next steps of GDA and will need to be assessed by the relevant ONR specialist inspectors in mechanical engineering, internal/external hazards, etc.
EKP.2 Fault tolerance	The sensitivity of the facility to potential faults should be minimised.		
EKP.3 Defence in depth	Nuclear facilities should be designed and operated so that defence in depth against potentially significant faults or failures is achieved by the provision of multiple independent barriers to fault progression.		
EKP.4 Safety function	The safety function(s) to be delivered within the facility should be identified by a structured analysis.		
EKP.5 Safety measures	Safety measures should be identified to deliver the required safety function(s).		