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NUCLEAR REACTOR TECHNOLOGY ASSESSMENT FOR NEAR TERM DEPLOYMENT

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

IAEA NUCLEAR ENERGY SERIES No. NR-T-1.10 (Rev. 1)

NUCLEAR REACTOR TECHNOLOGY ASSESSMENT FOR NEAR TERM DEPLOYMENT

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FOREWORD

The IAEA's statutory role is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world". Among other functions, the IAEA is authorized to "foster the exchange of scientific and technical information on peaceful uses of atomic energy". One way this is achieved is through a range of technical publications including the IAEA Nuclear Energy Series.

The IAEA Nuclear Energy Series comprises publications designed to further the use of nuclear technologies in support of sustainable development, to advance nuclear science and technology, catalyse innovation and build capacity to support the existing and expanded use of nuclear power and nuclear science applications. The publications include information covering all policy, technological and management aspects of the definition and implementation of activities involving the peaceful use of nuclear technology.

The IAEA safety standards establish fundamental principles, requirements and recommendations to ensure nuclear safety and serve as a global reference for protecting people and the environment from harmful effects of ionizing radiation.

When IAEA Nuclear Energy Series publications address safety, it is ensured that the IAEA safety standards are referred to as the current boundary conditions for the application of nuclear technology.

As IAEA Member States embark on initiatives to establish or reinvigorate their nuclear power programmes, the IAEA publishes information on identifying the complex tasks and processes associated with such an undertaking. A major challenge — especially for embarking Member States — is the process of reactor technology assessment (RTA) for near term deployment. An RTA includes the entire selection process for the most suitable reactor technology to meet the objectives of a Member State's nuclear power programme. Documenting and justifying this RTA decision making requires detailed knowledge of reactor technology and best practices.

Several aspects of the infrastructure development programmes that support nuclear power development in embarking Member States interact directly with an RTA. The initial stage of an RTA requires the principal objectives to be defined by Member State decision makers. The RTA then develops and delivers the necessary technical evaluation for a project feasibility study, bid invitation, evaluation and contracting, and reactor deployment phases. The same approach may also be applied to Member States seeking to expand their existing nuclear power programmes.

This publication explains how an RTA is carried out and how the process and results enable decision making for planning and implementing nuclear power in each phase of an infrastructure development programme. The RTA methodology provides decision makers with the documentation to support their conclusions. The preparation for and application of the RTA methodology as described in this publication create a vehicle for capacity building in Member States through technology training provided by the IAEA.

The RTA methodology has been revised to incorporate developments since its first publication in 2013 and includes feedback from comprehensive training workshops offered to Member States introducing or expanding on their nuclear power programmes. This publication incorporates and harmonizes these new developments and experiences into a refined RTA methodology.

Reactor technology assessment is a continuous and iterative process, with ever increasing requirements for the level of detail to support the decision making. To enable the sound identification and selection of reactor technologies, the RTA methodology is significantly more than a review of technology design attributes. In this respect, the availability of objective technical information, databases and tools to perform a detailed comparative assessment of different reactor technologies and types requires consistent and technical information to support RTA training and its application.

The aim of this publication is to help embarking Member States understand the complexity involved in the selection of the most suitable reactor technology and the obligations associated with and responsibilities of an unbiased assessment. This publication can also be used by Member States that already have nuclear power programmes developed to assist in their selection of a potential nuclear power plant.

The IAEA officers responsible for this publication were T. Jevremovic and M. Krause of the Division of Nuclear Power.

CONTENTS

SUMMARY

The development or expansion of a nuclear power programme is a major undertaking requiring careful planning, preparation, institutions and human resources. The IAEA's Milestones approach identifies 19 infrastructure issues and three progressive phases, for a successful initiation and development of a national nuclear power programme. Within the framework of the Milestones approach, the nuclear reactor technology assessment RTA is a decision making methodology, based on numerous technical considerations of nuclear technology translated into key elements (KEs) with subsets of key topics (KTs). The whole process helps Member States to evaluate and assess KEs and KTs quantitatively as part of the feasibility studies, and eventually select the most suitable reactor technology that is consistent with national requirements, needs and objectives.

This publication is intended to provide guidance on the use of the RTA process and to establish requirements and criteria for its objective application when starting or expanding a nuclear power programme. It is based on the experience and good practices in Member States, as well as on lessons learned and feedback from numerous RTA training workshops and integrated nuclear infrastructure review missions conducted between 2012 and 2019. The main objectives of this revision are to:

— Incorporate recent developments in the nuclear power landscape, by introducing elements relevant to small and medium sized or modular reactors (SMRs), non-electric applications and tightly coupled nuclear–renewable energy systems;
— Introduce the role of the RTA within the IAEA Milestones approach;
— Reduce the overlap of KTs between different KEs, by consolidating KEs and reducing the number of the RTA KEs;
— Clarify the meaning and scope of the various KEs and KTs.

This publication:

— Identifies the need to establish, or develop, clear and prioritized policy objectives for the national nuclear power programme as the prerequisite to initiate the RTA;
— Establishes RTA as the decision making methodology for the evaluation and documentation, by a competent RTA team, of nuclear technologies for deployment;
— Describes RTA for near term deployment within the context of the adjoining major tasks of the Milestones approach, delivering the essential technical evaluations for the project's pre-feasibility study at the programme level (Phase 1), the feasibility study for a specific NPP project (Phase 2), the bid invitation and/or evaluation (Phase 3) and, finally, contracting the reactor deployment;
— Focuses on the process for performing RTA by gathering and refining expert opinion to identify the most important features and components for the evaluation;
— Demonstrates, with comprehensive examples, the application of decision making processes to perform RTA in a manner that integrates with the IAEA technical approach for the evaluation of bids for the NPP;
— Provides comprehensive RTA tables for RTA KEs and KTs and guidance on their use.

1. INTRODUCTION

1.1. BACKGROUND

The IAEA developed the RTA methodology as a relatively comprehensive guide in advising its Member States in the process of identifying, evaluating and selecting available technology options. The methodology includes but is not limited to large water cooled reactors (WCRs), SMRs and their broad applications (electricity production, non-electric applications, hybrid energy systems) both among newcomer countries and in those countries with expanding nuclear power programmes.

Reactor technology assessment contributes to the evaluation, selection and deployment of the most suitable technology, generally deployable in the near term, and specifically in time for the planned implementation of the nuclear power project, to meet the objectives of a national nuclear power programme. The goals and objectives that describe the rationale for initiating a nuclear power programme or a particular nuclear project need to be specified and understood at the outset. Only then can technical elements be linked to policy objectives. This assures that the technical and economic comparison of the candidate NPP designs and associated technologies will be assessed with objective intention against the conditions, constraints and needs of the country, so that the most suitable design for electricity generation and/or other applications can be selected.

The RTA methodology, when applied objectively and consistently throughout the development of a national nuclear power programme, enables the decision makers to eventually choose the NPP type that will best fulfil their national policy objectives, which may also include a set of utility requirements and criteria. This process is established with respect to the other major elements of nuclear power programme development. There are several applications where Member States will perform and apply the RTA process and, although the detail of the evaluation and the scope of the selection will vary between phases, there is a common approach and issues raised at each phase are carried through to the next phase. At each subsequent stage of the IAEA Milestones approach, the detail of the RTA would increase, with the initial stage at the pre-feasibility study (PFS) in Phase 1 being used to determine what technologies would be feasible, and later stages with more detailed studies on specific infrastructure issues available, allowing differentiation between specific design options. More precisely, the IAEA RTA methodology can be used as per the three phases in the IAEA's Milestones approach [1]:

- During the PFS, Phase 1;
- During the feasibility study (FS), Phase 2;
- In preparation of the bid invitation specifications (BIS) in Phase 2 and to support the evaluation of bids in Phase 3;
- As a decision making tool in preparation for contract negotiations (Phase 3).

The need to provide Member States with an RTA methodology for near term deployment was initially suggested by the IAEA Technical Working Group on Water Cooled Reactors. As the first follow-up activities, the IAEA organized technology assessment workshops in 2007, 2008 and 2011 to identify and discuss approaches and results developed from the current practice of technology assessment. As feedback from these meetings, the Member States emphasized their desire for the IAEA to capture and formalize, through a specific document, an RTA methodology for their use based upon their particular needs. This resulted in the IAEA Nuclear Energy Series No. NP-T-1.10, Reactor Technology Assessment for Near Term Deployment, published in 2013. The RTA methodology was based primarily on the accumulated experience and expertise for large WCRs and examples and suggestions of how to apply it. Since the 2013 publication, more frequent and comprehensive RTA training workshops were conducted that generated practical lessons learned that are incorporated in this revision. In 2019 and 2020 a series of two consultancy meetings were conducted with experts from various Member States in finalizing this revision.

Major developments in the nuclear industry worldwide include new builds of large commercial NPPs in newcomer countries, offerings of new innovative reactor designs, reactor technology transfer from established to embarking NPP technology holders, life extensions and midlife refurbishment. Furthermore, there is increased interest in SMRs, non-electric applications, and tightly coupled nuclear–renewable energy systems. The content of this revision incorporates and harmonizes these developments and experiences into a refined RTA methodology.

1.2. OBJECTIVE

The original publication and this revision provide Member States with current guidance to support informed decision making when choosing among various available reactor designs to determine the NPP technology that best meets the national needs. The RTA methodology described in this revision provides a technology neutral systematic approach to evaluate the technical merits of the various NPP technologies already available on the market or expected to be commercialized in the near future. The evaluations are based on each user's objectives, requirements and criteria. In addition, this revision clarifies the variety of definitions and applications of RTA in the IAEA training workshops, as well as in the national use. It provides both basic and detailed descriptions of the way in which RTA methodology is planned, executed, evaluated, documented and reported.

The objective is to provide a clear understanding of the RTA methodology, demonstrate the basic elements of RTA methodology and its steps, and provide enough information and guidance to build and conduct RTA in practice.

1.3. STRUCTURE

Section 2 describes when and why an RTA is performed in the larger framework of developing strategies and making decisions when implementing a national nuclear power programme. Section 3 outlines the main steps to be taken prior to starting with the RTA methodology, which is described in detail in Section 4. Section 5 explains the ten key elements and corresponding key topics, altogether defining the IAEA RTA methodology. Section 6 provides a brief conclusion on the RTA methodology and its applications. How to apply the RTA methodology in practice is supported by the examples provided in the Annex.

1.4. SCOPE

The scope of the RTA methodology described herein is for deployment of large WCRs and SMRs of all major types for electricity production and non-electric applications, and their integration with other energy resources. The approach for the introduction of a nuclear power programme is described in the IAEA publication Milestones in the Development of a National Infrastructure for Nuclear Power [1]. The RTA methodology is directly applicable to the pre-feasibility study, feasibility study, invitation and evaluation of bids and, to a lesser extent, the NPP deployment programme.

Relevant examples of the application of the RTA methodology are provided in the Annex to explain its key features and step by step application. Actual implementation in practice is expected to be more detailed with increased specificity to the country.

1.5. USERS

The primary users of this publication are the nuclear energy programme implementing organization (NEPIO), utilities/operator organizations, or others who are responsible for, or involved in, the process of selecting NPP technology. Technical experts of Member States and others actively involved in planning and developing a nuclear power programme and nuclear power project, and those in advising government or utility officials may also benefit from this publication.

The decision makers for reactor technology selection and implementation are the ultimate users of this publication. Accordingly, it is expected that reactor suppliers, architect engineers and constructors, and equipment manufacturers will also benefit from an understanding of how reactor designs and technical proposals will be evaluated, assessed and selected. IAEA Member States also need to obtain or have access to reliable and comprehensive information to perform these comparisons between different NPP designs, such as in the IAEA Advanced Reactors Information System (ARIS) database [2].

The best source of data and information is typically that provided by the technology holder or reactor vendor[1]. The IAEA encourages technology holders to develop a standard technical description of their product and submit it to ARIS, with an emphasis on addressing the RTA methodology key elements and topics that are described in this publication.

Reactor technology assessment is a decision making methodology that contributes to several evaluations in the development of a nuclear power programme. It follows and is dependent upon the national nuclear programme policy objectives. If a Member State then determines that the nuclear power programme is feasible, or if a nuclear programme already exists in the Member State, this RTA methodology provides the input to, and the technology related evaluation process in support of, the bid specification and the evaluation of bids.

This revision develops and presents a comprehensive description of the RTA methodology incorporating lessons learned and feedback received from the IAEA Member States. The RTA is performed and integrated from the initial definition of national nuclear power programme objectives to NPP operation. Member States are encouraged to utilize the complement of IAEA resources and publications to assist in each stage of their nuclear power programme development, reactor technology identification and assessment, technology selection and deployment. Therefore, detailed references that can support the definition and evaluation of KEs and KTs are provided in each subsection of Section 5, while more general or high level references are provided at the end of this publication.

2. REACTOR TECHNOLOGY ASSESSMENT WITHIN A NATIONAL NUCLEAR POWER PROGRAMME

The relationship between the RTA methodology and other activities in nuclear power project development is discussed in this section, and the types of organizational approaches, human resource allocations and deliverable content and quality are presented. Key expectations that have to be understood to commence, and to deliver, a successful RTA methodology are described.

[1] Technology holder here refers to the company/consortium/organization responsible for the design and development of the NPP and its major systems, while the IAEA Safety Glossary [3] defines vendor as a "design, contracting or manufacturing organization supplying a service, component or facility".

2.1. REACTOR TECHNOLOGY ASSESSMENT TIMELINE

RTA methodology supports various decision points within the nuclear power infrastructure programme [1, 4]. It is understood that the RTA develops hand-in-hand within the phases of the nuclear power infrastructure programme.

Figure 1 displays the RTA timeline during the nuclear infrastructure development programme, and how it connects with the major tasks of the PFS (Phase 1), the NPP FS (Phase 2) to the full bidding process (Phase 3). The RTA activity may be initiated during Phase 1 as a part of the FS to identify technologies that are consistent with national needs, requirements and criteria. Prior to the PFS, Member States are encouraged to include in their programmes the necessary capacity building to ensure understanding of the available reactor technology choices, as well as the concept of the RTA methodology. The RTA then continues from the end of Phase 1, where the national policy objectives for the nuclear power programme are defined, up to the selection of the reactor technology for the first project. Therefore, the RTA is a contributor to the PFS and its results; key information developed in the PFS is then used to transition into the RTA to perform and support the decision making process prior to and into the invitation and evaluation of bids. The level of detail and level of effort of the RTA varies as a function of the programme phase. As shown in Fig. 1, the RTA may be conducted as part of the preparation for the PFS in Phase 1. Once the PFS is completed, the RTA moves to the detailed technology evaluations performed in Phase 2, including the FS where the candidate technology and reactor types are determined for the selection process for bid specification and invitation [5].

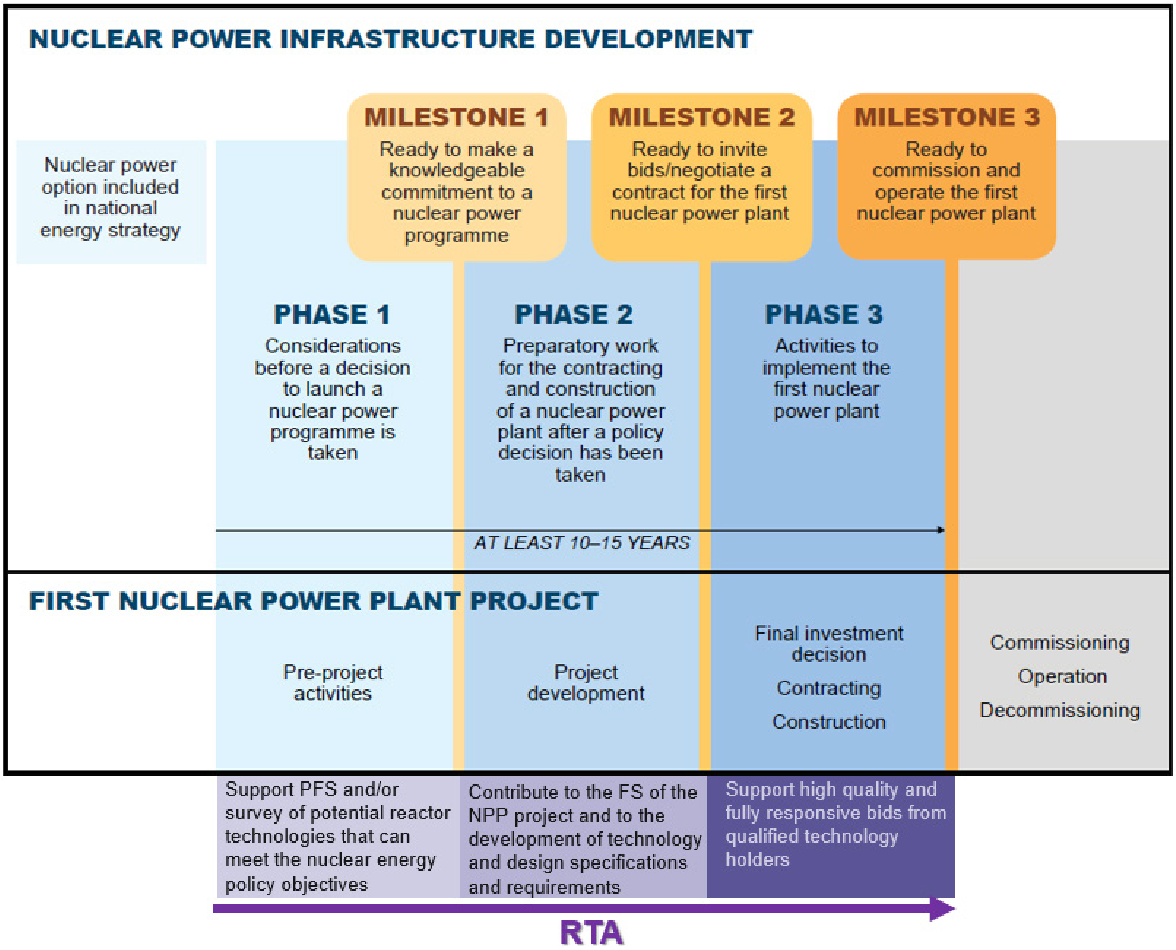


FIG. 1. RTA for near term deployment within the IAEA nuclear infrastructure development programme [1].

For a newcomer country in the early stages of an infrastructure development programme, awareness of the RTA requirements may assist in refining the infrastructure building process or identifying infrastructure and human capacity building needs. The process needs to be linked directly with the programme policy formulation and early evaluations, in which the Member State determines whether the nuclear power option is feasible and viable. Once these policy objectives are determined and validated, an initial RTA can be formulated and conducted before the end of Phase 1 of the infrastructure development programme.

For a Member State expanding its nuclear power programme, these policy objectives are expected to exist at the initiation of the national nuclear power programme expansion.

2.1.1. Reactor technology assessment in Phase 1

The nuclear power PFS is performed, typically by the NEPIO, to implement Phase 1 of the nuclear power programme to determine whether the technology of nuclear power is an appropriate technical and economic option for a specific application, such as electricity or heat generation. The role of the RTA is to first describe and then categorize the available technologies, which may be suitable, to support the decision to proceed by documenting the choice for the NPP technology or a range of technologies. This will define and describe the reactor technologies to support meaningful evaluations to achieve appropriate conclusions and guidance resulting from the PFS. The degree to which the details of the selected NPP technology may be developed at this stage in the nuclear power programme will depend upon the focus and specificity of the policy objectives that have been prescribed for the programme, as well as the goals and expectations that have been developed for the nuclear power programme.

For the PFS or the initiation of the RTA, the review of the NPP technology could be put together in the form of a market survey. The basics of this survey can be obtained from documentation, evaluations and training available through the IAEA and augmented by information obtained directly from technology holders. In the PFS, the RTA will examine the NPP technology in combination with the associated components of the fuel cycle. The results from the PFS, which will be brought forward to the next stage of the RTA (supporting the FS in Phase 2, see Subsection 2.1.2), will include additional details on achievable metrics and ranges for the utility functions that will describe both the NPP and fuel cycle technologies.

The PFS market survey examines the spectrum of those reactor types that may be capable of attaining the nuclear power programme policy objectives. A matrix of these objectives can assist in identifying the types of reactor technologies that could be acceptable in terms of design, availability, constructability and record of performance. A further survey and evaluation would be expected to focus on the additional purposes and goals developed for the specific NPP project that is being evaluated in this PFS.

To this purpose, the IAEA ARIS database of advanced reactor technologies [2] is available for assembling an appropriate set of reactor types for the first cut evaluations. Numerous additional references are available to examine the possible technology approaches to the first levels of consideration, based on the programme objectives and NPP project goals. Therefore, it is appropriate to identify those objectives and goals, or to summarize them from earlier sections of the PFS report. Once this is completed, subsets of the NPP technologies appropriate for the national nuclear power programme will become evident.

For the PFS, the RTA methodology is applied as a screening evaluation to review the available NPP technologies against the national programme policy objectives and NPP project goals (if already available). This can be performed by a limited scope RTA that will still match the national nuclear power programme goals and national policy objectives. This evaluation process performed in the PFS is expected to define conditions or constraints that cause the NPP to be financially and technically attractive compared with alternative approaches or in combination with them (e.g. oil, gas, coal, hydro, renewables). These conditions are then applied in the detailed RTA during Phase 2 to prepare for a bid invitation and evaluation.

The specific focus areas for use in the performance of the PFS are identified for the purposes of the screening evaluation; for example:

(a) NPP technology screening:
   (i) Programme or project specific requirements:
      — Country specific needs, conditions;
      — Size, connectivity, resiliency and stability of the local or national electric grid and plans to move to more distributed energy systems;
      — Seismicity of the identified or selected site;
      — Availability of water resources for ultimate cooling (or use of dry cooling);
      — Accessibility to waterways and/or roads for the transportation of large components or modules;
      — Performance considerations including power level, operability (e.g. black start (to restart the plant without off-site electrical power), island mode on a microgrid, minimum grid-sync power), manoeuvrability (load-following, flexibility to integrate with intermittent energy sources), inspectability, maintainability, availability factor and reliability;
      — Fuel procurement for long term supply;
      — Nuclear safety parameters, such as safety margins, defence in depth, passive versus active safety systems, and probabilistic and deterministic safety evaluation comparisons;
      — Existing nuclear power technologies and existing capacities in other countries.
   (ii) Considerations with respect to the assumption of programme or project risks:
      — Desired level of technology maturity or innovation;
      — Technology maturity risk the country is willing to assume;
      — Level of completion of a design, its regulatory status/licensability, construction and operational history;
      — Use of advanced construction techniques;
      — Energy economics and cost-competitiveness against alternatives;
      — Source of funding and financing mechanism;
      — The use of reactor technologies for non-electric applications including process heat, desalinization and hydrogen production.
   (iii) Technology holder or other programme or project relationship considerations:
      — Technology transfer arrangements with supplier's countries;
      — Regional and international partnerships among user's countries;
      — Long term assurance of availability of structures, systems, components, replacement parts and technical support for the entire design lifetime.
(b) Nuclear fuel cycle screening per NPP technology:
   (i) The key features that have the potential to differentiate between reactor technologies or reactor types are identified and elaborated, such as:
      — Considerations related to the design, procurement and operating experience for the nuclear fuel materials, fabrication, operational expectations and experience;
      — Fuel performance experience extent and quality of experience;
      — Impact of the fuel cycle on the NPP operation including refuelling outages;
      — Long term assurance of fuel supply;
      — Multiple sources of fuel supply and/or procurement options;
      — Technology holder fuel supply arrangements, including fabrication and enrichment services;
      — Spent fuel management options, including spent fuel take-back.
   (ii) The RTA methodology is applied to assess the extent to which these features have the potential to differentiate between reactor technologies or reactor types.
   (iii) The results of the fuel cycle technology evaluation are examined together with the general FS technology assessment results to develop the final guidance of NPP technology to be selected for deployment.

2.1.2. Reactor technology assessment in Phase 2

In parallel to supporting input for the FS, the RTA in Phase 2 provides the more detailed decision making process for the nuclear power project and supports the preparation of the BIS where the reactor type(s) and technology holders are selected by the owner/operator organization. At this point an owner/operator organization is typically established as the second key organization along with the NEPIO, possibly having absorbed the NEPIO RTA team or created a new team with deeper expertise. In some cases, an open bid process rather than an invitation to specified technology holders may be used, in which case the RTA would be used to screen the bids in Phase 3 (see Subsection 2.1.3).

Phase 2 ends with the milestone Ready to invite/negotiate bids.

2.1.3. Reactor technology assessment in Phase 3

The purpose of the RTA here is to assist the invitation and evaluation of bids to obtain high quality and fully responsive bids from qualified technology holders for the application required and to evaluate those design options presented in competitive bids against one another to contribute to the final decision of the technology to be constructed as per the IAEA Nuclear Energy Series NG-T-3.9, Invitation and Evaluation of Bids for Nuclear Power Plants [5].

The RTA team completes its final evaluation when all the information associated with the bidding process has been received. Here the candidate designs will be evaluated with the methodology with each of the key elements selected for decision making to determine how well each has met the expectations of the owner/operator.

2.2. REACTOR TECHNOLOGY ASSESSMENT OVERVIEW

Reactor technology assessment represents a decision making methodology that helps Member States to evaluate and select the most suitable reactor technology option available to fulfil the national energy requirements and its nuclear policy objectives. Figure 2 shows the steps within the IAEA RTA methodology, as applicable to the RTA in Phase 2 of the Milestones approach, representing guidance on how to perform the evaluation of available technical options. During Phase 1, some of the pre-RTA steps may need to be performed by the RTA team, in particular the identification of NPP technologies that have the potential to meet the requirements, in order to facilitate conducting a meaningful RTA to support the PFS. In some cases, the potential NPP technologies are identified even earlier, by entities outside the NEPIO from external political, economic or societal considerations.

Elements in Fig. 2, such as inputs and outputs, are described in the following subsections; the pre-RTA steps are outlined in Section 3 and the steps involved in the RTA methodology are detailed in Section 4.

2.2.1. National policy goals, constraints and requirements

The national nuclear policy objectives are commonly based on:

- National energy plan;
- Relevant national strategies;
- Economic and financial goals and constraints;
- National infrastructure — current status and future expectation;
- Local demographics and physical infrastructure;
- Regulatory and safety requirements;
- Physical security, cybersecurity, physical protection and safeguards requirements;
- Site and environmental conditions.

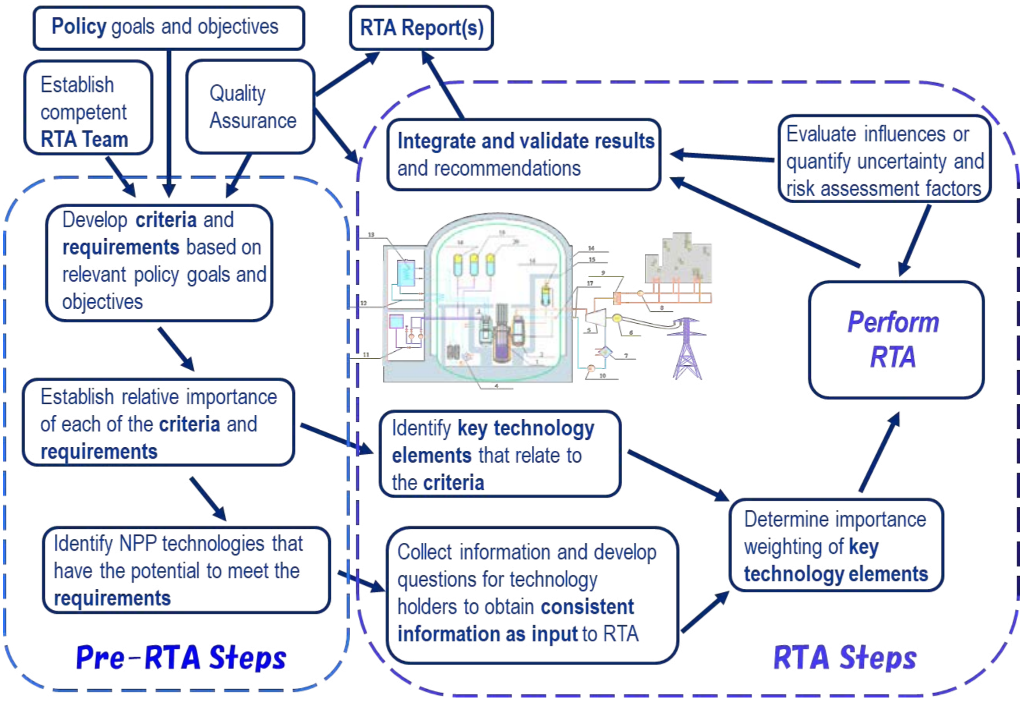


FIG. 2. The RTA methodology steps, as applicable in Phase 2 of the IAEA Milestones approach.

When national policy on safety and other high level design requirements are clearly defined, reactor designs and technologies that would not meet these requirements are not considered.

2.2.2. Establishing a competent national RTA team

The RTA team's role is to gather credible and consistent information for the different reactor technologies of interest, using the importance weighting structure of the policy objectives, as well as to understand the technology and the RTA methodology components and structure sufficiently well to ask the right questions to the technology holders to derive the necessary comparative data. Furthermore, the RTA team is expected to gather or develop methods to evaluate the technology holders' critical data and responses to key questions, such that the technology holder responses can be ensured to be fully applicable to the reactor technology application under assessment.

As described in Subsection 2.1, the RTA develops hand-in-hand within the phases of the nuclear power infrastructure programme. Therefore, both the organizational structure to support the RTA and the size and qualifications of the RTA team are determined with the organizational development in the infrastructure capacity building programmes. It is imperative that the NEPIO (Phase 1) and owner organization (Phase 2) take full responsibility for the conduct and results of each phase of the RTA. Reporting within the organization is established such that the technical and managerial RTA team is assembled and directed to perform the mission by top management. Accordingly, the RTA team's results and suggestions will be reported directly to the top level decision maker(s) in the owner organization. The use of consultants as well as local expertise from universities is advised to augment or supplement the RTA team with specific expertise. Consultants report their input to the RTA team management, so that it becomes a part of the team's analysis and results. Generally, consultant reports are not the sole basis for major decisions by the RTA team.

The RTA team's size and characteristics, in terms of the types of disciplines that are represented by an organization or consultant participants and local experts, will also vary as a function of the phase of the RTA. In addition, the results of the RTA work in the NEPIO policy studies or in the early PFS will determine the nature of the detailed technical evaluations necessary over the course of subsequent RTAs.

A range of personnel and disciplines, as well as organization personnel and consultant breakdowns, is required to represent various levels of infrastructure development. In general, the RTA team will possess full expertise in design, engineering, construction and operation of the NPP and its environs. It will also have adequate knowledge on fuel, fuel cycles and waste management.

The baseline description shows that a newcomer Member State may require higher levels of external resources (advisory or contributing consultancy) in the early stages of the technical assessment for the PFS. This approach is also an important component of capacity building for the newcomer country. A Member State with an existing nuclear power programme, which is preparing the FS for expansion, may use consultants only for specialized tasks that the organization has not performed routinely and be more reliant on existing local expertise. The IAEA can offer at any phase of the national power programme development relevant training courses in supporting the national capacity development and sustainable building and support the learning process on the RTA methodology within the framework of a national nuclear power programme.

2.2.3. Reactor technology assessment quality assurance

This subsection describes the quality expectations of the RTA process and describes the documentation that is suggested in the performance of this work. It is important that the quality and documentation expectations are established early in the process of the RTA and that they are designed to be applied consistently throughout this endeavour.

2.2.3.1. Performing

This publication outlines the considerations and steps involved in performing RTA. The three main ingredients in ensuring high quality during the performance of RTA in any phase of the nuclear power programme development are: (1) well defined national requirements for the nuclear power programme; (2) sufficient objective information on the reactor technologies being assessed; and (3) an RTA team with qualifications that cover all key elements of the RTA.

2.2.3.2. Documenting

The benefit of the proper documentation associated with RTA development and determination relates to the role that technology assessment will play in highlighting the key issues that have to be examined for the chosen technology design, construction and commissioning. A high level of detail and quality in the documentation can enable an enhanced understanding of the technology, reactor options and technology holder design differentiation important in the selection, construction and operation processes. This process may be active over a long period within the integrated infrastructure development. Therefore, as soon as the Member State begins the capacity building and feasibility study, it is advised that they specify the formality of the documentation and a quality programme to support this.

Since the assumptions and analyses in the RTA for the FS will support key decisions by the organization, it is important that such information is appropriately documented and that the analyses, assumptions and input to decision making are subjected to quality document preparation, review and approval practices. In this process it will be important for the organization originating the documents to clearly state the purposes for which each document is to be applied. Methods for augmentation and revision of these documents can ensure that the programme follows formal processes in the document change process.

As with other tasks in the RTA work scope, there are dual purposes here. Firstly, the documents associated with these decisions are important corporate records in a formal way. Secondly, the use of formal documentation and a quality programme can be a strong component of capacity building within the organization, especially for the newcomer Member State. The results of the RTA process at one stage will build upon the previous work and will support ongoing and future decision making. If the work upon which these decisions are made is not built using a quality programme, and that causes decisions to be based upon faulty data, assumptions, or analyses, then the subsequent decision making will become unsupportable. The decision to build quality documentation that is produced using a quality programme is important to establish early in the process.

3. PRE-REACTOR TECHNOLOGY ASSESSMENT STEPS

Assuming the national energy planning results have led to the (possible) inclusion of nuclear power in the energy mix, a survey of available nuclear technologies that could meet the objectives is initiated in the form of a PFS [1]. Before conducting RTA for assessing these available technology options in Phase 1 of the IAEA Milestones approach, several important inputs need to be available or established.

As shown in Fig. 2, these RTA pre-steps consist of:

— Developing criteria and requirements based on relevant national policy goals and objectives;
— Establishing the relative importance of each of the criteria and requirements;
— Identifying the NPP technologies that have the potential to meet the requirements.

3.1. DEVELOPING USER CRITERIA AND REQUIREMENTS

To develop user criteria and their bases for the purposes of RTA, it is necessary to refer to the overall national nuclear project objectives. The goal of RTA, whether it is done by the NEPIO in Phase 1 or by the owner in Phase 2, is to differentiate one reactor technology and/or design from another, in order to determine the technology or design that best achieves these goals. Following six decades of reactor design and operational experience, in the past four decades several organizations have developed user requirement documents that specify in detail how key and specific technical objectives need to be met for a licensable, successful NPP design. These documents provide clear considerations to the designer to achieve user requirements. However, this generally creates a matrix of detailed design information and data too unwieldy to use in differentiating between technologies or designs. As a result, the user criteria and bases for RTA need to be derived by the programme or project developer, potentially with user requirement documents as considerations. Prior to the RTA, the national or corporate policy objectives are identified to describe in a specific manner what the nuclear power programme or project intends to achieve. One approach a newcomer Member State may use to assemble the policy objectives for its nuclear power programme is to review the outcomes of the national energy policy document where the following questions are to be answered:

— Under what conditions does this programme or project make sense?
— Why has this course of action been chosen for this endeavour?
— What are those specific outcomes that will create the expected success?

These policy objectives are to be developed to the point where they are documented clearly in writing for the benefit of the stakeholders and for use by the RTA team. In addition to ensuring that these policy objectives are identified with clear descriptions, for the decision making process to begin, the policy maker needs to prioritize these policy objectives by assigning a relative importance weighting to each one. The ranges of conditions that lead to (or allow) project success facilitate these weightings.

Once these policy objectives and their respective importance are defined for the project, the RTA team identifies those technology features which can best be used to demonstrate how each of these policy objectives will be evaluated or measured. Depending on the working relationship between the RTA team and the policy maker(s), this process may be iterative. However, at some point in time the policy maker(s) will fix the set of prioritized policy objectives and allow the RTA team to perform the detailed process to derive the results and guidance from the RTA. This involves creating subsets of KEs that affect the policy objectives and evaluate their respective importance and performance for the candidate reactor designs.

The organization setting the policy objectives is expected to be familiar with resource documents such as the IAEA common user considerations (CUC). The RTA team is expected to be familiar with the details set forth in several other sets of guidance documents that explain user requirements and criteria. These other documents detail several different approaches, which can be evaluated and incorporated as desired into the objectives and technical components for the Member State's RTA effort. Reference [6], Common User Considerations (CUC) by Developing Countries for Future Nuclear Energy Systems, published by the IAEA in 2009, describes the needs as expressed by newcomer Member States in terms of development and deployment of new nuclear power technology. These needs were derived on the basis of the input of a large number of experts, acting as 'technology users' and representing 35 developing countries. The report also incorporates the guidance of experts from 'technology holder' countries, as well as lessons learned from several international and IAEA activities, including user requirements programmes in technology holder countries.

The CUC publication covers the general technical and economic characteristics of NPP and fuel cycle options (including waste management facilities), as well as associated support services requested by potential users of future nuclear technology in developing countries. Utility requirements for the purposes of the RTA can be developed on the basis of existing harmonized approaches of utilities operating in different contexts and with different regulations. Compliance of reactor designs with these established utility requirements provides confidence that their technology is ready for use.

Other publications may be controlled by the originating organizations and may not be readily accessible to Member States. For example:

— The European Utility Requirements (EUR) [7] were developed by several European utilities beginning in 1995 with the goal of establishing a common consensus related to the design expectations and requirements for future light water reactors (LWRs) in Europe. The resulting base document of four volumes covers major policies and objectives, as well as the generic and specific nuclear and conventional system designs. Several additional volumes cover a variety of design and technology types. The document is also applicable to nuclear power plant installations in other markets.
— The EPRI Advanced Light Water Reactor Utility Requirements Document (ALWR URD) [8] and the EPRI Utilities Requirement Document for Small Modular Reactors [9] present clear and comprehensive utility requirements for advanced LWRs in the United States of America. The ALWR URD was developed with the management and coordination of the Electric Power Research Institute (EPRI) and under the leadership of EPRI member utilities. The focus of the development and application of the document was on both reactor and facility design and licensing, such that the URD has been referenced heavily in the Nuclear Regulatory Commission licensing proceedings for these reactor types.

Policy objectives are prepared by Member States based on their energy planning needs, while the technology features can be drawn from a compilation prepared from IAEA infrastructure, CUC, and bid specification and evaluation documents [5]. The approach in RTA is to classify technology features that support each policy objective into KEs and KTs. Then these are evaluated using two critical factors analysis:

— The importance (weight) that each KE and KT holds for the decision maker for the reactor technology application under evaluation;
— The comparative value (score) the RTA team determines and assigns to each reactor technology that is being evaluated.

The listings of general criteria are commonly separated into two subsets, although there is usually expected overlap between them. They are:

(a) General user criteria identified as candidate policy objectives that can be treated in a direct and open manner in the assessment process. Examples of general user criteria based on the IAEA CUC, EUR and EPRI ALWR URD publications are shown in Table 1. Depending upon the Member State's nuclear power programme or the nuclear project under consideration, these criteria may be selected as policy objectives, or technical criteria; however, some of these publications carry a considerable cost. The primary listing of general user criteria for the NPP technology is shown in the first column of Table 1;
(b) General technical criteria are in addition to the high level user criteria and are included in Table 2 as specific to the NPP.

General criteria in consideration of programme or project risk: There are a variety of factors that increase the risk of a major technological project; likewise, there are factors or decisions that may reduce this risk. The risk impacts due to these factors may be actual or perceived. These are termed 'non-technical' factors because they are generally not associated directly with the reactor technology. Examples of such factors may relate to the long term consequences of technology selection, technology holder historical construction performance, reactor technology transfer history or opportunities, political considerations, national resource constraints or opportunities, or human resources availability. These non-technical considerations or factors can have a prominent impact, if not a determining role, on the decision. For example, some may be grouped as items under a policy objective such as national economic development, public involvement and support, or even programmatic risk minimization.

TABLE 1. EXAMPLES OF GENERAL USER CRITERIA FOR NUCLEAR POWER PROGRAMME OBJECTIVES

General user criteria	Common user considerations [6]	European utility requirements [7]	Utility requirements document [8]
Sustainability	Sustainable lifetime operation	Sustainable lifetime operation	Programme policy statement
Power generation demand	To be owner specified	Plant size	To be owner specified
Electrical grid characteristics	To be owner specified	Specific planning capacity provided	To be owner specified
Site characteristics	Remote location; design to accommodate external events	To be owner specified	To be owner specified
Environmental impact	Off-site release limits	Off-site release limits	Off-site release limits
Nuclear safety	Policy statement for safety programme requiring analysis of design, adequate safety margins and consideration of design extension conditions (including severe accidents)		
Regulation and licensing	Compliance with regulations and standards	Compliance with regulations and standards	Major focus of overall URD effort
Radiation protection	Occupational radiation exposure compliance	Occupational radiation exposure compliance	Occupational radiation exposure compliance

TABLE 1. EXAMPLES OF GENERAL USER CRITERIA FOR NUCLEAR POWER PROGRAMME OBJECTIVES (cont.)

General user criteria	Common user considerations [6]	European utility requirements [7]	Utility requirements document [8]
Nuclear fuel cycle policy	Assurance of fuel supply	Fuel cycle cost targets and guidance on policy	Expected assurance of fuel supply
Nuclear waste management	Spent fuel, waste and decommissioning services	Spent fuel and radioactive waste disposal targets	Spent fuel, waste and decommissioning services
Safeguards	Intrinsic proliferation resistance	To be incorporated in design	Programme policy statement
Security and physical protection	Intrinsic physical protection	To be incorporated in design	Programme policy statement on sabotage protection
Emergency planning	Plan and implementing process derived in concert with nuclear safety requirements		
National participation	User involvement, technology transfer	User considerations	User considerations
Industrial development	Cost reduction through local content	User considerations	User considerations
Human resource development	Technical and project management development	Programmatic component	Programmatic component
Economics	Generation costs; Construction schedule	Generation costs; Construction schedule	Programme policy statement; Economics
Project financing	Type of contracts; Supplier support	User considerations	Programme policy statement; Economics
Other	Public perception; Assurance of component and spare parts supply; Supplier qualification	Additional key technical features described	Additional key technical features described

TABLE 2. EXAMPLES OF GENERAL TECHNICAL CRITERIA FOR NPP TECHNOLOGY

General technical criteria	Common user considerations [6]	European utility requirements [7]	Utility requirements document [8]
Proven technology	High maturity level	Licensable Standardized	Programme policy statement
Standardization	Cost and component/spare parts replacement	Main policy statement	Programme policy statement

TABLE 2. EXAMPLES OF GENERAL TECHNICAL CRITERIA FOR NPP TECHNOLOGY (cont.)

General technical criteria	Common user considerations [6]	European utility requirements [7]	Utility requirements document [8]
Simplifications	Simplified design while assuring performance	Policy statement	Programme policy statement
Plant lifetime	50–60 years	40 years without refurbishment; 60 years extension	60 years
Availability	Equal to or greater than 90%	Capacity factor >90 % Refueling < 20 days	Top tier requirements: plant performance
Operability and manoeuvrability	Safe shutdown on load rejection; base load is expected	Detailed requirements provided	Top tier requirements: plant performance
Inspectability and maintainability	Accept current practice and demonstrated improvements	Performance assessment methodology is specified	Programme policy statement
Refueling schedule	18–24 months	Flexible between 12 and 24 months	18–24 months
Nuclear island	Level of detail not developed	Provides generic and European preferences	All details provided
Conventional island	Level of detail not developed	All details provided	All details provided
Electrical systems and components	Level of detail not developed	All details provided	All details provided
Instrumentation and control systems	Level of detail not developed	All details provided	All details provided
Balance of plant	Level of detail not developed	All details provided	All details provided
Civil works and structures	Level of detail not developed	All details provided	All details provided
Plant simulator	Level of detail not developed	All details provided	All details provided
Mechanical, instrumentation and control, electrical equipment	Level of detail not developed	All details provided	All details provided
Architectural finish	Level of detail not developed	All details provided	All details provided

3.2. IMPORTANCE RANKING

A Member State's NEPIO or owner/operator using the RTA methodology establishes their own candidate listing based on its own objective assessment against the relevant user, technical and risk criteria.

Table 3, which recasts the criteria from Tables 1 and 2, represents an example of what might be considered by a Member State's NEPIO or owner/operator as the candidate policy objective listing for their policy objectives. This listing is not intended to be complete; rather it is an example drawn from findings related to the CUC, combined with the other cited references.

In the next step, the policy makers identify those candidate policy objectives that would be of highest importance. Those objectives that are rearranged in rank order for this example are shown in Table 4.

The number of objectives to be selected from this set will vary and will generally depend on the breadth of scope for the individual objectives and on the approach used for the RTA. Typically, four to eight policy objectives or goals might be chosen.

The relative importance or weighting of each of these objectives have to be then assigned to be consistent with policy. This is accomplished by reviewing the rank ordered list and then assigning the weights such that the total point value matches the typical scoring for the applied RTA. The suggested weighting will sum to 100%.

3.3. IDENTIFYING POTENTIAL TECHNOLOGIES

The last pre-RTA steps are to identify NPP designs and technologies that have the potential to meet the user and technical criteria; this step will be outputting a list of technologies that have the potential to meet the general criteria and policy objectives. This list now represents the set of technologies to comparatively assess by applying the RTA methodology (Fig. 2).

TABLE 3. EXAMPLE OF CANDIDATE POLICY OBJECTIVES FOR A NATIONAL NUCLEAR POWER PROGRAMME

Candidate policy objectives for a Member State nuclear power programme aim to achieve electricity production for energy independence and national industrial development
Candidate policy objectives:
Maintain and enhance nuclear safety performance
Achieve nuclear electricity production at a competitive cost
Meet the national energy plan for on-line production with a capacity and time frame specified by the plan
Develop national participation through acquisition of nuclear technology
Use proven technology (such that the design concepts and plant components have been demonstrated in application)
Ensure that substantial long term technical support is available from the technology holder's organization or from other industry relationships
Ensure sustainability through assurance of components supply over the facility's lifetime
Ensure fuel supply through assurance of materials supply, proven fuel design and performance and diversity of suppliers
Promote industrial development to support plant construction and long term component production related to fuel supply
Minimize the impact of the programme on the local environment
Achieve long term closure option for the nuclear fuel cycle
Minimize construction and financing costs by ensuring that the proposed construction schedule is met
Maximize the value this first project contributes to the long term nuclear energy programme
Ensure that this project builds and sustains national and local human resource development

TABLE 4. EXAMPLE OF POLICY OBJECTIVES AND ASSIGNED RANGE OF RELATIVE IMPORTANCE FOR ELECTRICITY PRODUCTION, ENERGY INDEPENDENCE AND NATIONAL INDUSTRIAL DEVELOPMENT ASSUMING A NEAR TERM DEPLOYABLE NPP

Relative importance	Policy objective
High	Meet the national energy plan with on-line production with an electric power capacity and timing specified by the plan
High	Achieve nuclear electricity production at competitive cost in Member State
High	Establish and maintain nuclear safety
High	Develop national participation through acquisition of nuclear technology
Medium	Ensure substantial long term technical support is available from the technology holder organization and/or from other industry relationships
Medium	Use proven technology (such that the design concepts and plant components are demonstrated in application)
Medium	Minimize construction and financing costs by ensuring that the proposed construction schedule is met
Medium	Ensure fuel supply through assurance of materials supply, proven fuel design and performance, and diversity of suppliers
Low	Promote industrial development to support plant construction and long term component production related to fuel supply
Low	Ensure that the project builds and sustains national and local human resource development
Low	Minimize the impact of the programme on the local environment
Low	Maximize the value this first project contributes to the long term nuclear energy programme
Low	Ensure sustainability through assurance of components supply over facility lifetime
Low	Achieve long term closure option for the nuclear fuel cycle

4. REACTOR TECHNOLOGY ASSESSMENT METHODOLOGY

This section describes the method that may be utilized to perform quantitative analysis in the RTA, as illustrated with the examples provided in the Annex.

4.1. DECISION MAKING METHODOLOGY

The decision or selection matrix in the RTA displays the objective or subjective assessment of each of the NPP options against a list of KEs and KTs. This simple approach assigns percentage weights to each KE/KT and scores or rankings to each NPP option, on a comparative or absolute basis, then adds the results to derive an overall score or ranking.

A rational decision making process presupposes that there is one best outcome. Because of this assumption, it is often classified as an optimizing decision making methodology. The objective function could be formulated around minimizing the technology performance risk and minimizing cost and schedule risk, while under a set of technology boundary conditions (resilience, flexibility, adaptability, safety, environmental impacts, security, reliability, sustainability). Such a methodology also assumes that it is possible to consider each of the KEs/KTs for every option and to know or predict the future consequences of each. Finally, a rational decision making methodology is meant to negate the role of emotions or opinions and also to eliminate the biases in decision making. However, certain KEs and/or KTs may be difficult to quantify absolutely or even comparatively and therefore may require a large resource and time commitment of the RTA team to obtain or generate the required information.

The generic decision making methodology is summarized in the eight-step model shown in Table 5 and related to specifics of the RTA methodology.

There are several multicriteria decision making methodologies that could be used including MAXMIN, MAXMAX, SAW, AHP, TOPSIS, SMART, ELECTRE [10]. Some advanced techniques use a hierarchical structure, such as the analytic hierarchy process (AHP) that can endogenously assess relative importance across indicators as well as incorporate data that are both quantitative and qualitative in nature; however, the AHP can become computationally intensive. The decision making methodology presented in this publication is based on the simple multi-attribute rating technique (SMART). It corresponds to step 6 in Table 5, where NPP options are scored against the KEs/KTs previously defined and evaluated for their importance and weighted against the national needs and goals (steps 2, 3 and 5 in Table 5). While 5, 7 and 10-point scoring scales are the most commonly used, the SMART methodology allows for use of a smaller range if the data do not discriminate adequately, so that, for example, alternatives which are not significantly different for a particular KE/KT can be scored equally. This is mostly important when confidence in the significance of the differences is low. In these cases, less of the range is used to ensure that low confidence data differences do not present unwarranted discrimination between the alternatives. When quantitative data are unavailable, qualitative reasoning, expert judgement and/or consensus scoring can be substituted and documented in the final RTA report.

TABLE 5. AN EIGHT STEP DECISION MAKING METHODOLOGY

Generic step	Specific to RTA methodology
1. Define the problem	To select the best nuclear power plant technology to meet national needs by minimization of performance and costs/schedule risks, and maximization of national industrial development
2. Determine requirements for the desired solution	Example: nuclear power plant net electrical output, minimum design life, technology maturity, licensing status, etc.
3. Establish the goals	KEs/KTs described in this publication (Section 5)
4. Identify alternatives	Type of nuclear power plants (e.g. large WCRs, SMRs)
5. Develop rationales for relative importance of goals	As described in this publication (with examples provided in the Annex)
6. Select decision making methodology	RTA methodology as described in this publication (Section 4)
7. Evaluate alternatives and select the best one	Conduct and document the RTA along the examples provided in this publication (with examples provided in Annex)
8. Validate the selection against step 1	Not explicitly discussed in this publication

A decision analysis and iterative refinement in RTA methodology is illustrated in Fig. 3, with two use case examples for illustration of possible RTA application scope.

The rating and scoring for the selected KEs/KTs is to be consistent across all KEs/KTs, such that no bias is introduced in the assessment. Modifying the scoring approach from one feature to another is a way in which bias may be introduced and is to be avoided. One example of poor practices that would cause an error in constructing the evaluation process is setting the scoring value range differently for the KEs/KTs that are perceived to have lesser or higher importance, for example, choosing a lower and narrower range to apply for less important features (say, 1–5) and a broader range (say, 1–10) for more important features. The numerical scoring ranges that are assigned (lower and upper bounds) are always the same.

Typical ranges for scoring NPP options in the suggested decision making methodology is 1–5 (used throughout this publication) or 1–9, with intermediate scores either allowed or not. These ranges are convenient because they have an integer midpoint, they have well accepted verbal designations and they relate easily to a qualitative scoring approach. Figure 4 provides visualizations of 5-point and 9-point scoring scales.

Table 6 shows a poor and good example on how to set an unbiased range for scoring, taking as an example the KT on capacity factor, with the three reactors (A, B, C) having capacity factors of 75%, 88% and 95%, respectively. In framing the range of values per KE/KT it is commonly accepted to use the industry performance information, historical data for a reasonable and relevant time period and use a reasonable and relevant range of values for assessing technologies. Examples of scoring and rationales for scoring per every KE and KT are detailed in the Annex.

4.2. SELECTION OF TECHNOLOGIES FOR THE ASSESSMENT

There are numerous nuclear reactor designs available for near term deployment to produce electricity or heat, and many more under development [2]. They cover a wide range of power outputs, design features, market, technology and licensing readiness, and construction and operational history, which need to be compared against the basic requirements of the national nuclear power programme.

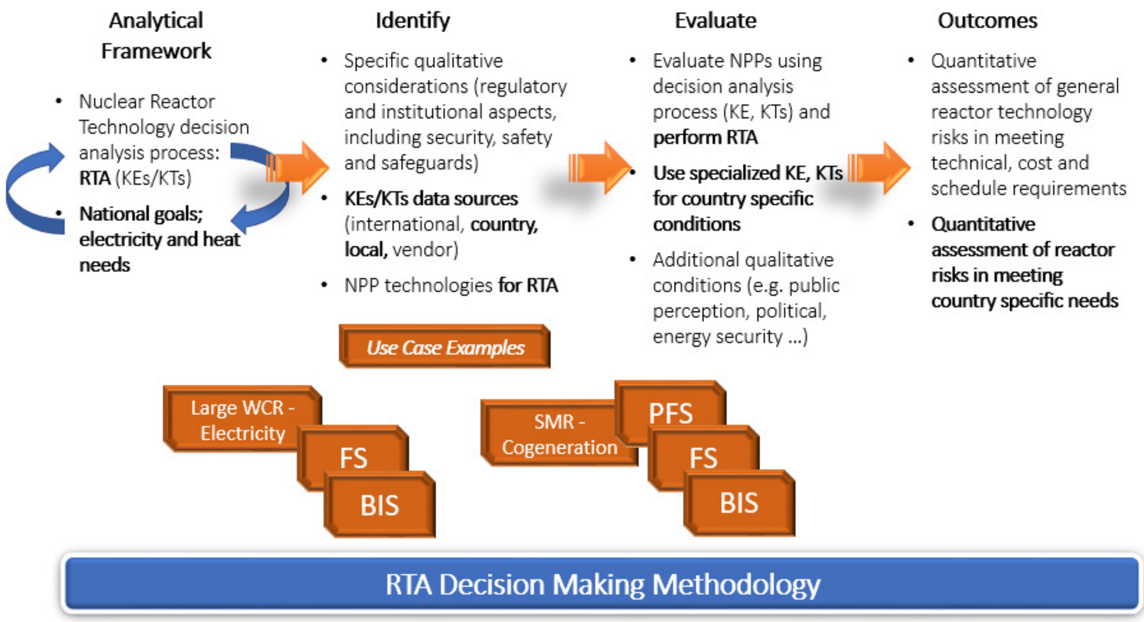
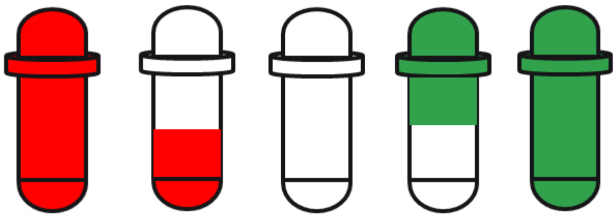


FIG. 3. Decision analysis and refinement in RTA methodology.



NPP design satisfies the KEs/KTs to:

Little extent / not acceptable	Lesser than medium extent, but greater than a little extent	Medium extent	Greater than medium extent, but not to a large extent	Large extent
5 – point scoring system				
1	2	3	4	5
9 – point scoring system				
1	3	5	7	9

FIG. 4. Scoring system in decision making methodology.

TABLE 6. SETTING AN UNBIASED RANGE OF SCORING: EXAMPLE FOR NPP CAPACITY FACTOR[a]

Poor: Too wide range (0% – 100%)					
Range	0%	25%	50%	75%	100%
Score	1	2	3	4	5
Nuclear power plant design				A (75%)	B (88%) C (94%)
Good: Reasonable and relevant range (80% – 100%)					
Range	<80%	80 to 84%	85 to 89%	90 to 95%	>95%
Score	1	2	3	4	5
Nuclear power plant design	A (75%)		B (88%)	C (94%)	

[a] The net capacity factor is the unitless ratio of an actual electrical energy output over a given period of time to the maximum possible electrical energy output over that period. NPPs are at the high end of the range of capacity factors, ideally reduced only by the availability factor (i.e. maintenance and refuelling).

Before embarking on the assessment of potentially suitable technologies, all unsuitable technologies will ideally be eliminated, with justification and reference to the national policy goals and the developed or adopted set of user requirements. Only then can the potentially suitable technologies be assessed against the user criteria, using the RTA methodology described in this publication, or a similar method.

It is important that the RTA compares alternative options that are similar, for example, if a ~1200 MWe generation capacity is required, this could be accomplished in different and equivalent ways, yet the various potential technologies to achieve it may not be similar enough to be compared with a single RTA. For example, a single large reactor unit or several co-located SMRs could achieve a ~1200 MWe generation capacity. Each of the solutions can achieve essentially the same goal, but they are very different when it comes to their assessment against some KEs and KTs, making an RTA that mixes large WCRs and SMRs almost impossible to perform objectively. In this case, two RTAs are suggested, one for different large WCR technologies and another for different SMR technologies. Only then can the assessment of the alternative technologies in all key elements consistently consider each alternative as a solution for the same problem and the same requirements. The decision on one large or several SMRs then needs to be done at a different level.

4.3. SOURCES OF INFORMATION

Significant design information on advanced WCRs, as well as SMRs and non-water cooled reactors, is available in the IAEA ARIS database [2]. Other reliable public sources that can provide further details, or even more up-to-date information, can be found in third party studies, on vendor web sites, through direct inquiries to the vendor, and most importantly, any licensing submissions or environmental impact assessments that have been published by the regulatory authority of any country. Many of these can support RTA during Phase 1.

To obtain more detailed and precise design and economic performance information, it is expected that the potential buyer (NEPIO or operating organization) will need to sign a nondisclosure agreement with the technology holder to learn more about their reactor technologies and interact with potential vendors, who may or may not have interest in bidding on the NPP project. The technology holders will ideally then respond to a well thought out and uniform set of questions developed by the RTA team. This is essential in performing RTA during Phase 2, leading up to the BIS. The selection of candidates for the invitation to bid results from the work of this stage.

More KE specific sources of information are given in each subsection in Section 5 and, along with more high level references, at the end of this publication.

5. KEY ELEMENTS AND KEY TOPICS

The RTA methodology consists of ten KEs that group the user and technical criteria, or KTs, as a base for decision making. They are:

— KE1: Site and environment;
— KE2: Fuel cycle;
— KE3: Nuclear safety;
— KE4: Nuclear island design and performance;
— KE5: Balance of plant (BOP) design and grid integration;
— KE6: Balance of plant (BOP) design for purposes other than electricity production;
— KE7: Safeguards and protection;
— KE8: Technology readiness;

— KE9: Project delivery;
— KE10: Economics and financing.

Every KE is further defined by its KTs. Not every KE is defined by the same number of KTs, nor will each of them have the same importance (expressed as % weight). The correct use of the RTA methodology as described in this publication is based on the approach as sketched in Fig. 5. In the first step the importance is assigned to every KE that is based on developed rationales derived from general user criteria and national policy objectives. The following step requires a comprehensive analysis of each KE by evaluating corresponding KTs for NPP technologies under consideration. The associated weights will reflect on the importance of the KTs and/or lack of information available for their thorough analysis. Comprehensive rationales are required to be developed prior to assigning the weights. These rationales will help in each phase of applying the RTA and will support developing communication with the technology holders, preparation of the bids and their evaluation.

Table 7 provides examples on how these ten KEs can contribute to the scope of an RTA performed for a nuclear power project during Phase 1, Phase 2 and Phase 3 of a national nuclear infrastructure programme. It also shows how the relative importance could vary between Phase 1 or Phase 2 RTA for either a large WCR or SMRs based electric generation plant, and for a non-electric application during Phase 2. As these are merely examples, the RTA team has the task to select these in relevance to the national NPP programme or project. These are derived from a compilation of Member States' feedback and expert opinions and are intended to demonstrate the process of ranking the KEs. The policy team determines the ranges, values and the rationale for the weighting factors in the RTA.

Technology readiness may be an important consideration in Phase 1, if innovative technologies are assessed, or not at all, if it is a pre-requirement imposed by national policy. During Phase 3 the RTA team will engage significantly with the technology holders and either ascertain the design's market readiness and safeguardability, or eliminate it from the evaluation; therefore, KE 7 and KE8 are excluded from the RTA scope. Economic KEs, namely KE9 and KE10, are often very difficult to quantify in the early Phase of a national nuclear power programme or may be handled outside the RTA team's scope of work, but in most cases, they are very important in the later phases that involve financial and schedule commitments.

Table 7 shows that the majority of the KEs are considered at an increasing level of detail through the process of the RTA to provide good support to the invitation to bids. The NEPIO or owner/operator is expected to actively engage the technology holders in the FS and invitation to bid process, and to provide directions with regard to the decision making process. A summary description related to it is provided in the IAEA Nuclear Energy Series NG-T-3.9, Invitation and Evaluation of Bids for Nuclear Power Plants [5].

While the following subsections describe each KE and KT in some detail, and in some cases provide suggestions for detailing further through subtopics (STs), Table 8 provides a summary map of the KEs and KTs covered by the RTA as described in this publication.

Numerical examples for scoring alternative technologies against each KT are provided in the Annex. For example, the KEs for decision making that are most likely to influence the process of evaluation

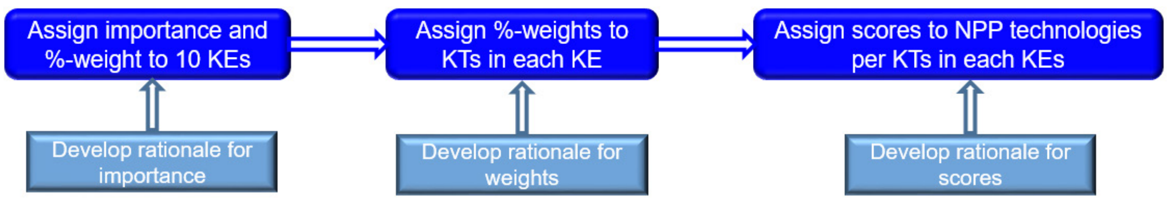


FIG. 5. Implementation of RTA methodology.

TABLE 7. EXAMPLES FOR PLANNING THE SCOPE OF THE RTA IN DIFFERENT MILESTONE PHASES AND FOR ASSIGNING THE IMPORTANCE OF KEY ELEMENTS FOR DIFFERENT RTA APPLICATIONS

RTA Key Elements	Included in the RTA during:			Importance in an RTA performed for:				
	PFS for nuclear power programme (Phase 1)	FS for NPP project (Phase 2)	Bid evaluation (Phase 3)	Large WCRs		SMRs		Non-electric with SMRs (Phase 2)
				Phase 1	Phase 2	Phase 1	Phase 2	
KE1: Site and environment	No	Yes	Yes	N/A	H	N/A	M	H
KE2: Fuel cycle	Yes	Yes	Yes	H	M	M	M	M
KE3: Nuclear safety	Yes	Yes	Yes	H	H	H	H	H
KE4: Nuclear island design and performance	Yes	Yes	Yes	M	H	M	M	M
KE5: Balance of plant design and grid integration	Yes	Yes	Yes	M	M	L	L	N/A
KE6: Balance of plant design for other than electricity production	N/A	N/A	N/A	N/A	N/A	N/A	N/A	H
KE7: Safeguards and protection	No	Yes	No	N/A	L	n/a	M	M
KE8: Technology readiness	Yes/No	Yes	No	M	L	H	H	H
KE9: Project delivery	No	Yes	Yes	N/A	H	N/A	M	M
KE10: Economics and financing	No	Yes	Yes	N/A	H	N/A	H	H

Note: Some KEs are ranked low, not to reflect their absolute importance, but reflecting the relative importance to other KEs in this particular application or that they are unlikely to be a differentiator. H: high; M: moderate; L: low; N/A: not applicable.

associated with the FS, once a site has been selected, are the site and environment, the grid specific characteristics and those other parameters that affect the preliminary economic evaluations.

On the other hand, the RTA that provides input for the PFS may exclude KE1, because no sites have been selected, and KE7 regarding the safeguards, because it is simply assumed that all IAEA safeguards conditions will be met before commissioning. Purely technical KEs, namely KE2 to KE5/6, can always differentiate between different technologies and will ideally always be part of the RTA in any phase. Their relative importance may differ depending on the preferred fuel cycle options and size of the NPP relative to the grid.

TABLE 8. SUMMARY MAP OF THE KEs AND KTs

KT Number	KE1 Site and environment	KE Fuel cycle	KE3 Nuclear safety	KE4 Nuclear island	KE5 Balance of plant and grid integration
1	Site seismicity	Fuel materials and components	Implementation of defence in depth (DiD) philosophy	Plant size	Net thermal efficiency
2	Meteorology and hydrology	Fuel product supply chain	Safety design philosophy	Plant availability and capacity factors	Grid electrical code requirements
3	Water resources	Fuel unit fabrication	Degree of diversity and redundancy	Plant lifetime	Protection against internal and external hazards
4	Population	Fuel operating experience	Protection against internal and external hazards	Standardization	Standardization of major components
5	Site access for construction and operation	Refueling outage	Response to off-site power loss	Simplification	Power requirement from grid under normal operation
6	Site size	Fuel flexibility	Completeness of OLCs, SAR, PSA, O&EP, SAMG	Constructability	Ability of grid to accept generating capacity
7	Environmental & radiological impact	Suitability to indigenous fuel fabrication	Results of deterministic safety analysis	Operability, inspectability, maintainability and reliability	
8	Other external events	Medium term spent fuel storage capacity	Results of probabilistic safety assessment	Manoeuvrability	
9		Long term spent fuel storage	Mitigation of severe accidents	Plant control and protection architecture	
10			Operational expectations affecting safety	Radiation protection	
11			Fuel storage facility safety		
12			Management system		

TABLE 8. SUMMARY MAP OF THE KEs AND KTs (cont.)

KT Number	KE1 Site and environment	KE Fuel cycle	KE3 Nuclear safety	KE4 Nuclear island	KE5 Balance of plant and grid integration
1	Net thermal efficiency	Safeguards by design	Readiness of the SMR design	Owner/operator scope of supply	Capital costs
2	Compatibility with local use requirements	Special nuclear materials management	Licensing and/or certification status for the design	Supplier/technology holder issues	Operation and maintenance (O&M) costs
3	Protection against internal and external hazards	Physical protection of the NPP	Language	Project schedule capability	Fuel costs
4	Standardization of major components	Cybersecurity protection of the NPP		Technology transfer and technical support	Spent fuel management costs
5	Electrical power requirements			Project contracting options	Decommissioning costs
6	Demand following and storage capabilities			Services offered for the front end of fuel cycle (fresh fuel supply)	Financing
7	Maximum output capacity (heat equivalent and quality)			Services offered for the back end of fuel cycle (spent fuel management	
8	Integrated energy systems				

5.1. SITE AND ENVIRONMENT

KE1 considers site specific parameters with impact on the NPP design, which could differentiate among the technologies under consideration, based on the following eight KTs:

KT 1.1 Site seismicity;
KT 1.2 Meteorology and hydrology;
KT 1.3 Water resources;
KT 1.4 Population;
KT 1.5 Site access for construction and operation;
KT 1.6 Site size;
KT 1.7 Environmental and radiological impact;
KT 1.8 Other external events.

Importance rationale (Table 7): Interaction between site characteristics and the features of the NPP design may be a strong differentiator for a large capacity site with WCRs but of a medium importance for a smaller capacity site with one or a few SMRs, due to the lower land and water resources requirements. The potential need for a site to be located close to populated areas is also a consideration for this KE's importance.

Description: Site parameters are compared with the site parameters envelope offered by the technology holder for the proposed (standard) NPP design. The eight KTs are described as follows while examples are provided in the Annex.

This IAEA publication can provide further guidance on importance and weighting appropriate to this KE: IAEA Safety Standards Series No. SSG-35, Site Survey and Site Selection for Nuclear Installations [11].

KT 1.1 Site seismicity

Site seismicity (i.e. the range of expected magnitude and frequency of peak ground acceleration (PGA)), considers the safe shutdown earthquake ground motion and operating basis earthquake ground motion, per the IAEA Safety Standards Series No. SSG-9, Seismic Hazard in Site Evaluation for Nuclear Installations [12]:

> "Site area studies should include the entire area covered by the nuclear power plant, which is typically one square kilometre. The primary objective of these investigations is to obtain detailed knowledge of the potential for permanent ground displacement phenomena associated with earthquakes (e.g. fault capability, liquefaction, subsidence or collapse due to subsurface cavities) and to provide information on the static and dynamic properties of foundation materials (such as P-wave and S-wave velocities), to be used in site response analysis […]."

The site seismic level ground acceleration can differentiate the NPP technologies if designs are flexible or not to accommodate the site specific conditions. Usually, the comparison is based on the design values of safe shutdown earthquake ground motion (also called design basis earthquake (DBE)), and/or operating basis earthquake ground motion (also called operating basis earthquake (OBE)). Seismic induced tsunami is considered in KT 1.8.

The following questions may support a collection of information relevant to this KT:

— What are the design features assuring the seismic resistance of core support and fuel assemblies in maintaining their integrity under severe seismic related accidents?
— To what degree can the design be modified to allow for higher seismic loads?
— Based on the seismic design, does the evaluation include the effects of the soil–structure interaction on the soil sites, or only on the rock sites?
— Are the auxiliary and containment buildings designed to be built on a common basemat?
— Does the design include innovative seismic features, such as seismic isolation?
— What is the design margin ultimately necessary to prevent an early radioactive release or a large radioactive release in the event of levels of earthquake exceeding those considered for design (DBE, OBE), derived from the hazard evaluation for the site?

KT 1.2 Meteorology and hydrology

IAEA Safety Standards Series No. SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations, [13]:

> "[…] supplements and provides recommendations on meeting the requirements for nuclear installations established in the IAEA Safety Requirements publication on Site Evaluation for Nuclear Installations […] with regard to the assessment of meteorological and hydrological hazards. It thus complements other Safety Guides that deal with the protection of nuclear installations against external natural events and human induced events by means of site selection and site evaluation assessments and corresponding design features and site protection measures […]."

The importance of site meteorological conditions in the RTA are twofold:

(a) How the relevant local extreme meteorological conditions (e.g. wind rose specifically the most frequent wind directions, strong tornadoes, sandstorms, lightening and hail, thunderstorms, waterspouts, extremely high or low temperatures, ice, fog, relative humidity) differentiate the NPP technologies;
(b) How these extreme conditions influence differentiation among the NPP technologies with regard to atmospheric dispersion of radioactive releases during various NPP operational conditions. The assessment of NPP technologies in regard to atmospheric dispersion of radioactive materials is closely related to KE 1.4 and KE 1.7.

Similarly, the importance of site hydrological conditions in the RTA are twofold:

(a) How the extreme hydrological conditions (such as flood due to extreme rainfall, or immediate snow melt, or tsunami, and similar site specific conditions) differentiate the NPP technologies;
(b) How these extreme conditions influence differentiation among the NPP technologies with regard to emergency supply/use of water and dispersion of radioactive releases in water bodies.

The following questions may support a collection of information relevant to this KT:

— How does the design assure safe operation under highly elevated average air temperatures and high humidity under design basis hazards and under beyond design basis hazards (more severe but less frequent than design basis hazards)?
— How does the design assure safe operation under sandstorms in summer?
— How does the design assure safe operation in winter storms and under excessive ice accumulation in winter months?
— What allowance has been made for sea level rise during plant lifetime?

KT 1.3 Water resources

The IAEA publication on Efficient Water Management in Water Cooled Reactors, IAEA Nuclear Energy Series No. NP-T-2.6 (2012), describes the water management addressing the quantities of water needed for condenser cooling during operation, for construction (during the flushing phase), fire protection and for inventory control, including make-up to the primary coolant system and discharge from the radioactive liquid waste treatment system. This publication also provides an overview of the water usage in advanced NPP designs including the SMRs and is of practical use in the RTA.

The NPP technologies are assessed on the basis of water resources required for NPP make-up, blowdown and margin required for operation versus site specifics in this regard, as well as on condenser cooling water source and cooling water temperatures (for non-safety related cooling systems and safety related cooling systems, and ultimate heat sink (UHS) considerations).

The following questions may support a collection of information relevant to this KT:

— Can the design be adapted to the site specific environmental conditions such as ambient seawater salinity?
— Can the design use various water sources, including that from a wastewater treatment plant or a dedicated desalination plant?
— Can the design be adapted to cooling water of different temperatures?
— What are the design basis parameters related to UHS low water level conditions (e.g. due to low flow site, drought)?

KT 1.4 Population

The population distribution and population density within the site region may influence the assessment of NPP technologies by differentiating them due to a difference in the required exclusion zone size and in respect to the developed effective emergency response plan.

The potential changes in the severity and/or the frequency of natural external events, as well as changes in the population distribution in the site region, the present and future use of land and water, the further development of existing NPPs or the construction of other facilities that could affect the safety of the NPP or the feasibility of planning effective emergency response actions may differentiate the assessed NPP technologies.

KT 1.4 is related to KT 1.1, KT 1.2, KT 1.5 and KT 1.7. The following questions may support a collection of information relevant to this KT:

- How flexible is the design to adjust to the site specific population characteristics?
- How adjustable are the design provisions and safety margins allowing the implementation of the site specific emergency response plan?

KT 1.5 Site access for construction and operation

The adequacy of transportation routes is specifically important for transportation of the heaviest components. This aspect may differentiate the assessed NPP technologies; for example, the routes are not adequate to transport the heaviest components therefore there will be an economic impact in improving the transportation routes, or there is no possibility to improve the transportation conditions and that technology will score very low. The access to the required infrastructure for NPP construction and operation is site specific, and technologies may be differentiated only based on their specific requirements not found in other designs.

The following questions may support a collection of information relevant to this KT:

- What is the footprint of the major facilities on the site?
- What is the size and weight of the largest component, and how do you propose to transport it to the site?
- What are the transportation and on-site storage/laydown challenges expected for executing the modular construction plan?
- What other site specific issues could affect the site preparation schedule and costs?

KT 1.6 Site size

The NPP footprint requirement is compared to the site size; the technologies may differ, and this can represent the differentiator especially for SMRs. Equally important is the available site size and its possibility for expansion in the future to accommodate the construction of additional units.

The following questions may support a collection of information relevant to this KT:

- What is the range of acceptable plant ratings for this procurement and for this plant site?
- How flexible is the plant footprint in adjusting to the site size limits?

KT 1.7 Environmental and radiological impact

This KT addresses the impact of the NPP during its lifetime, including construction, operation and decommissioning on the site surrounding environment, such as air, water, soil, food chain and population health.

The differentiators among assessing NPP designs could be the potential impact of cooling towers to the environment (affecting local climate such as humidity, fog, ice, shadows) in comparison to the effect on nearby water bodies used as a heat sink, where the differentiators among assessing NPP designs are related to the exhaust water temperature and the effects on aquatic life.

Other differentiating aspects could be based on noise, release of hazardous or toxic chemicals, solid, gaseous and liquid operational wastes and radiological release in respect to normal and accidental conditions and their effects on the environment (air, water, soil, food chain, population).

The NPP designs may also be assessed on their effects on local industry and economy: how the designs differ in respect to promoting local industrial advancements, boosting the local economy, increasing and securing long term employment.

For a given site the differentiator among assessing NPP design is related to preparation of the site for construction based on the NPP size and overall layout: how the designs differ in respect to land preparation for construction.

For a given site the NPP overall visual imprint may be found challenging to the nearest population. The assessment of the NPP visual impact is based on the differences among the NPP designs in this respect, or in flexibilities of the NPP layout and design to accommodate the reduced visual disturbance in the site region.

The following publication may support the assessment: Managing Environmental Impact Assessment for Construction and Operation in New Nuclear Power Programmes, IAEA Nuclear Energy Series No. NG-T-3.11 [14].

The following questions may support a collection of information relevant to this KT:

— What is the impact of local temperature variation on plant performance and MWe output?
— What are the off-site release limits during normal operation?
— What type and how much operational waste is generated and how is it handled (e.g. on-site storage, treatment, need for specialized supporting industry)?
— What are the environmental effects during operation, including noise, chemical releases, operational wastes, radiological and thermal discharges?
— What are the effects on the site and its environs during preconstruction and construction activities?
— What would the environmental effects be during decommissioning, including noise, contamination and waste generation?

KT 1.8 Other external events

Some of the external events of potential consideration that may differentiate the NPP designs being assessed are described with this KT. It is site specific and the RTA team may modify and adjust this KT accordingly. The external events are usually grouped into those caused by natural events and those caused by human activities. External events related to seismicity, weather and water bodies are covered in KTs 1.1, 1.2 and 1.3, respectively.

The NPP technologies may differ due to (non-)existing flexibilities to incorporate additional design modifications in thus assuring safe operation under extreme conditions caused by external events, such as but not limited to accidental aircraft impact, nearby explosions and dispersions of toxic materials, volcanic hazards (ballistic projectiles, ash, lava), electromagnetic interference, forest fire or some of the following:

— Avalanche;
— Biological events;
— Coastal erosion;
— Electromagnetic interference;
— Externally generated missiles; industrial or military facility accident; military actions;
— Forest fire;
— Accidental aircraft impacts;
— Solar storms;

— Toxic gas;
— Transportation accidents;
— Volcanic activity.

The following are specific to integration with a hydrogen or chemical plant in a non-electric application:

— Release of hydrogen (risk of detonation);
— Release of oxygen;
— Leak of corrosive/toxic chemicals/material;
— Release of gases into a common ventilation system for a co-located cogeneration plant.

The supporting publication is Safety Reports Series No. 92, Consideration of External Hazards in Probabilistic Safety Assessment for Single Unit and Multi-unit Nuclear Power Plants [15].

The following questions may support a collection of information relevant to this KT:

— How does the design adapt to the site exposed to volcanic ash?
— What are the design modifications sustaining an avalanche?
— What are the protective measures against extended external fire?
— What is the adopted approach for accidental aircraft impacts?

5.2. FUEL CYCLE

KE2 considers the fuel cycle, which could differentiate among the technologies under consideration, based on the following nine KTs:

KT 2.1 Fuel materials and components;
KT 2.2 Fuel product supply chain;
KT 2.3 Fuel unit fabrication;
KT 2.4 Fuel operating experience;
KT 2.5 Refuelling outage;
KT 2.6 Fuel flexibility;
KT 2.7 Suitability to indigenous fuel fabrication;
KT 2.8 Medium term spent fuel storage capacity;
KT 2.9 Long term spent fuel storage.

Importance rationale (Table 7): In general, the importance may be considered medium because it is recognized that fuel costs are small in comparison to capital costs of a large NPP and SMRs but designs that are very small/micro carry their capital cost distribution differently, as a larger portion of their upfront capital is usually fuel related. Once the NPP is in operation, however, the performance of the fuel and the fuel cycle and the in-plant management of fuel have a major impact on NPP operation and operating costs. Therefore, the comparative offerings and technology holder experience regarding fuel and fuel cycle performance may be considered with high importance in some cases.

Description: All operations associated with the production of nuclear energy, including mining and processing of uranium or thorium ores; enrichment of uranium; manufacture of nuclear fuel; operation of nuclear reactors (including research reactors); reprocessing of spent fuel; all waste management activities (including decommissioning) relating to operations associated with the production of nuclear energy; and any related research and development activities. The front-end steps from mining to enrichment are not discussed because they are much less relevant with respect to RTA than the following ones.

The nine KTs are described as follows while examples are provided in the Annex.

KT 2.1 Fuel materials and components

This KT focuses on the availability of required fuel materials and components for solid fuel designs that may differentiate among the assessing technology designs. Liquid fuel designs, such as molten salt reactors, are not covered and would need a different set of KTs, which could be developed by the RTA team.

Fuel materials such as natural and enriched uranium, and mixed oxide (MOX) are available for large WCRs and SMRs. In the case of fuel based on enriched uranium, >5% and <20%, named in the nuclear industry as high assay low enriched uranium (HALEU) used in SMRs, potential suppliers would be less available. Enriched reprocessed uranium could become an option for countries with reprocessing and recycling capabilities of the spent fuel.

The supporting publication is IAEA Safety Standards Series No. NS-G-2.5, Core Management and Fuel Handling for Nuclear Power Plants [16].

The following questions may support a collection of information relevant to this KT:

— What is the level of fuel enrichment used in the reactor?
— What are the specifics in case of the use of HALEU? What are the constraints?
— Is accident tolerant fuel available for this design?
— Other than the fuel material, what other materials (e.g. alloys in the fuel assemblies, control rods, etc.) require special consideration in terms of supply and/or waste issues?

KT 2.2 Fuel product supply chain

This KT discusses the fuel supply chain. The importance of this KT differs between deployable large WCRs and innovative WCRs and SMRs and therefore will have different importance. A large supply chain is already available for fuel for large WCRs. On the contrary, in the case of innovative large WCRs and SMRs with innovative fuel, a new supply chain may be needed.

The fuel product supply chain consists of manufacturers of fuel assembly components (e.g. pellets, rods, grids, springs), the fuel assemblies, transporters and transport flasks. For large WCRs, a supply chain exists. The latter can be used for some SMRs. Transporters could be the same but with different casks depending on the fuel design.

The following questions may support a collection of information relevant to this KT:

— What is the duration and stability of the supply chain relationships?
— Has the fuel operated successfully, if so, for how many total reactor-years?
— What agreements exist with fuel product suppliers (e.g. enriched UF_6, triso fuel, …)?
— What is the availability of alternate fuel and materials suppliers?
— Will the fuel supply chain be ready when the designs are ready to come to market?
— Does the contract include several mines or plants for the supply of materials and components?
— How can security of supply be obtained or achieved? Quantity of stocks? Price of stocks? Number of suppliers for these materials in the world?
— What is the stock level for the fuel product?

KT 2.3 Fuel unit fabrication

This KT discusses the demonstrated ability of the technology holder to manufacture fuel units (e.g. assemblies, bundles, TRISO fuel) and how this may differentiate the designs under assessment.

Fuel assemblies for large WCRs and SMRs can be produced in the same fuel plant along with the applicable regulations and equipment used. For large WCRs, contracts are possible with different technology holders, while initially the same experience may not be available for SMRs.

The following questions may support a collection of information relevant to this KT:

- Which fuel suppliers have fabricated fuel for this reactor design type?
- How many suppliers (with how many fuel plants) can provide fuel assemblies for this reactor?
- Can an existing fuel plant that produces fuel assemblies for large WCRs also produce fuel assemblies for the SMRs?

KT 2.4 Fuel operating experience

This KT addresses the vendor's operating experience for the nuclear fuel materials, fabrication, operational expectations and experience.

Operators and vendors have decades of fuel operating experience regarding the large WCRs. This is not the case for SMRs. Because of the differences in reactor core size, loading plans are different for large WCRs and SMRs. In the case of a similar fuel assembly design, the experience gained from large WCRs can be used for SMRs, but because of different assemblies' sizes in SMRs and large WCRs, the flow and heat transfer parameters are not the same, requiring at least thermohydraulic qualification tests.

The following questions may support a collection of information relevant to this KT:

- What is the fuel design type and what is the industry experience in other NPPs with this fuel type and design?
- Are there any fuel design changes that are being proposed for this reactor design that have not been demonstrated through industry experience?
- Are there available computer codes and core design methodologies in order to optimize a core loading plan and safety evaluation for fuel reloading?
- Are the examination tools for large WCRs usable for SMRs?

KT 2.5 Refuelling outage

This KT addresses all aspects of fuel refuelling based on which the technology holders are to be assessed.

Outages are necessary for refuelling but also for maintenance operations. Having the possibility to choose the outage date allows for adaptations to forecast in the changes of electricity demand. SMRs are often typically designed for longer fuel cycle length. Some SMR designs use HALEU, and consequently, less outages are necessary. Because of the difference in power between large WCRs and SMRs, the duration of outages has more impact on the operation of large WCRs.

In SMRs, the number of fuel assemblies is less than in large WCRs, which reduces the duration of outages. But in the case of integral reactors, the preparation time is longer because equipment has to be removed.

The SMRs fuel assemblies can be shorter compared to large WCRs, therefore the risk of being damaged during refuelling operations is reduced. Because of their shorter size, the fuel assemblies in SMRs can be handled vertically all the way to or from the spent fuel pool or short term storage without the need of tilting the fuels' assemblies into a transfer tube, as in pressurized water cooled reactors (PWRs).

Internal components within the vessel of SMRs are supposed to be easily removed from the vessel and replaced by refurbished ones. The removed components can then be subsequently inspected, repaired and maintained outside of the outage critical path, as in PWRs.

The following questions may support a collection of information relevant to this KT:

- What is the proposed fuel cycle length? What is the potential for fuel cycle length extension?
- What are the expected refuelling batch sizes and enrichments to achieve the proposed fuel cycle length?
- What is the capacity factor during operation?

— What is the expected fuel cycle outage duration?
— What assumptions are made for fuel movement and manoeuvrability in the fuel outage?
— Is full core discharge assumed in the refuelling plan?
— What is the industry experience base for refuelling cycle lengths and refuelling outages for this facility design?
— What is the number of fuel assemblies in the reactor?
— Can training be offered for discharge and refuelling operations?
— What are the transport and handling operations due to fuel reloading?
— Do the refuelling outages represent enough opportunity for most maintenance operations?
— Is it possible to detect damaged assemblies during the outage?
— Is it possible to examine all the fuel assemblies during the outage?

KT 2.6 Fuel flexibility

This KT addresses the level of flexibility of NPP operation with respect to different fuel types, including higher uranium enrichment levels, HALEU or MOX fuel and availability and competitiveness of different fuel materials and components for NPP design. Fuel flexibility could also mean flexibility in the use of different fuel enrichments.

The following questions may support a collection of information relevant to this KT:

— Does the design allow for flexibility with different fuels, including higher uranium enrichment, MOX, thorium fuel?
— What are the assembly average and peak rod fuel burnup values?
— What are the design and licensing features that limit fuel burnup?
— What part of MOX — 20%, 50%, 100% — can be used in the reactor?

KT 2.7 Suitability to indigenous fuel fabrication

Fuel fabrication for large WCRs is an oligopolistic market. Fuel fabrication plants for very specific fuel assemblies for SMRs do not exist.

The following questions may support a collection of information relevant to this KT:

— Is it possible to have a transfer of technology for a fuel production plant?
— What components of fuel assemblies can be produced locally?
— What are the duration and the budget for building a fuel fabrication plant?

KT 2.8 Medium term spent fuel storage capacity

This KT describes how technology holders differentiate regarding solutions for the medium term spent fuel storage capacity and potential for its increase.

Wet and dry fuel storage solutions are available for large WCRs. Similar solutions could be used for SMRs that have a similar fuel design. For more innovative SMR fuels, new storage designs may need to be developed.

The following questions may support a collection of information relevant to this KT:

— How many spent fuel assemblies can be stored in the spent fuel pool?
— For how many years of operation does the plant have a storage capacity of spent fuel assemblies?
— Is it possible to increase the storage capacity of a spent fuel pool in the future?
— What are the operations during the transport of spent fuel from the reactor spent fuel pool away from reactor storage?
— For how long can the spent fuel assemblies be stored in the selected spent fuel storage system?

— Which type of inspection of the spent fuel storage system is needed and at what frequency is the inspection planned/required?

KT 2.9 Long term spent fuel storage

This KT describes how technology holders differentiate regarding solutions for the long term spent fuel storage capacity and potential for increases in the future. Alternatively, fuel take-back is an option to consider here. IAEA Nuclear Energy Series No. NF-T-3.3, Storing Spent Fuel until Transport to Reprocessing or Disposal [17], describes the requirements for long term spent nuclear fuel (SNF) storage configurations:

> "Spent fuel storage configurations can accommodate uncertain storage periods, to facilitate ageing management and to provide flexibility for future steps to achieve an acceptable end point. Key decisions in selecting a spent fuel storage configuration to meet present and future needs relate to how spent fuel will be stored, whether spent fuel will be packaged for transport or disposal prior to or after storage, what components will be relied upon to perform essential safety functions, and how safety performance will be demonstrated with sufficient certainty to satisfy regulatory requirements. Each decision affects future options. Available alternatives are evaluated to select a strategy that can be sustained over extended storage periods while maintaining flexibility and adaptability to accommodate the full range of plausible future scenarios.
>
> Geological repositories will require spent fuel to be placed in suitable disposal containers before emplacement underground. For many States pursuing repositories, the design (e.g. capacity and material specifications) for the disposal container and the acceptance criteria for the contained waste form are not yet settled. This has significant implications for spent fuel storage on how and when spent fuel is placed into containers."

Large WCRs and SMRs spent fuel can be stored in different systems at the same site and/or in the same deep geological disposals.

The following questions may support a collection of information relevant to this KT:

— Does the process and infrastructure/equipment exist for transferring spent fuel from one storage system to another?
— What are the necessary examinations of a long term spent fuel storage?
— When is it possible to have dry storage for spent fuel?
— After how many years is it possible to transfer fuel to a deep geological repository?
— Is it possible to reprocess spent fuel?

5.3. NUCLEAR SAFETY

KE3 considers NPP operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation hazards, which could differentiate among the technologies under consideration, based on the following 12 KTs:

KT 3.1 Implementation of DiD philosophy;
KT 3.2 Safety design philosophy;
KT 3.3 Degree of diversity and redundancy;
KT 3.4 Protection against internal and external hazards;
KT 3.5 Response to off-site power loss;

KT 3.6 Completeness of OLCs, SAR, PSA, O&EPs, SAMGs;
KT 3.7 Results of deterministic safety analysis;
KT 3.8 Results of probabilistic safety assessment;
KT 3.9 Mitigation of severe accidents;
KT 3.10 Operational expectations affecting safety;
KT 3.11 Fuel storage facility safety;
KT 3.12 Management system.

Importance rationale (Table 7): Nuclear safety is expected to be included at the policy objectives level and the highest KE contribution level. It has the potential to be a strong differentiator. Gathering consistent and accurate information from technology holders for appropriate comparisons is required. A wide variety of metrics is available, yet the approach is to select a reasonable number that will appropriately represent the capability of the NPP with respect to nuclear safety.

Description: The achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation hazards [18]. According to INSAG Series No. 12 [18], the DiD concept is a fundamental safety principle that provides an overall strategy for safety measures and features of NPPs. When properly implemented in the design, this concept ensures the fulfilment of all three fundamental safety functions as they are defined in IAEA Safety Standards Series No. SSR-2/1, Safety of Nuclear Power Plants: Design (Rev. 1) [19] as follows:

— Control of reactivity;
— Removal of heat from the reactor and from the fuel in storage;
— Confinement of radioactive materials, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.

SSR-2/1 (Rev. 1) [19], para. 2.12, further states:

> "The primary means of preventing accidents in a nuclear power plant and mitigating the consequences of accidents if they do occur is the application of the concept of defence in depth […]. This concept is applied to all safety related activities, whether organizational, behavioural or design related, and whether in full power, low power or various shutdown states. This is to ensure that all safety related activities are subject to independent layers of provisions so that if a failure were to occur, it would be detected and compensated for or corrected by appropriate measures. Application of the concept of defence in depth throughout design and operation provides protection against anticipated operational occurrences and accidents, including those resulting from equipment failure or human induced events within the plant, and against consequences of events that originate outside the plant."

The challenge is to examine those functions and technology features that will best differentiate the overall safety level of the assessed designs, based on the differentiation in each KT, for example, between weak differentiators like DiD and strong differentiators like deterministic safety analysis (DSA) and probabilistic safety assessment (PSA). This will be dependent on the spectrum of reactor designs under consideration. Since nuclear safety is a policy objective, it is necessary to conduct the assessment so that the approach and the results can be clearly explained to policy makers.

The following IAEA documents can provide further guidance on the importance and appropriateness of this KE at a high level, while a description of some of the KTs also include a list of more specific supporting publications:

— IAEA Safety Standards Series No. SF-1 Fundamental Safety Principles [20];
— IAEA Safety Standards Series No. SSR-2/1 (Rev.1) Safety of Nuclear Power Plants: Design [19];

— IAEA Safety Standards Series No. SSR-2/2 (Rev.2) Safety of Nuclear Power Plants: Commissioning and Operation [21];
— Defence in Depth in Nuclear Safety, INSAG Series No. 10 [22];
— Basic Safety Principles for Nuclear Power Plants, INSAG Series No. 12 [18];
— IAEA Safety Standards Series No. SSG-54 Accident Management Programmes for Nuclear Power Plants [23].

The 12 KTs are described as follows and examples are provided in the Annex. While most KTs refer to generally applicable topics, liquid fuel designs, such as molten salt reactors, are not covered.

KT 3.1 Implementation of defence in depth philosophy

This KT supports the assessment of the NPP designs regarding the philosophy followed in achieving the DiD in the design. The DiD concept is applied to all safety related activities, whether organizational, behavioural or design related, and whether in NPP full power, low power or various shutdown states. This is to ensure that all safety related activities are subject to independent layers of provisions so that if a failure were to occur, it would be detected and compensated for or corrected by appropriate measures.

IAEA Safety Standards Series No. SSR-2/1 (Rev.1), Safety of Nuclear Power Plants: Design [19], Paragraph 2.13, identifies five levels of DiD that have to be provided for an NPP. It is accepted that each level of DiD may fail. Failure of the defences at one level will ideally be intercepted and mitigated by the following level. The levels of defence will ideally be independent to the extent practicable. The levels are described as follows [19]:

"(1) The purpose of the first level of defence is to prevent deviations from normal operation and the failure of items important to safety. This leads to requirements that the plant be soundly and conservatively sited, designed, constructed, maintained and operated in accordance with quality management and appropriate and proven engineering practices [...].
(2) The purpose of the second level of defence is to detect and control deviations from normal operational states in order to prevent anticipated operational occurrences at the plant from escalating to accident conditions [...].
(3) For the third level of defence, it is assumed that, although very unlikely, the escalation of certain anticipated operational occurrences or postulated initiating events might not be controlled at a preceding level and that an accident could develop. In the design of the plant, such accidents are postulated to occur [...].
(4) The purpose of the fourth level of defence is to mitigate the consequences of accidents that result from failure of the third level of defence in depth [...]."

The most important objective for this level is to ensure the confinement function, thus ensuring that radioactive releases are kept as low as reasonably achievable. Level 5 is described as follows [19]:

"(5) The purpose of the fifth and final level of defence is to mitigate the radiological consequences of radioactive releases that could potentially result from accident conditions. This requires the provision of an adequately equipped emergency response facilities and emergency plans and emergency procedures for on-site and off-site emergency response."

The provision in the design of a series of physical barriers as well as a combination of safety features provided to meet the requirements for DiD at each level will ideally be independent to the extent practicable. Note that many of the organizational or procedural provisions for DiD are not discussed here and are only touched on in KT 3.12 Management system.

IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [20] establishes the fundamental safety objective, safety principles and concepts that provide the bases for the IAEA's safety standards and its safety related programme, while SSR-2/1 (Rev. 1) [19] was written specifically for WCRs and was mainly aimed at large power reactors. The requirements for DiD may not all apply directly to reactors with other technologies and SMRs. For instance, the core melt phenomenon does not apply to some advanced designs. If this can be demonstrated with high confidence, then fewer levels of DiD may be needed for SMRs or a graded approach for DiD implementation could be adopted, that is, the safety design and analysis requirements will ideally be commensurate with the associated radiological risk and their subsequent effects on the population and the environment.

Supporting publications:

— Safety Reports Series No. 46, Assessment of Defence in Depth for Nuclear Power Plants [24] provides specific technical information on implementation of the DiD concept in the siting, design, construction and operation of NPPs and describes a method for verifying capabilities for implementation of DiD in existing NPPs. This publication provides a very comprehensive assessment methodology for DiD but is complex and is not advisable for the early stages of an RTA since many of the specific safety principles cannot be tested.
— IAEA Safety Standards Series No. SSR-2/1 (Rev.1), Safety of Nuclear Power Plants: Design [19] states that the "[r]equirements for nuclear safety are intended to ensure 'the highest standards of safety that can reasonably be achieved' for the protection of workers, the public and the environment from harmful effects of ionizing radiation that could arise from NPPs and other nuclear facilities[…]" and discusses the safety in design and the concept of the DiD.
— IAEA Nuclear Energy Series No. NP-T-2.2, Design Features to Achieve Defence in Depth in Small and Medium Sized Reactors [25] includes information on "the state of the art in design approaches used to achieve defence in depth in pressurized water reactors, pressurized light water cooled heavy water moderated reactors, high temperature gas cooled reactors, sodium cooled and lead cooled fast reactors, and non-conventional designs within the SMR range". Particular attention is given to those approaches that eliminate accident initiators or prevent accident consequences by incorporating inherent and passive safety features and passive systems into the safety design concepts of such reactors.
— Western European Regulators Association (WENRA) Reactor Safety Reference Levels [26] discusses DiD and adopts the IAEA SSR-2/1 approach.
— European Utility Requirements Revision E [7] adopts a different approach to IAEA in subdividing levels of DiD.

The following questions may support a collection of information relevant to this KT:

— How does the safety analysis of the NPP design demonstrate the correct implementation of the DiD concept, that is, performance of the structures, components, operating systems and safety systems that provide multiple barriers for preventing releases to the environment during normal operation, anticipated events and postulated accidents?
— What is the overall approach for incorporating the DiD concept in the design?
— What are the approach and philosophy followed to achieve as far as practicable independence between DiD levels in the design, including assurance of containment integrity?
— What are the general safety design criteria with regard to emergency core cooling system capacity, single failure/redundancy, diversity, separation and independence as it relates to reactor trip and emergency cooling?
— What requirements are incorporated into the design of the UHS? What assumptions are made with regard to assurance of UHS availability? What systems are required to function to assure heat transfer to the UHS? Under what conditions can you avoid core and fuel pool damage with sustained loss of UHS?

KT 3.2 Safety design philosophy

This KT differentiates among the candidate designs for provisions of combinations of passive, active and inherent safety features. Its importance may be different for large WCRs (evolutionary versus innovative) and different for SMR designs.

An active component is defined in the IAEA Safety Glossary [3] as "[a] *component* whose functioning depends on an external input such as actuation, mechanical movement or supply of power." Examples of active components are pumps, fans, relays and transistors.

A passive component is defined in Ref. [3] as "[a] *component* whose functioning does not depend on an external input such as actuation, mechanical movement or supply of power." Examples of passive components are heat exchangers, pipes, vessels, electrical cables and structures.

Certain components, such as rupture discs, check valves, safety valves, injectors and some solid state electronic devices, have characteristics that require special consideration before designation as an active component or a passive component.

Passive components typically rely on smaller driving forces than active components, for example, natural circulation coolant flow driven by density differences rather than forced circulation driven by pumps or gravity insertion of control rods rather than driven by motors, air or water pressure. This renders passive components more vulnerable to small opposing forces, such as gas locking of liquid filled cooling loops or friction in gravity driven components. However, the fundamental driving forces of passive systems, such as gravity, are not subject to failure. This makes them inherently reliable.

Safety standards and international regulatory agencies do not normally indicate a requirement or even preference for active or passive components or systems. The adequacy of safety provisions of the design is normally evaluated by means of radiation protection programmes, DSA and PSA. Examples of high level safety programmes and criteria are:

— Radiation protection programme, which is used to ensure that public and worker annual dose limits are met;
— DSA, which is used to ensure that dose limits for anticipated operational occurrences, design basis accidents and (if applicable) design extension conditions are met;
— PSA, which is used to establish that a balanced design has been achieved, to provide assurance that 'cliff edge' effects are prevented and to ensure that the established safety goals for core melt frequency and large early release frequency are met.

In a number of designs of NPP, systems that use passive components may employ less redundancy than equivalent systems that use active components. The design will ideally take due account of the failure of a passive component unless it can be demonstrated that the failure of such component is very unlikely. Provided the target system reliability can be achieved, systems with active or passive components can potentially meet the safety criteria. The PSA is the key tool for assessing the adequacy of the overall design provisions.

SMRs typically make greater use of passive systems and components. Their smaller size and lower power rating allow designers to achieve design objectives without the need for active components: for example, decay heat removal by natural circulation and dissipation to the atmosphere.

SSR-2/1 (Rev. 1) [19] states that:

> "Requirements for nuclear safety are intended to ensure the 'highest standards of safety that can reasonably be achieved' for the protection of workers, the public and the environment from harmful effects of ionizing radiation that could arise from nuclear power plants and other nuclear facilities […]."

SSR-2/1 (Rev. 1) [19] also discusses safety in the design. Reference [25] provides a description of design approaches regarding:

> "[…] the elimination of accident initiators/prevention of accident consequences through design and incorporation of inherent and passive safety features and passive systems into safety design concepts of such reactors."

Supporting publications:

- IAEA General Safety Guide GSG-7 Occupational Radiation Protection [27];
- IAEA General Safety Guide GSG-8 Radiation Protection of the Public and the Environment [28];
- IAEA Specific Safety Guide SSG-2 (Rev.1) Deterministic Safety Analysis [29];
- IAEA Safety Standards Series No. GS-G-4.1, Format and Content of the Safety Analysis Report for Nuclear Power Plants [30];
- IAEA Specific Safety Guide SSG-3 Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [31];
- IAEA Specific Safety Guide SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [32].

The following questions may support a collection of information relevant to this KT:

- What has been the approach to optimize the incorporation of inherent features, passive and/or active safety systems into the design?
- What is the design approach (including conceptual and engineering design, operational procedure and training) to reduce or alleviate human errors?
- Does the design approach consider practical elimination of accidents leading to early or large radioactive releases, that is, demonstration that such accidents are impossible or due to very dissimilar conditions (e.g. reactor pressure vessel) and with what justification?

KT 3.3 Degree of diversity and redundancy

This KT differentiates among the assessed designs by the degree of diversity and redundancy in safety systems. Diversity means that the same function can be achieved by means of different design, while redundancy refers to the number of similar systems or components able to perform the safety function. Both terms imply independence.

The events and lessons learned as a result of the accident at the Fukushima Daiichi nuclear power plant have resulted in a heightened appreciation of the need for and value of diverse water systems, alternative cooling systems, electrical systems and steam driven cooling systems. It is important that new NPP offerings include both first line and secondary safety systems with these diverse characteristics. In addition, whenever valued risk reduction benefits can be demonstrated, diverse features will ideally be coupled with redundancy of equipment components or systems. These include power supply and water delivery systems.

Redundancy provides a measure of protection against random component failure. Diversity provides a measure of protection to reduce the possibility of common mode failures between redundant components of safety systems or across systems. Application of both concepts and careful use of fail-safe design are important in ensuring system reliability. For some passive systems belonging to several DiD levels (e.g. reactor vessel), redundancy could not be economically feasible therefore diversity is not of primary concern.

PSA is the main tool for verifying and assessing the adequacy of redundancy and diversity. To ensure reliable results from the PSA it is important that adequate accident sequence, human error and system failure modelling approaches are used and well founded initiating event frequencies, failure rates for components and human reliability data are available. In order to support the design and verification of regulatory safety goals, a PSA will ideally be performed involving a comprehensive list of initiating events and hazards and all NPP operating modes (i.e. including at power and shutdown modes) unless a limited scope is specified.

The use of PSA for assessing redundancy and diversity may differ between the large WCRs and SMRs as the levels of maturity of the PSA modelling approach and available data are different.

Redundancy and diversity play an equally important role in both large WCRs and SMRs, though designs that make greater use of passive components may be able to demonstrate less redundancy is needed to achieve the required system reliability, provided the functional reliability of passive systems is satisfactorily assessed. The results of PSA are discussed in more detail in KT 3.8.

SSR-2/1 (Rev. 1) [19] may support further assessment, specifically it states the following: "Requirements for nuclear safety are intended to ensure 'the highest standards of safety that can reasonably be achieved' for the protection of workers, the public and the environment from harmful effects of ionizing radiation that could arise from nuclear power plants and other nuclear facilities […]" and further discusses diversity and systems reliabilities.

The following questions may support a collection of information relevant to this KT:

— What is the approach to incorporate redundancy, train separation and diversity of sensors or other components for initiating safety functions (e.g. reactor trip, emergency cooling) into the plant design?
— Is a diverse UHS available?
— To support justification of redundancy and diversity design requirements, how are the initiating event frequency, failure rate data for components and human actions derived for the PSA?

KT 3.4 Protection against internal and external hazards

This KT is connected to KT 1.8:

> ***KT 1.8 External events***
>
> Some of the external events of potential consideration that may differentiate the assessing NPP designs are described with this KT. It is site specific and the RTA team may modify and adjust this KT accordingly. The external events are usually grouped into those caused by natural events and those caused by human activities. External events related to seismicity, weather and water bodies are covered in KTs 1.1, 1.2 and 1.3, respectively.
>
> The NPP technologies may differ due to (non-)existing flexibilities to incorporate additional design modifications in thus assuring safe operation under extreme conditions caused by external events, such as but not limited to accidental aircraft impact, nearby explosions and dispersions of toxic materials, volcanic hazard (ballistic projectiles, ash, lava), electromagnetic interference, forest fires.

These two KTs are not to be addressed in the same way in the assessment, rather KT 1.8 considers to what degree the design is adaptable to the extreme events at the site while KT 3.4 addresses which external events are already incorporated within the design taking into account the design criteria and design margins, and if multi-effects are considered in the design and the PSA of the NPP.

Internal and external hazards for the proposed site or sites are identified or enveloped by a systematic process. Events that cannot be screened out by extreme low frequency are to be addressed by the design. At the early RTA stages, detailed information may not be available and only generic assumptions will be available for the candidate designs. At later stages more information becomes available, and better differentiation among the designs is possible.

Structures, systems and components are designed to withstand loads arising from the identified internal and external hazards, or sufficient redundancy and separation are to be included in the design to ensure that safety functions can be met. For example:

— External seismic loads may be addressed by robust design that ensure a high confidence of a low probability of failure of important structures, systems and components;
— Internal fires may be addressed by provision of adequate fire barriers between trains of safety systems and provision of sufficient redundant trains to ensure that the safety function of the system can be achieved.

Operating and emergency procedures are to include the necessary responses to internal and external hazards.

External hazards are site dependent and are the same for large WCRs and SMRs. However, the way the event loads affect the NPP depends on the NPP design and its layout. For some non-electric applications, depending on the coupling of the non-electric BOP to the nuclear steam supply system (NSSS), the BOP itself could be considered as an external hazard to the NSSS. For example, when coupling an NPP to a hydrogen plant, the reciprocal effects in the case of accident in either plant may affect the other one. In the case of using cogeneration for desalination, the loss of the desalination plant might pose a safety issue for the NPP as loss of load.

Internal hazards depend directly on the NPP design, layout and choice of coolant. For example, internal fires may present a greater challenge in sodium cooled plants, whereas internal flooding may be a bigger problem in water cooled plants.

Deterministic hazard analyses complemented by modern probabilistic tools such as seismic PSA or seismic margin assessment, internal fire PSA, internal flooding PSA and adherence to appropriate design and construction standards can quantify the risk and ensure a consistent coverage of internal and external hazards for designing the layout of the NPP, specifying the locations and loads of safety related items with adequate margins and defining specific hazard protection features.

The supporting IAEA publication is External Events Excluding Earthquakes in the Design of Nuclear Installations [33].

The following questions may support a collection of information relevant to this KT:

— Which internal and external events are considered in the design of the nuclear island and what are the associated design criteria and margins?
— How have severe internal and external events or multiple external/internal events been considered in the design and operating procedures?
— Have multi-unit effects due to hazards been considered in the design and PSA of the plant site?
— Has a consistent return frequency been used for different types of internal and external events?
— Does the plant PSA include contributions from internal and external hazards?
— Do the operating and emergency procedures include consideration of internal and external hazards?
— How will the emergency plan for off-site protective measures take account of external hazards?

KT 3.5 Response to off-site power loss

This KT may be analysed based on the IAEA Safety Standards Series No. SSG-34, Design of Electrical Power Systems for Nuclear Power Plants [34] that "[…] provides recommendations on the necessary characteristics of electrical power systems for nuclear power plants and of the processes for developing these systems, in order to meet the safety requirements of SSR-2/1 (Rev. 1) [19]". The publication describes the importance of on-site and off-site power systems and their importance for safe operation of the NPPs. Based on this publication the RTA team may decide the importance of this KT and use it to assess the designs on their responses to off-site power loss.

NPPs differ widely in their vulnerability to off-site power loss. Large WCRs and SMRs may employ active or passive systems (or a mixture) to remove heat from the reactor to the heat sink. For all designs, it is important to ensure that there are no external hazards that can simultaneously cause a loss of off-site power and compromise the systems needed to remove core decay heat. Large WCRs are more likely to include active systems in the response to loss of off-site power than SMRs, but the same principles apply. Heat removal to the heat sink and essential containment support are expected to be possible for all NPPs. Acceptable durations of interruptions to these safety functions are expected to be quantified.

IAEA Nuclear Energy Series NG-T-3.8, Electric Grid Reliability and Interface with Nuclear Power Plants [35] may support further assessment.

The following questions may support a collection of information relevant to this KT:

— How long can the site operate without external supply of water or diesel fuel under loss of off-site power?
— What are the capabilities of a steam driven safety system (if any), emergency power supplies (including DC power) and diverse or redundant safety systems (including electric and non-electric cooling systems) in the event of loss of AC power?
— How long can the plant sustain a loss of all AC and DC power before core damage occurs?
— What is the capability of safety systems in terms of the grace period for operator action and available off-site power during a station blackout (loss of all AC power and emergency diesel generators) and for other major events?

KT 3.6 Completeness of OLCs, SAR, PSA, O&EPs, SAMGs

In this KT we look at some of the key documents selected because they are likely to be available early in the RTA process (at least in partial or outline form) and can be strong discriminators between designs. Key documents and analyses are refined throughout the design process of an NPP so that the operational limits and conditions (OLCs), safety analysis report (SAR), PSA and operating and emergency procedures (O&EPs) are fully consistent. These documents, analyses and procedures are also modified during a new build programme to conform with Member States' standards and regulatory requirements. Before proceeding to operation, all these documents are expected to be consistent with the final design.

It is possible that a candidate design may have prepared the key documents in accordance with the regulations and standards of another Member State. Such documents are valuable in judging the level of completeness of these documents.

Supporting publications:

— IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [36];
— IAEA Safety Standards Series No. GS-G-3.1, Application of the Management System for Facilities and Activities [37];
— IAEA Safety Standards Series No. GS-G-3.5, The Management System for Nuclear Installations [38];
— IAEA Safety Standards Series No. SSG-51, Human Factors Engineering in the Design of Nuclear Power Plants [39];
— IAEA Safety Standards Series No. NS-G-2.2, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants [40];
— IAEA Safety Standards Series No. GS-G-4.1, Format and Content of the Safety Analysis Report for Nuclear Power Plants [30].

An SAR is based on accepted standards or guides such as IAEA Safety Standards Series No. SSG-61, Format and Content of the Safety Analysis Report [30], and is usually modified to meet the owner/licensee's requirements as well. However, accepted SARs vary between Member States. Here it is assumed as a minimum that the SAR includes an overview of the NPP site; the NPP structures and layout;

the reactor and its associated systems; and the results of the DSA. It may also include the OLCs and the results (or a summary) of the PSA.

The SAR may be developed in stages as design and licensing proceed. Typical stages may include a preliminary SAR, a pre-construction SAR and a final SAR. For an RTA, the SAR is likely to be a preliminary SAR, though later stages may be available from projects in other Member States. Acceptance of an SAR by another Member State is a strong positive indicator.

Results of the DSA are discussed in more detail in KT 3.7.

OLCs, called technical specifications in some Member States, are the set of rules setting forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of an authorized facility (IAEA Safety Glossary 2018 [3]).

The OLCs are written to ensure that the NPP is operated within the limits specified in the design and safety analysis. Changes to the OLCs, the NPP design and safety analysis are to be synchronized to ensure that each supports the other. This is usually ensured by the management system and control procedures.

The management system is discussed in more detail in KT 3.12.

The PSA is a major tool used to assess the level of safety achieved by an NPP design. PSA computes the frequency of a particular endpoint being reached by summing the frequency of all identifiable routes to that endpoint. The computed endpoint depends on the level of PSA: core melting (Level 1 PSA), release of radioactive materials (Level 2 PSA). The paths to those endpoints include a comprehensive list of initiating events, consequential and other random failures or events and human errors. The frequencies of initiating events and the probabilities of component failures and operator errors are key inputs to the PSA.

DSA and PSA are complementary. DSA focuses on the acceptable consequences of a bounding set of postulated initiating events. PSA focuses on the frequencies of the unacceptable conditions occurring after, as far as possible, a comprehensive list of initiating events. DSA can use the initiating event frequencies from the PSA to identify and classify postulated initiating events. PSA uses the criteria and consequences from the DSA to classify event endpoints (acceptable or not acceptable conditions).

The results of PSA are discussed in more detail in KT 3.8.

O&EPs including severe accident management guidelines (SAMGs): Procedures are prepared together with the design and safety analysis to ensure that operator feedback is included and that there is sufficient time to train new operators prior to commissioning.

The procedures are typically divided into operating procedures, emergency procedures, severe accident management guidelines and an emergency plan. They are integrated to ensure that there are clear transitions from one set of procedures to another as the NPP state changes.

Requirements for all the above key documents and analyses are effectively the same for large WCRs and SMRs.

The following questions may support a collection of information relevant to this KT:

— Are the OLCs complete and are they provisional or final?
— Is the DSA complete?
— Does the PSA cover at power and shutdown states?
— Does the PSA consider internal and external hazards?
— Does the PSA cover Level 1 and Level 2?
— Are the operating procedures (normal, emergency and severe accident) complete?
— Does the design comply with regulations on radiation and safety related to NPP siting in the Member State?
— What regulatory approvals have been given for the design by regulatory bodies in other Member States?
— Given the Member State regulatory basis, what emergency plan does this design require?
— What regulatory review and approval has been performed for the emergency plan and severe accident management programme?

KT 3.7 Results of deterministic safety analysis

This KT differentiates the candidate designs addressing which deterministic methods are used to verify that the safety requirements for DSA are met and the DiD concept has been properly implemented.

Supporting publications:

— IAEA Safety Standards Series No. GSR Part 4 (Rev.1), Safety Assessment for Facilities and Activities [42], provides high level requirements for safety assessment of nuclear facilities, including large WCRs and SMRs.
— IAEA Safety Standards Series No. SSG-2, Deterministic Safety Analysis for Nuclear Power Plants [29], provides recommendations and guidance on deterministic safety analysis for designers, operators, regulators and technical support organizations. It also provides recommendations on the use of DSA in:
  - Demonstrating or assessing compliance with regulatory requirements;
  - Identifying possible enhancements of safety and reliability;
  - Obtaining increased operational flexibility within safety limits for NPPs.

The recommendations are based on the current good practices at NPPs around the world and derive mainly from experience in performing transient analyses and accident analyses for NPPs.

Margins to DSA acceptance criteria are an important factor in differentiating between designs as large margins indicate a robust safety case that will be resilient throughout the NPP life; for example, unexpected ageing effects or new discoveries will be less likely to lead to a reduction in output, to additional maintenance, or to unanticipated major component replacement. It is to be noted that safety margins quoted for candidate designs using or based on proven technology are likely to be more reliable than those for unproven designs.

Large WCRs and SMRs have similarly high level DSA safety criteria, usually expressed as public dose limits for anticipated operational occurrences and design basis accidents. These limits may be expressed on a per reactor or per site basis. In addition to the public dose limits, there are lower level criteria that also have to be met. These typically relate to protection of the barriers to release radioactive material.

Large WCRs have many similarities across designs and hence the low level criteria are often similar. Examples include avoidance of fuel melting, avoidance of fuel clad dryout, avoidance of fuel sheath failure, limitation of primary and secondary coolant system overpressure, integrity of containment.

SMRs have more diverse designs, and while the high level dose criteria are likely to be the same as for large WCRs, the low level criteria are expected to be very different. Note that few Member States' regulatory bodies have articulated low level DSA safety criteria for SMRs.

— Molten salt reactors will not have criteria related to the fuel matrix and fuel clad (there is no fuel clad and the fuel is molten). The absence of these barriers to fission product release simplifies the DSA, but places greater emphasis on the remaining barriers; the primary coolant boundary and the containment boundary.
— Gas cooled reactors typically use ceramic fuel and are not vulnerable to fuel clad dryout. Absence of a volatile liquid coolant (such as pressurized water) leads to simpler modelling of postulated accidents and can bring greater confidence to predictions.
— Sodium cooled reactors typically have large margins to coolant boiling due to the low pressure and high boiling point of the coolant. However, the possibility of a large positive void reactivity coefficient and short effective neutron lifetime can result in very energetic core disruptive accidents if a void is introduced to the core, or the core geometry is compressed. Such accidents are to be eliminated with high confidence.

The following questions may support a collection of information relevant to this KT:

- What are the safety performance features of the primary reactor coolant system loop (conventional or integrated) adopted in the NPP design?
- Are there any conditions under which the plant could have a positive coefficient of reactivity (e.g. moderator temperature)? If so, what is the justification and control of this condition?
- How were postulated initiating events identified, grouped and classified into plant states?
- What are the margins to dose limits for the NPP design?
- What are the barriers to fission product release in the NPP design?
- What are the margins to barrier related safety criteria for the NPP design?
- Does the DSA address computer code validation and code accuracy?
- Are uncertainties in calculations fully addressed?

KT 3.8 Results of probabilistic safety assessment

PSAs can be performed with different purposes at different phases of the NPP design. For example, a system level PSA can be done at an early stage of the NPP design with an assumed failure frequency of important systems (normally based on similar designs). This provides the system designer with a target reliability for the system that has to be confirmed at a later stage by modelling the system at a component level with appropriate component failure frequencies.

PSA can be performed with different scopes. Examples include:

- Plant operating modes: power operation or low power and shutdown modes;
- Internal events (important for assessment of system design) or external events (important for assessment of vulnerabilities to common cause failures); separate external event PSAs may be performed for seismic, fire and flood events.

Member States and regulatory bodies normally set safety goals for core damage frequency (CDF) and large early release frequency (LERF) or large release frequency.

Level 1 PSA is used to calculate core melt (or damage) frequency for the core in the reactor vessel and, if required, in the spent fuel pool. This provides a good measure of the preventive capability of the design, in particular if sufficient redundancy and diversity is provided. The CDF safety goal does not apply to molten salt reactors but the regulatory bodies in the Member States may have developed alternative safety goals to verify the adequacy of the safety systems for molten salt reactors.

Level 2 PSA is used to compute the radiological release frequency and represents a good measure of the containment capability.

PSA is required for both large WCRs and SMRs. The complexity and the size of PSA models depend on the PSA scope and the evaluated consequences. Clearly, a simpler NPP with fewer systems and components has a simpler PSA.

For modular reactors or multiple units on the site, the Member State or its regulatory body may also use site based safety goals to address multiple core damage accidents.

Supporting publications:

- IAEA Safety Standards Series No. SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [31];
- IAEA Safety Standards Series No. SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [32].

The following questions may support a collection of information relevant to this KT:

- — What are the CDFs and the LERFs associated with internal and external events in different operating modes (with external event assumption bases)?
- — What are the emergency planning zone (EPZ) needs considered in the PSA?
- — What regulatory or peer reviews have been conducted on the PSA methodology, analyses and results?
- — What improvements have been made in the safety design based on the insights from PSA and from operational experiences of the same or similar designs?

KT 3.9 Mitigation of severe accidents

This KT concerns the characterization of the design with respect to its response in the event of a severe accident involving core melt and/or severe change in the fuel geometry in the core. The intent is to evaluate the spectrum and timing of accident release events that are considered in design, and the technical and programmatic response plans and facilities that have been developed as part of the facility design.

The evaluation includes NPP systems, as well as programmes and procedures for emergency and severe accident responses for the plant and its surroundings.

Core composition can lead to challenges with production of combustible gases, for example, zircaloy oxidation with steam producing hydrogen, or molten core–concrete interaction producing hydrogen and carbon monoxide. If flammable gases can be produced, the design is expected to include provisions to prevent energetic events that could challenge containment integrity, such as hydrogen detonation or deflagration. Severe accident behaviour can vary dramatically between designs. Large WCRs may rely on in-vessel retention of molten core materials, while others (generally reactors above a certain size) may have a core catcher. SMRs have a smaller core size which can simplify core cooling, particularly for designs that use high melting point core materials. Some SMRs are said to have eliminated core melt accidents.

The following questions may support a collection of information relevant to this KT:

- — What is the assessed frequency of a severe accident with a large release or early release of radioactivity?
- — What are the severe accident mitigation systems and how are they redundant to and diverse from other safety and operating systems?
- — What systems are included in the design specifically to minimize the consequence of potential severe accidents?
- — What specific post core damage management is implemented (e.g. in-vessel melt retention, ex-vessel core cooling with core catcher)?
- — What systems are required to function to maintain the in-vessel retention or core catcher integrity?
- — What specific combustible gas management is implemented in containment (e.g. ignitors, mixing, venting, passive autocatalytic recombiners)?
- — In the event of severe core damage, what measures are available to prevent containment failure?
- — Describe expectations for severe accident on-site and off-site response. What systems and equipment are needed for severe accident on-site and off-site response?
- — Does the NPP design include connection points to allow use of portable supplies of cooling water and electrical power? Do emergency operating procedures and SAMGs provide instructions in their use?
- — What emergency response facilities and capabilities are part of the proposed NPP design?

KT 3.10 Operational expectations affecting safety

There are several areas of work that become very important during the operational phase of an NPP's life and the technology holder needs to make active preparations to meet the nuclear safety challenges that will arise in these areas. Four important areas are as follows:

Peer review: The IAEA offers a range of review services that can provide peer feedback on various topics important to nuclear safety. Many of these are relevant to the pre-operational stage. The World Association of Nuclear Operators peer reviews and Institute of Nuclear Power Operations evaluations are commonly used during the operational phase of an NPP's life, and both organizations offer pre-operational reviews. Early contact with peer review organizations and preparations to meet their expectations are a positive indicator of a strong nuclear safety culture.

Operational experience: Technology holders and operators of NPPs typically have an effective operational experience (OPEX) programme to learn from worldwide experience. This includes lessons learned from accidents, incidents and near misses. Technology holders and operators need to participate in sharing their experience on reactor components and ageing mechanisms. They need to participate in research that extends knowledge of reactor behaviour and materials science.

Human factors engineering: The human factors engineering programme during the design phase needs to anticipate and address challenges to the operation of the NPP as a whole, and inspection, maintenance and replacement of components that are needed during operation. Human factors engineering considerations need to be included in the design of safety related instrumentation and control from the conceptual stage.

Equipment qualification: During operation of a NPP, one of the challenges that may be faced is to maintain the qualification of equipment during maintenance and inspection. This can have a significant impact on nuclear safety as special precautions are necessary to ensure that all the protective measures required to ensure component environmental qualification are taken (for example, replacement of seals after inspection).

Supporting publications:

— IAEA Safety Standards Series No. SSG-50 Operating Experience Feedback for Nuclear Installations [43];
— IAEA Safety Standards Series No. SSG-51 Human Factors Engineering in the Design of Nuclear Power Plants [44];
— IAEA Draft Safety Guide DS514 Equipment Qualification of Items Important to Safety in Nuclear Installations [45].

There is no significant difference in the role of peer review, OPEX, human factors engineering or equipment qualification for large WCRs and SMRs. There is currently much more maturity in peer review services and OPEX programmes for large WCRs since there is much longer operating experience.

The following questions may support a collection of information relevant to this KT:

— Has the technology holder contacted peer review services to discuss or take part in a peer review?
— What OPEX programmes are available for the NPP design and in which programmes does the technology holder participate?
— What is the programme carried out by the technology holder to incorporate lessons learned from the accidents at the Chornobyl and Fukushima Daiichi nuclear power plants accidents and other incidents and near misses?
— What nuclear safety and operational reliability improvements have been made to the design as a result of the accident at the Fukushima Daiichi nuclear power plant and other accidents?
— What research and data sharing is carried on in materials science and ageing of NPP components?
— What other improvements have been made to the design as a result of regulatory or internal/external design reviews in the past five years?

— How is control room habitability assured in the event of an accident? What is the capability of and pathway to the alternate shutdown facility/panel?
— How is the human factors engineering programme making active preparation for the operational phase of NPP life?
— Does the design of safety related instrumentation and control include human factors engineering considerations?
— What active preparations have been taken to ensure that equipment qualification will be maintained during operation?
— What safety features are developed specifically to support automatic or manual load following capability?

KT 3.11 Fuel storage facility safety

Fuel handling and storage are provided in the design of an NPP to prevent damage to fuel in transfer from the reactor to the spent fuel storage and to protect the fuel while it is in storage at the site. Requirement 80 on fuel handling and storage systems in SSR-2/1 (Rev. 1) [19] states:

> "**Fuel handling and storage systems shall be provided at the nuclear power plant to ensure that the integrity and properties of the fuel are maintained at all times during fuel handling and storage.**"

Supporting publications:

— IAEA Safety Standards Series No. SSG-15, Storage of Spent Nuclear Fuel (Rev. 1) [46];
— IAEA Safety Standards Series No. SSG-63, Design of Fuel Handling and Storage Systems for Nuclear Power Plants [47].

Fresh fuel storage prevents inadvertent criticality, by a specified margin, by physical means or by means of physical processes, and preferably by use of geometrically safe configurations, even under conditions of optimum moderation. These conditions are to be met even in the event of two independent abnormal events occurring simultaneously.

Spent fuel storage typically uses water filled storage pools where spent fuel is submerged by a sufficient depth to provide fuel cooling and adequate shielding for workers in the area. The pool water may be borated to prevent criticality for enriched fuel, though it is more common to use boron steel storage racks to ensure passive safety.

Storage pools contain such a large inventory of water that, except in the case of a major leak, there is a long heat-up and boil-down time before fuel becomes uncovered. This allows significant time for recovery actions. Rapid draining is effectively impossible except for above ground pools. Penetrations to the pool at a low elevation will ideally be avoided and anti-siphoning measures will ideally be taken. Backup cooling and alternate make-up will ideally be provided. If rapid draining cannot be precluded, then a spray system may be implemented.

If the fuel become uncovered, overheating can lead to exothermic zircalloy–steam reactions in fuel cladding, releasing fission products and generating significant quantities of hydrogen. Paragraph 6.68 of SSR-2/1 (Rev. 1) [19] specifies that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is practically eliminated as follows (footnotes omitted):

> "For reactors using a water pool system for fuel storage, the design shall be such as to prevent the uncovering of fuel assemblies in all plant states that are of relevance for the spent fuel pool so that the possibility of conditions arising that could lead to an early radioactive release or a large

radioactive release is 'practically eliminated' and so as to avoid high radiation fields on the site. The design of the plant:

(a) Shall provide the necessary fuel cooling capabilities;
(b) Shall provide features to prevent the uncovering of fuel assemblies in the event of a leak or a pipe break;
(c) Shall provide a capability to restore the water inventory.

The design shall also include features to enable the safe use of non-permanent equipment to ensure sufficient water inventory for the long term cooling of spent fuel and for providing shielding against radiation."

Large WCRs typically refuel either continuously (e.g. pressurized heavy water reactors) or in batches (e.g. LWRs) and move spent fuel assemblies or bundles to storage pools where they are retained for several years before being transferred to long term storage (e.g. in dry storage), disposal or reprocessing. SMRs may use similar techniques or may store the entire sealed reactor assembly after its useful life.

The following questions may support a collection of information relevant to this KT:

— Is there alternate cooling in the spent fuel pool in the case of a loss of integrity?
— Is boric acid required to ensure subcriticality in the spent fuel pool?
— Is the spent fuel pool stored above ground or in ground?
— Is hydrogen generation possible in the event of a loss of coolant or loss of cooling and how is it mitigated?
— For an SMR with whole core replacement, how long is the spent core to be stored on the site and what are the arrangements to transport it off the site?
— What type of neutron absorber, if any, is used in the spent fuel pool?

KT 3.12 Management system

The management system integrates all elements of management so that requirements for safety are established and applied coherently with other requirements, including those for human performance, quality and security and so that safety is not compromised by the need to meet other requirements or demands.

Supporting publications:

— IAEA Safety Standards Series No. GSR Part 2 Leadership and Management for Safety [36];
— IAEA Safety Standards Series No. GS-G-3.1, Application of the Management System for Facilities and Activities [37];
— IAEA Safety Standards Series No. GS-G-3.5, The Management System for Nuclear Installations [38];
— IAEA Safety Standards Series No. SSG-51, Human Factors Engineering in the Design of Nuclear Power Plants [39].

SSR-2/1 Rev.1 [19], Requirement 2, states that:

"**The design organization shall establish and implement a management system for ensuring that all safety requirements established for the design of the plant are considered and implemented in all phases of the design process and that they are met in the final design.**"

It is important that an effective management system is implemented early in the design process as re-establishing control is very difficult at a later stage and much rework may be needed.

Large WCRs and SMRs are designed, constructed and operated under a management system. There is no fundamental difference to this requirement between the two classes of NPP.

The following questions may support a collection of information relevant to this KT:

- — Is there a comprehensive management system in place to control the design and safety assessment of the NPP?
- — Is a human factors engineering programme in place?
- — How complete is the management system development to handle procurement, training, commissioning, operation, maintenance, inspection and change control?
- — What are the international and national standards used under the management system?
- — How will transfer of control from design to construction and from construction to commissioning and operation be managed?

5.4. NUCLEAR ISLAND DESIGN AND PERFORMANCE

KE4 considers site specific parameters with impact on the NPP design, which could differentiate among the technologies under consideration, based on the following ten KTs:

KT 4.1 Plant size;
KT 4.2 Plant availability and capacity factors;
KT 4.3 Plant lifetime;
KT 4.4 Standardization;
KT 4.5 Simplification;
KT 4.6 Constructability;
KT 4.7 Operability, inspectability, maintainability and reliability;
KT 4.8 Manoeuvrability;
KT 4.9 Plant control and protection architecture;
KT 4.10 Radiation protection.

Importance rationale (Table 7): The nuclear island design and performance can be an important differentiator in all or some of the KTs and it is more pronounced for SMRs than for large WCRs.

Description: Nuclear island design and performance are compared between the assessing NPP designs.

The following IAEA publications can provide further guidance on importance and weighting appropriate to this KE:

- — IAEA Nuclear Energy Series No. NP-T-2.5 Construction Technologies for Nuclear Power Plants [48];
- — IAEA Safety Standards Series No. SSG-50 Operating Experience Feedback for Nuclear Installations [43];
- — IAEA Safety Standards Series No. SSG-39 Design of Instrumentation and Control Systems for Nuclear Power Plants [49].

The ten KTs are described as follows and examples are provided in the Annex.

KT 4.1 Plant size

Substantial differences exist among large WCRs, as well as SMRs in plant size. Since calculation assumptions and approaches may vary among technology holders, it is important to develop an approach to ensure consistent results related to NPP output. The key questions examine the advertised versus the actual expected NPP output.

This KT includes the assessment of the NPP designs against the following values:

— NPP net electric output (MW(e));
— NPP non-electric output.

The RTA team is to be reminded that there are several alternatives to achieve the same electrical output from the NPP and therefore to maintain flexibility in the definition of the plant size. For example, the same NPP electrical output can be achieved by a single WCR unit or several co-located SMR units. Some SMR designs accommodate multiple reactors connected to one set of turbines/generators, in which case the traditional 'unit size', based on the nuclear island output, is not applicable to the NPP.

There are pros and cons to achieve an electrical output by a single WCR or multiple co-located SMR units:

(a) The pros for a single WCR unit, which are cons for multiple SMR units, include the following:
    — In general, a larger unit is more economical than multiple SMR units that achieve the same electrical output by the rule of 'economies of scale';
    — A larger turbine system usually has better efficiency, which leads to a higher net electrical output for the same thermal output from reactor unit(s).
(b) The pros for multiple SMR units, which are cons for a single WCR unit, include the following:
    — Revenue can be obtained by electrical power sales from completed units while constructing remaining units, and thus initial construction investment for the same output can be smaller;
    — Risk to stop the whole output by a major incident is smaller when there are multiple units;
    — Multiple outages can be distributed in a year so that total interruption of power output by an outage can be avoided and resources needed for outages are levelled off.

The following questions may support a collection of information relevant to this KT:

— What is the proposed plant size?
— What is the range of available unit sizes for this procurement and for this NPP site?
— What is the NPP net rated electrical output and NPP efficiency for the environmental condition profile specified for the site (heat sink, air, humidity…)?
— What is the gross electrical output on this site and what are the major contributors to the auxiliary power consumption?

KT 4.2 Plant availability and capacity factors

The plant availability is the fraction of time that the NPP is capable of fulfilling its intended purpose, while the capacity factor is the actual energy output of an electricity generating device divided by the energy output that would be produced if it operated at its rated power output for the entire year.

Plant availability and capacity factor have a strong influence on the power generation costs. These are classified as high importance factors because they can be directly related to operational effectiveness, and highly depend on the vendor's or other operators' operating experience of the NPP design and its components. Caution is to be used in assessments and their potential economic assessment (specifically in Phase 2, see also KT 10.2) considering the following:

— Experience in NPP availability and capacity factor for similar installed units (source of such information can be the IAEA Power Reactor Information System (PRIS) database);
— Materials assessment and life expectancy;
— Expected capacity factor for the new NPP, (e.g. from the IAEA ARIS database).

It is critical to ensure that a careful assessment is done to predict the availability and capacity factors for every assessing NPP design. Issues which are to be accounted for are:

— Local conditions of operation to capture variations in NPP thermal output (a site specific versus a generic calculation will give considerably more accurate information).
— Historical factors that depend upon local regulations and maintenance practices, so that a detailed review of the reasons for unavailability can be performed based appropriately on historical data from a country, region or company. Historical datasets may need to be adjusted based upon the reporting organization's definitions of capacity and availability factors, which depend on a variety of NPP capability conditions, including plant modifications over time, such as power uprates or other operational considerations or changes.

An example of setting an unbiased scoring for capacity factors is provided in Table 5.

There are some differences between large WCRs and SMRs in terms of plant availability and capacity factors:

— Almost all the SMR designs known at the time of this publication do not have actual operation experiences and track records on their plant availability and capacity factors unlike the operational large WCRs. The RTA team may consider asking for the vendor's justification why or how the claimed plant availability and capacity factors can be achieved.
— Historical records show that the conventional large WCRs have had various unexpected problems on their systems, components and materials in the beginning and middle of their lifetime, which caused downtime of the NPPs. From there, the NPP availability and capacity factors have been improved gradually by solving the issues. Since innovative SMRs may employ new systems, components and materials, they could follow the same or similar historical record, which means that the availability and capacity factors may be lower in the beginning and middle of their lifetime compared to the specifications.

Multiple co-located SMR units can have a clear advantage over the plant availability factor as an aggregation. As discussed in KT4.1, multiple outages among units can be distributed in a year so that total interruption of power output by an outage can be avoided and the availability factor can be achieved. The capacity factor, however, cannot be improved in this manner.

The NSSS design can be very different for large WCRs and SMRs. While the operating experience is less available for SMRs, their designs include more passive safety systems. Testing and qualification of passive safety systems needs different methodologies and commissioning and periodic testing could be more difficult for SMRs.

The following questions may support a collection of information relevant to this KT:

(a) In general:
 — What is the NPP availability of operating plants of this type?
 — What average NPP availability and capacity factors will you achieve for the local conditions, considering both technical capability and local regulations?
 — What is the demonstrated in-plant operational experience for major NSSS systems and components? How many years of operation?
 — What changes have been made to address past limitations to improve NPP availability and capacity factors?
 — What are the principal contributors to planned and unplanned capacity loss factors?
 — What operational experience and/or experimental validations do you have on the proposed design or on a similar design?
 — What experience do you have with different fuel cycles (18−24 months) used in this reactor or in similar reactors?

(b) SMRs specific:
  - What is the rationale for targeted plant availability and capacity factors to be achievable without actual operating experience and records?
  - What are the systems, components or materials that are newly employed in the nuclear power plant design and how do you make sure that they will not have unexpected issues that will prevent continuous plant operation?
  - Does the plant have a common control room for several reactors? If yes, what is the experience feedback?
  - What operational experience does the vendor have on the proposed BOP design or on a similar design in other power plants (coal, oil)?
  - What is the multiple unit operation/outage option to improve the plant availability factor?

KT 4.3 Plant lifetime

This KT addresses the length of time that the NPP is designed to operate. The differentiator among the assessing NPP designs is based on the national goals.

The NPPs based on large WCRs have a designed operational lifetime that is typically 60 years or longer, and technology holders maintain margins associated with this determination. Extra consideration of this KT would be warranted for assessment if the NPP design has features that are known to have limited lifetimes. Different considerations are given to the assessment of SMR designs.

The assessment of the NPP designs may include the following values:

- Economic design life and physical life specification (years);
- Limiting equipment or components and their design lifetimes;
- Replaceability of components intended to be changed during plant lifetime (including impact on plant lifetime availability);
- Major component design margins (e.g. vessel fluence, containment, physical structures);
- Materials assessment and availability including spare parts and capability of common suppliers;
- List of components and structures that meet the design life requirements without refurbishment;
- Design margin analyses to assure plant lifetime optimization for major components and structures;
- Long term assurance for component and replacement parts availability;
- Design lifetime refurbishment plan:
  - Definition of the main scenarios and economic estimates involving component replacement and outages for refurbishment;
  - Identification and estimate of specific maintenance, surveillance and condition monitoring results that are crucial for determining schedules and periodicities for refurbishment;
  - Design lifetime assessment report analysing the industrial experience on ageing mechanisms and justifying the material selection;
  - Commercial and contractual treatment of described commitments.

The following questions may support a collection of information relevant to this KT:

- What is the programme for replacement of components and structures to attain the design lifetime? Describe the component replacement plan required to achieve design lifetime in terms of year intervals, including cost experience or cost estimates.
- What is the lifetime design margin for the reactor vessel, the containment or other major replacement cost items? What are the estimated replacement costs?
- What are the assumptions regarding regulatory factors that influence achieving the design lifetime?
- What are the assumptions regarding the availability of replacement components, materials, equipment and parts? What assurance is provided regarding the availability of suppliers or commonality of replacement supply?

- How does the total lifetime radiation embrittlement to the major reactor components and structure compare between the SMR design and operational large WCRs?

There are no significant factors affecting different lifetimes between the large WCRs and SMRs, since lifetime depends on components, structures and materials and their operation/maintenance/replacement policy, but not the size of the reactor unit. There are, however, some factors that need to be considered regarding the lifetime of SMRs. Some of the SMRs may have a higher power density and/or higher neutron spectrum, which may cause higher radiation embrittlement of the reactor components like reactor pressure vessel and concrete structures including containment. The selection of a reactor coolant in the innovative SMRs may impact the amount of radiation embrittlement as well. The higher embrittlement may cause a shorter lifetime for particular components.

KT 4.4 Standardization

Standardization is the extent to which NPP design, or its major components, can be built to an established standard design. Standardization has become a general expectation within the industry for NPPs under construction and will be important for SMR and other advanced reactor designs. Standardization can be equally important to an owner/operator with a new programme of only a few units, to ensure the opportunity to share and apply lessons learned from and with the standardized fleet, or if the plan is to eventually build many units or NPPs of the same or similar design. The KT is equally important for large WCRs and SMRs, but standardization will likely be more developed for SMRs because a greater part of the NPP will be manufactured in factories (prefabrication); this is a key condition for enabling the SMR industrial/business model.

Technology holders are expected to provide a comprehensive description of their standardization practices and related history that can be used to evaluate this KT and develop meaningful comparisons commensurate with the importance factor and potential connections of this KT to the policy objectives.

The following questions may support a collection of information relevant to this KT:

- To what extent are the equipment, components and materials standardized?
- What is the experience of availability of replacements of standard material, components and equipment?
- What long term assurance of supply can be given for standardized equipment that needs to be replaced?
- To what extent can standardized equipment and components of the nuclear power plant be adapted to suit local conditions?
- What is the level of experience in information sharing between users or owners of the standardized plants?
- What is the standardization philosophy for different site conditions such as seismic condition?

KT 4.5 Simplification

The minimization of the number of types of systems and components, without adverse impacts on the NPP economics, performance and safety, while improving ease of operation and maintenance are addressed with this KT.

Whereas simplification is considered very important, especially with respect to new technologies and for SMRs, it is expected to be a common approach for all technology holders. Therefore, it is not expected that simplification will be a strong differentiator, or that it will be a major contributor to the policy objectives. It could have a significant effect on economics of construction, operation costs for replacement parts, and labour requirements for operation and maintenance support. Simplification will ideally have a positive impact on nuclear power plant safety.

The NPP designs are assessed against this KT in more detail in Phase 2, for which technology holders are to provide a comprehensive description of their simplification practices and related history that can be used along with the KTs to develop meaningful comparisons commensurate with the importance factor and potential connections to the policy objectives. Caution is to be used to ensure that benefits related to simplification are not double counted.

In assessing the NPP designs the following values may be considered:

— Comparative design simplification for NSSS, components, operations and safety systems;
— Human–machine interface systems to simplify plant operation and facilitate maintenance;
— Plant construction simplification with design to facilitate on-line maintenance;
— Systems and equipment design and control room design to minimize demands on plant operators during normal and emergency conditions;
— Simplified control logics;
— Use of a minimum number of systems and components (e.g. pumps, valves, instruments, electrical equipment) to meet essential functional requirements;
— Design that facilitates plant construction;
— Building arrangement, equipment design and layout to simplify and facilitate maintenance;
— System redundancy to support on-line maintenance;
— Operator actions for transients/accidents (available/required response time).

The following questions may support a collection of information relevant to this KT:

— What changes have been made in this design to reduce operator actions during normal and transient operations? To what extent have operator actions been reduced?
— What is the experience with the proposed human–machine interface in the plant design?
— What improvements have been achieved in the maintenance conduct and procedures due to simplification of the design?
— What is the minimum number of reactor operators required considering the regulatory requirements?
— What are the simplifications in the SMR design and what are the benefits?

Simplification could be one of the common characteristics of the SMRs, since many SMRs try to overcome economic disadvantages implied by economies of scale by simplifying the systems. Also, the SMRs can employ some of the simplification strategies more easily than the designs of large WCRs. One of those examples could be natural circulation of reactor coolant.

KT 4.6 Constructability

This KT addresses the technologies and methods that will be used during the construction of the NPP and if they may present a differentiator among the assessing NPP designs. It is more relevant to address this KT in detail in Phase 2.

Constructability is considered to be quite important but may not be expected to be a key differentiator on its own for large WCRs. Rather the benefits will be seen in lower capital cost estimates, lower construction time periods (financing costs) and lower construction schedule risk factors. However, constructability may present a strong differentiator for SMRs as construction may require specific design and construction options (due to modularization) for the site addressing potential infrastructure limitations and on-site needs.

Technology holders are to provide a comprehensive description of their constructability features and proven construction practices that can be used to develop meaningful assessment commensurate with the importance factor:

- The actual versus planned experience with the construction plans for comparable designs, site and environs, workforce and locale considerations;
- Proposed construction schedules compared with their experience base with plant type and construction plan.

Caution is to be used to ensure that any benefits related to constructability are not double counted, given the tight coupling to capital costs.

The following values may be analysed in assessing the large WCR NPP designs:

- Detailed construction plan, management, schedule, resources and interfaces;
- Use and completeness of construction work packages;
- Size of laydown area;
- Planned construction housing facilities;
- Extent of modular construction, demonstrated capability, manufacturing, management, transportation and installation support requirements;
- Construction quality assurance (QA) programme and demonstrated site construction quality;
- Construction safety and health programmes during and after construction;
- Civil site preparation and earthwork requirements and schedule;
- Use of smart construction techniques and construction management.

The following questions may support a collection of information relevant to this KT:

- What is the proposed construction schedule, including major milestones, and what is the experience record with intended partners?
- What advanced construction techniques have you included in the plan?
- Have you executed the proposed construction plan or a similar plan? If so, describe the actual versus expected performance.
- What is the modular construction plan and what modular construction techniques are you proposing to use?
- What are the transportation challenges expected for the reactor vessel and other major components, and what equipment (e.g. crane size) is required for installation?
- What has been the experience in the use of modular construction techniques — actual versus planned? What issues were identified, how were they resolved and what was their impact on cost and schedule?
- What are the construction challenges expected in delivering the proposed design for operation on schedule, given the available site, environs and workforce?
- What are the bulk material quantities for construction (e.g. concrete, steel)?
- What land area and water source and supply are required for construction, including all layout and temporary use land areas?
- Have you used consistent systems of measurement units throughout the design documents?

The use of modular construction techniques is suitable for SMRs. Since each of their components and structures are smaller than in large WCRs, it is easier to prefabricate them in a factory or at different sites and transport them to the construction site. While the modular construction technique may bring big benefits, such as the potential for shortening a construction period and cost if correctly applied, there are challenges associated with this technique, such as difficulties to adjust the interfaces (e.g. pipe fitting) between modular components at the construction site. Success of the modular construction mainly depends on careful preplanning and lessons learned. The adaptation of the modular construction technique may bring, therefore, a risk if the contractor/supplier has no or little experience of modular construction.

Another aspect is that some of the SMRs are of an innovative plant layout, such as underground design concept, which may require construction techniques that are not used in conventional NPP construction. Therefore, those techniques may or may not be adaptable and reliable in the construction of an SMR.

The following additional questions may be addressed to the SMR vendors:

— What is your experience of the modular construction technique in past nuclear construction projects and what are the lessons learned?
— What are the construction methods not used in conventional nuclear construction projects to be applied in the SMR construction and how are you sure that those methods are adaptable and reliable for timely construction of the SMR design?

KT 4.7 Operability, inspectability, maintainability and reliability

This KT addresses the methods, technologies and experience involved in maintaining safe and reliable operation of an NPP.

All these areas of performance are a function of the capabilities of the owner/operator as well as the NPP design features and are expected to be addressed in detail in Phase 2. To the extent that these areas also appear in other KTs on availability and capacity factors, for example, overlap will ideally be avoided.

In assessment of the NPP designs the following values may be of relevance:

— Operational and design margins in normal operating modes;
— On-line and off-line maintenance expectations versus experience;
— Fuel transfer system capability and refuelling outage duration versus experience;
— Remote technology options for inspection, monitoring and maintenance;
— Plant trip response (design versus experience), including trips of the reactor, turbine, feedwater and main condensate pump;
— System redundancy and logic switchover systems to avoid trips;
— Mean time between failure and mean time between maintenance for key components;
— Design margins (design, technical specification limits and operational margins), including equipment redundancy;
— Reactor normal shutdown and cooldown process descriptions;
— Emergency remote shutdown requirements versus capabilities;
— Critical path comparisons;
— Major maintenance comparisons;
— Major component and reactor internal replacement comparisons;
— Ability to remove and transport major components (on-line/off-line);
— Replaceability/reparability of control and instrumentation systems;
— Containment accessibility during NPP operation.

The following questions may support a collection of information relevant to this KT:

— What plant commissioning support (e.g. startup support) will be provided?
— What support is available after plant commissioning, including plant operation and maintenance?
— What reactor simulators are provided and what is the proposed operator training programme and schedule?
— What experience base exists for this simulator facility and for the operator training programme offered?
— What is the on-line and off-line maintenance programme plan and what is the experience base that supports its design?

- What is the mean time between failure and mean time between maintenance for the key components which affect overall plant availability?
- How have adequate space and access requirements for efficient maintenance been assured in this design?
- Which other maintenance operations require an outage? What is the periodicity? For how long? Can they be done during the refuelling outage?

Since the SMRs can be more expensive in operation and maintenance than the large WCRs, this KT is particularly important for the SMRs. For example, if the number of operators per unit is the same, the number of operators per kWe could be significantly larger in SMRs than in large WCRs. Although the operation and maintenance cost are both in principle controlled by operational practice, the SMRs will ideally accommodate engineering features built in the design that can mitigate the impact of the operation and maintenance cost. Examples of those features are a centralized operation centre for multiple units and maintenance system with big data connected with remote sensor components. Securing enough space for maintenance is also a challenge, particularly for SMRs. Many of the SMR designs are reactors of a first kind and actual operational or maintenance experiences are inexistent.

The following additional questions may be asked to SMR vendors:

- What are the engineering features built in the design to overcome operation and maintenance cost increase?
- What is the commercial prospect of constructing the SMRs worldwide and in our region for the next 10−20 years?
- Do you have any plan to help create a users/owners' group for the SMR design?
- What is the design strategy to achieve a smaller footprint for SMR but secure adequate space and access for efficient maintenance?

KT 4.8 Manoeuvrability

This KT addresses the capability of the NPP to cope with varying demands from the grid. It can also address the capability of an NPP design to be integrated with a renewable energy system at the site or in close proximity to the site.

The importance of this KT could be in the range of low to high. For example, a grid with high demand fluctuation will need manoeuvrability capability, so that the NPP design features that support such operation could become critical to success. The assessment needs to take into consideration changes in demand across NPP's lifetime and plans on NPP integration with renewables into a tightly coupled hybrid energy system for electricity production and non-electrical applications.

Specifically, for Phase 2, the experience base of the technology holder needs to be established through reviews of technology holder information, historical experience and discussions with other operators of the equipment (or similar equipment) when possible. Evaluating the technical features and the key features of load following without data from experience is difficult. Careful consideration is to be given to historical data regarding fuel reliability in load following situations, ensuring that the experience is applicable to the proposed fuel design and NPP operational modes.

The following values may be considered in assessing the NPP designs against this KT:

- Operational margins, design margins in normal operating modes;
- Load following and related operational manoeuvrability versus specifications;
- NPP design capabilities for integration with renewables into a tightly coupled hybrid energy system for electricity production and or non-electric applications;
- Impact of power generation by renewables on manoeuvrability needs;
- Emergency remote shutdown requirements and capabilities;
- Waste generation;

— Fuel power and ramp rate limit constraints;
— Heat-up and startup, power ascension;
— Reactor normal shutdown and cooldown;
— Load rejection requirements versus capability;
— Steam bypass system requirements versus capacity.

The following questions may support a collection of information relevant to this KT:

— What are the load following capabilities of the NPP and what is the experience of load following operation?
— What is the load rejection capability of the NPP without shutdown?
— What are the operational margins of the NPP design during normal full power and load follow operation?
— What are the capabilities for remote shutdown?
— How much increase in waste generation will be caused by load following (e.g. boron inventory management considerations in large WCRs)?
— Are there any concerns on fuel mechanical conditioning and capability during load following (e.g. pellet–cladding interaction)?
— What is the power scalability of the reactor?
— What kind of dedicated core protection design is needed to support this power scalability of the reactor?
— What is the power scalability of the plant: power per module/unit, maximum number of modules/units, relation between nuclear power modules/units and turbine island(s), cooperation of power modules/units, concept of operating of multi-modules/multi-unit plant?
— What is the load following capacity?
— Are the load following plant capabilities based on core power flexibilities?
— In how many minutes can the reactor reduce its production by 50% or 80%?
— What are the impacts on cost and maintenance for a reactor practicing load following?

The large WCRs and SMRs are not expected to have fundamental differences in their manoeuvrability, but their embedded engineering features may impact their manoeuvrability. For example, some of the SMRs incorporate natural circulation of reactor coolant, which may have an impact on load following rate performance. Some of the SMRs employ a thermal or power storage system combined with their power generation system. In that case, manoeuvrability has to be looked at not by the reactor system itself but in combination with the storage system. In the case of multiple co-located SMR units in a plant, manoeuvrability can be considered by a set of SMR units instead of each SMR unit. For example, if there are ten small units, which is equivalent to a large WCR, 50% reduction in power can be achieved by temporally shutting down five units. Innovative SMRs may employ a higher neutron spectrum than conventional large WCRs and restart-up characteristics after shutdown due to a Xe buildup effect may be different.

The following additional questions may be addressed to SMR vendors:

— What is the expected load following rate (% power change/minute)?
— Do you have any option to accommodate a thermal or power storage system to improve manoeuvrability?
— Is the neutron spectrum different from the conventional WCR; how does the difference impact the restart-up capability after reactor shutdown?

KT 4.9 Plant control and protection architecture

This KT addresses the overall plant instrumentation and control system including the reactor protection system, which is the most critical safety control function in NPP. It is speculated here that the digital control system is largely adapted although a digital control system and analogue control system may coexist.

The plant control architecture defines the logical hierarchy and structure of control functions and requirements, and each control platform implements control functions according to the NPP control architecture. The NPP control architecture is usually developed by the NPP vendor and the control platforms are supplied by controller vendors. While the NPP control architecture has a strong connection to the reactor design, control platforms are usually independent from the reactor design, and thus robustness and reliability of a NPP control platform can be generally assessed by itself and apart from its reactor design.

The following values may be considered in assessing the NPP control architecture and platforms within this KT:

— Plant protection architecture:
  - Existence of a licence for architecture as a part of the plant licence in the country of origin or any other Member States (Nuclear Regulatory Commission Design Certification as an example);
  - Existence of integrated architecture to define requirements over safety and non-safety control platforms and reactor island and turbine island control platforms;
  - Degree of redundancy and diversity;
  - Human factor consideration for human–machine interface design;
  - Integration and separation of plant control system, plant internet of things system and enterprise information technology (IT) system including enterprise asset management;
  - Earlier availability of plant simulator;
  - Transparency of control design and its verification and validation (V&V) process.

— Plant control platform:
  - Existence of a license/certification/accreditation on the digital control platform in the country of origin or any other Member States (International Electrotechnical Commission's safety integrity level certification as an example);
  - Consideration against future obsolescence issues on control spare parts;
  - Automatic and remote/on-line diagnosis of component failure;
  - Ease of maintenance and repair of control panels, components and parts.

— Plant protection architecture and plant control platform:
  - Licensability of a digital control architecture and platform in the assessing country;
  - Robustness in cybersecurity and existence of its accreditation;
  - Establishment of configuration management between architecture (requirements) and platform (implementation);
  - Independence between architecture and platform.

The following questions may support a collection of information relevant to this KT:

— Has the plant design including its plant control architecture been licensed in the country of origin or any other Member States?

— Have the control platforms to be used in the plant been licensed, certified or accredited in the country of origin or any other MSs?

— Are the plant control architecture and platforms licensable in my country?

— Is the plant control architecture integrated so as to define requirements over safety and non-safety control platforms and reactor island and turbine island control platforms?

— What extent does the plant control architecture consider redundancy and diversity of platforms?

— How robust are the plant control architecture and platforms against cyberattacks?
— Is a configuration management system established and available for users?
— Are the plant control architecture and platforms independent, and are platforms replaceable without change of the architecture?
— How many years will you supply the original control spare parts?
— How easily can operators find cause of failure or malfunction of control components?
— How easily can maintenance personnel undertake maintenance and repair of control panels, components and parts?
— How are the plant control system, plant internet of things system and enterprise IT integrated for efficient operation while maintaining security of the plant control system against cyberattacks?
— Is the plant control architecture designed to accommodate a plant simulator that is ready in the early stage of the construction for operator training?
— To what extent is the plant control design and its V&V process shared with users?

Generally, SMRs are simpler and employ more passive safety features than large WCRs. The passive safety features take advantage of natural forces like gravity and require less active control functions. Therefore, the importance of this KT may be low for SMRs.

KT 4.10 Radiation protection

This KT refers to the ways in which the protection of people from the effects of ionizing radiation is achieved. Individual characteristics of the radiation protection design and planning may show important variance among assessing NPP designs; however, in the general radiation protection programme, it is expected that radiation protection in the work environment will be maintained according to rules, policies and laws.

SSR-2/1 [19] addresses radiation protection in the NPP design:

> “**The design of a nuclear power plant shall be such as to ensure that radiation doses to workers at the plant and to members of the public do not exceed the dose limits, that they are kept as low as reasonably achievable in operational states for the entire lifetime of the plant, and that they remain below acceptable limits and as low as reasonably achievable in, and following, accident conditions.**”

The other suggested publication is IAEA Safety Standards Series No. NS-G-1.13, Radiation Protection Aspects of Design for Nuclear Power Plants [50], which addresses radiation protection by design. Therefore, the differentiation between the NPP designs may include the following values:

— Separation of clean and radiation areas; radiation area zoning in the NPP design plan;
— The ALARA (as low as reasonably achievable) principle and radiation protection procedures, shielding and radiation monitoring implementation in design, including rationale for ALARA improvements via design;
— Procedures and shielding required for exposure reduction during operation, refuelling and maintenance;
— Remote maintenance equipment design and usage;
— Access control and layout design criteria;
— Estimated total annual site personnel dose exposure;
— Personnel exposure estimates during operation, refuelling and maintenance activities;
— Available projections versus actual exposure and exposure reduction comparisons during operation, refuelling and maintenance activities;

- Operational waste generation, for example, spent ion exchange resins used for purification of the primary water system and wet SNF storage facility, replacement of contaminated off-gas systems and generators, activated core waste, liquid waste generated during outages;
- Post-accident vital areas accessibility and shielding.

Specifically, in Phase 2 the dose evaluation bases are analysed carefully with the goal of validating estimates versus actual experience from radiation protection programmes and personnel exposure. This KT is independently assessed to the KTs in KE1.

The following questions may support a collection of information relevant to this KT:

- What information is available and demonstrates clear benefits with regard to comparative dose reduction and ALARA application?
- Under which International Commission on Radiological Protection standards was the NPP designed?
- Are any special radiation protection facilities or support industries needed to handle operational waste?

From experience, the plant radiation dose level slightly varies by NPP designs. It is because the variation is mostly dependent on NPP's operational and maintenance processes and practices, including operational radioactive waste generation and management. In this regard, large WCRs and SMRs are not expected to be fundamentally different regarding radiation protection characteristics provided that both have proper radiation area control and radiation shielding designs. The SMRs are more receptive to adapt innovative design features including those to reduce dose levels. The following are some of the examples:

- Excluding the use of materials that can be a major dose contributor such as cobalt;
- Core and/or fuel design that does not allow fuel failure to contribute to the dose increase;
- Design consideration for ease of decontamination processes such as chemical decontamination and clean-up lines during outages;
- Build in remote sensing to reduce human interactions with high dose components;
- Build in an advanced radiation area monitoring system to prevent human access to a high dose area;
- Ease of maintenance and provision of enough maintenance space in a high dose area;
- Minimization of possibilities of future replacement or modification of high dose components, such as reactor internals and steam generators.

The following additional questions may be addressed to SMR vendors:

- Do you use materials which can be a major dose contributor such as cobalt?
- What engineering features are built into the design that contribute to dose reduction?
- How does the fuel and core design prevent dose increase from fuel failure?
- Does the design include advanced remote sensing and monitoring systems?
- Are the structures of reactor and primary systems simple and spacious enough for easy maintenance?
- What is the expectation of possibilities of future replacement or modification of high dose components, such as reactor internals and steam generators?

5.5. BALANCE OF PLANT DESIGN AND GRID INTEGRATION

KE5 considers the balance of plant design and grid integration of a NPP, which could differentiate among the technologies under consideration, based on the following six KTs:

KT 5.1 Net thermal efficiency;
KT 5.2 Grid electrical code requirements;
KT 5.3 Protection against internal and external hazards;

KT 5.4 Standardization of major components;
KT 5.5 Power requirements from grid under normal operation;
KT 5.6 Ability of grid to accept added generating capacity.

Importance rationale (Table 7): Unique or challenging features of the grid arrangement for the BOP interface in both initial and lifetime operation is critical to the NPP's safe, economic and reliable operation. Standardization of major components may differentiate among SMR designs for different types of coolant. Protection against external hazards may be a differentiator for both large WCRs and SMRs and depends on the adaptability of a design to the site conditions.

Description: The interface between the BOP, site and the grid system in which it is to operate, including normal operation, off-normal operation, including external hazards and transient conditions on the grid and combinations thereof.

The following IAEA documents can provide further guidance on importance and weighting that could be appropriate to this KE:

— IAEA Nuclear Energy Series No. NG-T-3.8 Electric Grid Reliability and Interface with Nuclear Power Plants [35];
— IAEA Safety Standards Series No. SSG-34 Design of Electrical Power Systems for Nuclear Power Plants [34];
— IAEA-TECDOC-1778 Nuclear Power in Countries with Limited Electrical Grid Capacities: The Case of Armenia [51].

The six KTs are described as follows and examples are provided in the Annex.

KT 5.1 Net thermal efficiency

This KT refers to a NPP's net thermal efficiency, defined as the ratio of electrical output (MWe) to the core thermal power (MWth). It therefore includes the assessment of the NPP designs against the NPP efficiency defined as net MWe/MWth.

The net thermal efficiency mainly depends on steam (temperature, pressure and quality), condenser coolant temperature, type and performance of the steam turbine system including moisture separator reheater, house load and heat losses from the primary loop. In SMR designs, the turbine system is smaller and, in general, smaller turbine systems can be less efficient. On the other hand, house load per output can be smaller in some SMRs when compared to conventional large WCRs. An example is the integral type WCRs employing natural circulation in the primary coolant, which eliminates substantial house load for coolant circulation pumps. Non-water cooled SMRs, such as sodium cooled reactors and high temperature gas reactors, have a higher net thermal efficiency due to higher coolant temperatures.

The following questions may support a collection of information relevant to this KT:

— What is the NPP (range of) net thermal efficiency in normal full power and load following operations (MWe to grid/MWth)?
— How large is the house load and how does it affect plant net thermal efficiency? What is the steam condition (temperature, pressure and quality) from the reactor island?
— What is the expected cooling method of the condenser (sea/river/lake water cooling or cooling tower) and what is the justification of this selection on the optimization of the net thermal efficiency gain and impact to the environment?
— What is the rationale that the proposed turbine system is optimal to the proposed reactor in terms of the net thermal efficiency?
— What is the consideration in design to maximize the net thermal efficiency to adjust short term environmental temperature change (summer versus winter) and long term environmental temperature change (global warming effect)?

— What is the strategy to minimize house load?

KT 5.2 Grid electrical code requirements

This KT refers to grid code requirements, that is, the technical specifications that define the parameters for a NPP to be connected to a public electric grid to ensure safe, secure and economic functioning of the electric system. Requirements include basic parameters, such as alternating current (AC) frequency and allowable variations in frequency and voltage and may include many more regulated requirements. The following are important code requirements that can be commonly found in the technology provider countries.

— Requirement for plant resistance against variations in frequency and voltage: typical values for frequency and voltage variations are ± 5% and ± 10%, respectively;
— Requirement for fault ride through capability of continuous operation under grid disturbance;
— Requirement for frequency sensitive code: capability of plant power output changes in response to a change in grid frequency, in such a way that the plant assists with the recovery to target frequency;
— Requirement for island mode: capability of continuous operation in isolation from the national or local electricity distribution network.

This KT is different from KT 5.7, which deals with the capacity of the local electrical grid to accept the NPP.

The share of electricity supply from a single NPP is suggested not to exceed 10% of the total electricity capacity produced by all energy resources in a single grid. The situation is different in the case of numerous interconnections with other grids. As a first approximation, the value of 10% of the grid capacity is a rule of thumb commonly accepted to be the upper limit for the capacity of any single additional unit of any type in order to prevent instability and unreliability of the grid system. Based on this grid limitation, an SMR is considered a viable option for countries with small and vulnerable electricity grids. Moreover, if the total output is the same between multiple co-located SMR units and one large WCR unit, the impact to the grid with loss of power by a unit is less with SMRs than with WCRs. These are not grid code requirements, but important considerations for grid stability.

The following questions may support a collection of information relevant to this KT:

— What are the grid code restrictions applicable to the power station and to what extent does the design meet them?
— Does the plant electrical system design comply with local AC frequency and allowable variations in frequency and voltage?
— Does the plant have required capability of fault ride through?
— Does the plant have required capability of frequency sensitive mode?
— Does the plant have required capability of island mode?

KT 5.3 Protection against internal and external hazards

This KT addresses the differences between the assessed designs in regard to the external events considered within the design of the BOP portion of the NPP only. This KT is connected to KT 1.8 and to KT 3.4:

KT 1.8 External events

Some of the external events of potential consideration that may differentiate the assessing NPP designs are described with this KT. It is site specific and the RTA team may modify and adjust this KT accordingly. The external events are usually grouped into those caused by natural events and those caused by human activities. External events related to seismicity, weather and water bodies are covered in KTs 1.1, 1.2 and 1.3, respectively.

The NPP technologies may differ due to (non)existing flexibilities to incorporate additional design modifications in thus assuring safe operation under extreme conditions caused by external events, such as but not limited to accidental aircraft impact, nearby explosions and dispersions of toxic materials, volcanic hazards (ballistic projectiles, ash, lava), electromagnetic interference, or forest fires.

KT 3.4 Protection against internal and external events

These two KTs ought not to be addressed in the same way in the assessment, rather KT 1.8 will ideally consider to what degree the design is adaptable to the extreme events at the site while KT 3.4 will ideally address which external events are already incorporated within the design taking into account the design criteria and design margins, and if multi-effects are considered in the design and the PSA of the NPP.

There is no distinctive difference between the large WCRs and SMRs to evaluate this KT.
The following questions may support a collection of information relevant to this KT:

— Which external events are considered in the design of the BOP and what are the associated design criteria and margins?
— How have severe external events or multiple external/internal events been considered in the design and operating procedures?
— What provisions are taken to protect the switchyard and its supplies from external events?
— What provisions are taken to avoid turbine damage, which could generate a turbine missile?

KT 5.4 Standardization of major components

This KT refers to the degree of major component standardization, which has an impact on long term cost and component/spare parts replacement availability. It may also be important that the standards used for the design of major systems are also applicable to, or at least adaptable to, local standards adopted by other industries in the Member State. The standardization of major components is important as it enables sharing information with other users. Another important aspect is that if a major component is standardized, obsolete design or discontinuity issues are prevented, because an enhanced design and technology will be introduced and made available. The nuclear specific components or nuclear grade components are usually not off-the-shelf products but are specifically designed and manufactured. Therefore, to find standardization of nuclear specific components may be challenging compared to general components. Some NPP vendors are also manufacturing some of the NPP components. In that case, they may tend to use their own products instead of the best standard products available in the market.

SMRs are smaller than WCRs and the capacities of the components are likely to be smaller than those used in large WCRs, which may benefit SMRs where conventional components are available off-the-shelf for thermal steam power plants. On the other hand, many SMR designs are of innovative technologies, which require special first of a kind engineering components.

The following questions may support a collection of information relevant to this KT:

— To what extent are the major structures, systems, components and materials standardized to form a standard plant design?
— What is the experience of availability of replacements of standard material, components and equipment?

— What long term assurance of supply can be given for standardized equipment that needs to be replaced?
— To what extent can standardized equipment and components of the nuclear power plant be adapted to suit local conditions?
— Is there a track record of malfunctions of the major system components? Is there a user group? Is there a vendor database?
— What is the rationale to select a particular component, especially in case that it is a plant vendor's component?
— Do you employ special components like specially designed pumps, valves and other precision mechanical equipment in order to enable innovative design?

KT 5.5 Power requirements from the grid under normal operation

This KT addresses the differences between the assessed designs in regard to their dependency on off-site electrical power under normal operation, and their ability to tolerate power interruptions and outages of varying time. Load following is addressed in KT 4.8. In addition to the requirements from grid codes as discussed in KT 5.2, the following are the requirements to be addressed within this KT:

— Capability to supply reactive power that contributes to grid stability, but is not able to be supplied by either solar power or wind turbines;
— Inertia of generator/turbine, which mitigates grid disturbance caused by a grid accident, which is not able to be obtained by either solar power or wind turbines;
— Capability of a black start.

The following questions may support a collection of information relevant to this KT:

— How does the NPP respond to grid frequency fluctuations, or short term grid power interruptions?
— What is the NPP's ability to house load the power station (% full power, duration, complexity to switch under blackout conditions) during a long blackout?
— What is the short term load rejection capability (e.g. via turbine bypass in a Rankine cycle) of the NPP without shutdown?
— What are the requirements for the emergency power systems on the grid?

KT 5.6 Ability of the grid to accept generating capacity

This KT refers to the Member State's, or regional electrical grid to accommodate the additional generating capacity of the planned NPP. The challenge of added capacity is a function of the overall capacity, the modularity of the NPP site and the ability of the reactor technology under consideration to load follow.

Since SMRs are smaller in electrical capacity than gigawatt sized WCRs, this KT is of low importance for SMRs. In the SMRs, their capacities can be increased by adding one unit by one unit, and therefore there is more time available to improve the grid's ability to accept generating capacity. The improvement of the grid's ability includes installation of a backup power source, and the necessary capacity of the backup power source can be smaller for the SMRs as well.

The following questions may support a collection of information relevant to this KT:

— What are the grid interface requirements and expectations?
— Over what practical range of capacity is the plant designed to load follow?
— How much additional capacity is needed for the backup power source in the grid system to accommodate the additional generating capacity for the planned NPP site?

5.6. BALANCE OF PLANT DESIGN FOR PURPOSES OTHER THAN ELECTRICITY PRODUCTION

KE6 considers the NPP’s capacity and compatibility with the requirements for non-electric production, such as industrial process heat, district heating, hydrogen production and/or seawater desalination, which could differentiate among the technologies under consideration, based on the following eight KTs:

KT 6.1 Net thermal efficiency;
KT 6.2 Compatibility with local use requirements;
KT 6.3 Protection against internal and external hazards;
KT 6.4 Standardization of major components;
KT 6.5 Electrical power requirements;
KT 6.6 Demand following and storage capabilities;
KT 6.7 Maximum output capacity (heat equivalent and quality);
KT 6.8 Integrated energy systems.

Description: The interface between the BOP design, site and the non-electric production facilities or systems, including normal operation, off normal operation, demand fluctuations, including external hazards and upset conditions on the electrical grid and combinations thereof.

The following IAEA documents can provide further guidance on importance and weighting that could be appropriate to this KE:

- IAEA Nuclear Energy Series No. NP-T-4.1 Opportunities for Cogeneration with Nuclear Energy [52];
- IAEA Nuclear Energy Series No. NP-T-1.17 Guidance on Nuclear Energy Cogeneration [53];
- IAEA Nuclear Energy Series No. NP-T-4.3 Industrial Applications of Nuclear Energy [54];
- IAEA-TECDOC-1885 Nuclear–Renewable Hybrid Energy Systems for Decarbonized Energy Production and Cogeneration [55];
- IAEA Nuclear Energy Series No. NR-T-1.18 Technology Roadmap for Small Modular Reactor Deployment [56];
- IAEA Nuclear Energy Series No. NR-T-1.24, Nuclear–Renewable Hybrid Energy Systems [57].

Large WCRs and SMRs differ in size and power output. Besides this there are several other differences of relevance to this KE. At the time of this publication, an SMR design has started commercial operation in a floating power unit, a high temperature gas cooled reactor (HTGR)-type SMR is preparing for startup commissioning and an integral PWR type SMR is finalizing construction. Several others are in different stages of licensing aiming for deployment by 2035, yet many are still under development. On the other hand, large advanced WCRs have been in operation since the mid-1990s and are built based on the operational experience of the previous generation NPPs over the last 60 years. SMRs promise improved economics and safety that can be reached through standardized designs that are factory built.

Large WCRs also experience standardization. The size of these reactors does not allow for factory fabrication and can be expected to be more tailored to site and environment specific requirements. The factory fabrication of future SMRs will most likely not allow much flexibility in the reactor designs. SMRs are further proposed to come in integrated designs so that nuclear and conventional islands are not separated as is the case with large WCRs, particularly PWRs. The integrated designs of SMRs can make the largest component of the SMR larger or equally large in volume or weight to the largest component of a large WCR. This is relevant for transportation and site selection.

The eight KTs are described as follows and examples are provided in the Annex. The eight KTs apply to large WCRs and SMRs in the same manner. The differences among them are captured in the suggested questions.

KT 6.1 Net thermal efficiency

This KT refers to a NPP's net thermal efficiency, defined as the ratio of MW equivalent output from the BOP (as direct heat, hydrogen or synthetic fuel heating value, etc.) to the core thermal power (in MWth). A higher thermal efficiency directly results in improved fuel utilization.

The considerable increase (roughly 1/3) of net thermal efficiency for nuclear process heat versus electricity production is one of the main arguments for — other than electricity production — using NPPs. SMRs show other possibilities to WCR technology and may be more applicable for purposes other than electricity production due to their smaller size and greater flexibility in power production.

The following questions may support a collection of information relevant to this KT:

- What is the net thermal efficiency for the proposed facility, for the intended application?
- What is the net thermal efficiency for the proposed site and environment?
- Does the net thermal efficiency change over the plant's lifetime; for instance, through the degradation of components?
- What site and application specific factors affect the net thermal efficiency?
- Can the net thermal efficiency be improved through additional cooling or other upgrades?
- How do different operational modes (base load operation and load following operation) affect the net thermal efficiency of the plant?
- What is the fuel burnup of the plant?

KT 6.2 Compatibility with local use requirements

Licensability of cogeneration systems requires great attention to what is possible or not based on decisions to be made by the user in consultation with the vendors. Licensing the industrial plant as part of the overall licensing process of the nuclear cogeneration system is considered a complex undertaking for both newcomer countries with limited experience of regulations as well as operating countries having stringent regulatory processes for licensing NNPs. There are various potential schemes for licensing nuclear cogeneration systems. Each may have advantages and disadvantages. The licensing process for nuclear cogeneration systems becomes more complex if these systems are considered in a similar fashion to the licensing process of conventional power plants (i.e. the industrial plant is to be considered as an integral part of the NPP and undergoes a similar licensing process). In a few countries, the licensing process of coupled industrial facilities to operating NPPs is considered similar to the licensing of any other industrial facilities since the coupling of the two systems does not jeopardize the overall safety of NPPs. However, other countries, based on their regulations, may insist on considering the industrial facility an integral part of the NPP, hence its licence after full safety analysis of the overall cogeneration plant is performed.

The power size of an SMR's module (unit) is equivalent to that of many ageing fossil fired power plants in many countries. In this regard, SMRs would be a viable option for replacement (potentially even on the same site) of ageing fossil fired power plants, thus reducing local pollution and contributing to mitigating climate change (e.g. CO_2 reduction).

The output of the BOP will need to comply with local regulations, usage requirements and quality of the product (e.g. drinking water purity, process heat temperature and quality, etc.), continuity of production and storage availability. Both large WCRs and SMRs need to fulfil the local user requirements. If these requirements are not met, the NPP needs to be redesigned.

The following questions may support a collection of information relevant to this KT:

- What are the interface requirements and expectations for the proposed facility (steam temperature, pressure, quantity, availability, etc.)?
- What is the expected lifetime of the non-nuclear process plant(s)?
- What is the expected lifetime of the nuclear plant(s)?

— What national nuclear and non-nuclear regulations need to be considered?
— Has a similar facility already been erected following these regulations?
— Has a similar facility already been erected following different regulations?
— In what increments can additional power units be added to the existing system?

KT 6.3 Protection against internal and external hazards

The NPP, as well as the coupled non-nuclear plant need to be adequately protected against external hazards (seismic events, storms, flooding, rain, snow, etc.). Depending on the coupling of the BOP to the NSSS, the NSSS itself could be considered an external hazard to the non-electric BOP. For example, when coupling a NPP to a hydrogen plant, the reciprocal effect in the case of accident in either plant may affect the other one. In the case of using cogeneration for desalination, the loss of the desalination plant might pose a safety issue for the NPP as loss of load.

With regard to cogeneration systems (i.e. combined nuclear and chemical facilities) and apart from their own specific categories of hazards, a qualitatively new class of events will have to be taken into account, which is characterized by interacting influences. Arising problems to be covered by a decent overall safety concept are the question of safety of the NPP in the case of a flammable gas cloud explosion or tolerable contamination by a transition of radioactive substances into the product gas. In addition, there are the comparatively more frequently expected situations of thermodynamic feedback in the case of a loss of heat source (nuclear) or heat sink (chemical). Potential hazardous events in connection with a process heat application system extensively investigated are:

— Fire and explosion of flammable mixtures;
— Atmospheric vapour cloud explosion in the vicinity of the reactor;
— Ingress of flammable gases into the reactor building;
— Tritium transportation from the core to the product (e.g. hydrogen, methanol);
— Thermodynamic interaction between nuclear and chemical plant;
— Isolation of desalination plant.

For the case of cogeneration for hydrogen production from NPPs, there are two significant safety issues originating in the thermochemical hydrogen production system to be coupled to the HTGR. One is hydrogen release and the other is toxic gas release. A basic safety design approach is to prevent accidental release of their materials and to mitigate their effect on the HTGR safety items and operators. Provision of separation distance between the HTGR and the hydrogen production system is a simple and reliable safety approach. But a long separation distance requires long helium piping and a larger plant site, which results in an increase of the plant’s overall economics.

Protection against external hazards relates to the safety of a plant and thus it is of highest importance. SMRs show greatly reduced radioactive inventory and usually have much higher safety margins due to their lower power density so that this KT carries slightly less weight for SMRs.

The following questions may support a collection of information relevant to this KT:

— What external hazards are considered in the standard design by the technology holder? Are these considerations compatible to the site and environment specific requirements?
— What external hazards need to be considered by the owner/operator organization? Are these different from the external hazards considered in the standard design?
— Are additional upgrades required to comply with the national regulations of the owner/operator organization?
— How are upgrades following changing regulations to be integrated once the plant has been built?
— What external hazards are not considered in the standard design of the technology holder?
— What are the reciprocal effects in the case of accident in either plant (NSSS and BOP) and how are the effects on the other considered?

KT 6.4 Standardization of major components

This KT addresses the extent to which the major components of the NPP can be made to an established standard. Standardized components increase safety and decrease costs. This KT is thus of importance for both large WCRs and SMRs. It may be of even higher importance to SMRs from an economic perspective. Power production with SMRs can be economic if these SMRs can be built in large numbers. This is only possible through standardization.

The following questions may support a collection of information relevant to this KT:

— To what extent are the major components standardized?
— What is the experience of availability of replacements of standard material, components and equipment?
— What long term assurance of supply can be given for standardized equipment that needs to be replaced?
— How many similar plants are in service, under construction and committed to construction?
— What is the level of experience in information sharing between users of the standardized plants?
— To what extent has standardization of equipment and components of the nuclear power plant been addressed in the licensing process?

KT 6.5 Electrical power requirements

This KT is concerned with the electrical power requirement of the NPP during standard operation and safe shutdown in the case of a blackout. Electrical power requirements affect the safety of the NPP and thus present the highest importance to both large WCRs and SMRs. Many SMRs have passive cooling capabilities in case of emergency shutdown that are made possible by the lower power density of the reactor core; a feature that is not given in large WCRs that will always require external energy for a safe emergency shutdown. This KT is thus more important for large WCRs than it is for SMRs.

The following questions may support a collection of information relevant to this KT:

— Does the plant have its own generation capacity to continue operation under blackout conditions?
— What power generating systems are available in case of an emergency shutdown under blackout conditions?
— What is the response time of those systems and for how long can they supply the electrical power requirements?
— What electrical power is required to continue operation under blackout conditions?
— What electrical power is required for safe shutdown under blackout conditions?
— What happens if this power is not supplied?
— Can the NPP operate independently with its own local power grid? What additional infrastructure would be required if any to have the NPP operate at a remote mining site?

KT 6.6 Demand following and storage capabilities

For purely non-electric application, this KT refers to the manoeuvrability with regards to non-electric demand from a single nuclear unit and storage capabilities of the BOP in the form of the product (e.g. hydrogen, desalinated water, etc.). For cogeneration, this KT includes flexibility in electricity load following and optimizing measures in the design and operation of the cogeneration systems for better economics, such as measures to recover waste heat or use of off-peak power for non-electric applications. Demand following (as a 'flexible operating plant' addressed in KT 6.8) capabilities are equally important for large WCRs and SMRs. It is relevant to realize though that large WCRs can have a considerably higher relative weight in a national energy grid (particularly in a newcomer country) than SMRs.

The following questions may support a collection of information relevant to this KT; they need to be adjusted, or made more specifically, depending on the type of cogeneration or purely non-electric application:

— How does the plant respond to fluctuations, or short term demand interruptions?
— What are the operational margins?
— What are the possibilities for load follow and related operational manoeuvrability versus standard base load?
— What storage technology and capacity are available?
— How efficient is the storage?
— What are the emergency remote shutdown requirements and capabilities?
— What are the fuel power and ramp rate limit constraints?
— What are the conditions/requirements for heat-up and startup, power ascension, reactor normal shutdown and cooldown?
— What are load rejection requirements versus capability?
— What are the steam bypass system requirements versus capacity?
— What are the load following capabilities and what is the experience of load following operation?
— What is the load rejection capability without shutdown?
— What are the operational margins during normal full power and load follow operation?
— What are the capabilities for remote shutdown?
— How much increase in waste generation will be caused by load following (e.g. boron inventory management considerations)?
— Are there any concerns on fuel mechanical conditioning and capability during load following (e.g. pellet–cladding interaction)?

KT 6.7 Maximum output capacity (heat equivalent and quality)

This KT refers to a NPP's overall output capacity, expressed in MW equivalent output from the BOP (as direct heat, hydrogen or synthetic fuel heating value, etc.). The maximum output capability of an SMR is more important than the maximum output capability of a large WCR since it is likely that more than one SMR unit will be built and some SMRs can take advantage of jointly used infrastructure, while this is not necessarily the case for large WCRs.

The following questions may support a collection of information relevant to this KT:

— What is the overall output capacity, expressed in MW-equivalent under local conditions? In what increments can this overall output capacity be increased?
— What BOP enables the highest overall output capacity?
— Can different BOPs be combined (e.g. hydrogen production, desalination)?
— Can the NPP be upgraded for higher output capacities?
— What is limiting the current output capacity?

KT 6.8 Integrated energy systems

This KT refers to the NPP being part of an integrated energy system as a 'flexible operating' plant, such as a nuclear–renewable hybrid energy system. Integration of the NSSS with the non-electric BOP or with other plants of a locally integrated energy system can also entail shared facilities (when NPP and other cogeneration plants use intake, outfalls, O&M services, etc.).

Large WCRs and SMRs can both be part of integrated energy systems. Due to the large power output and the limited temperatures that can be reached in large WCRs it is more likely that SMRs will fulfil this role.

Supporting publications:

— IAEA-TECDOC-1885 Nuclear–Renewable Hybrid Energy Systems for Decarbonized Energy Production and Cogeneration [55];
— IAEA Nuclear Energy Series No. NR-1.18, Technology Roadmap for Small Modular Reactor Deployment [56];
— NICE Future Nuclear Innovation: Clean Energy Future, Flexible Nuclear Energy for Clean Energy Systems, Clean Energy Ministerial [58].

The following questions may support a collection of information relevant to this KT:

— Can the NPP be integrated in a larger energy system with other intermittent energy sources, such as wind, solar power, etc.?
— What combinations of the NPP with renewable energy sources are possible/advised to accomplish the most efficient energy production?
— What renewable energy sources are available at the site?
— What happens if parts of the integrated system are unavailable? Can the NPP, the renewable system and a potential process plant operate independently? How is this accomplished?
— How can such an integrated system be up- or down-scaled to meet changing national energy demands?
— What are the lifetimes of the different components of the integrated systems?
— What experience exists with integrating the NPP with other energy producing plants?
— What experience exists with integrating the NPP with processing plants?

5.7. SAFEGUARDS AND PROTECTION

KE7 considers safeguards and NPP and site protection, which could differentiate among the technologies under consideration, based on the following four KTs:

KT 7.1 Safeguards by design;
KT 7.2 Special nuclear materials management;
KT 7.3 Physical protection of the NPP;
KT 7.4 Cybersecurity protection of the NPP.

Importance rationale (Table 7): The large WCR designs may not differentiate from the safeguards point of view. However, this aspect may be a differentiator for the SMRs, and especially for the first of a kind (FOAK). Although there may be differences in the details of the security plan and systems, it is expected that site security will be achieved by the responsible authorities.

Description: This KE addresses the safeguards and prevention and detection of, and response to, theft, sabotage, unauthorized access, illegal transfer or other malicious acts involving nuclear material, other radioactive substances or their associated facilities.

The following IAEA documents can provide further guidance on importance and weighting that could be appropriate to this KE:

— IAEA Services Series No. 21, Guidance for States Implementing Comprehensive Safeguards Agreements and Additional Protocols [59];
— IAEA Nuclear Energy Series No. NP-T-2.9, International Safeguards in the Design of Nuclear Reactors [60];
— IAEA Services Series No. 33, Safeguards Implementation Practices Guide on Provision of Information to the IAEA [61];

— IAEA Nuclear Security Series No. 19, Establishing the Nuclear Security Infrastructure for a Nuclear Power Programme [62];
— IAEA Nuclear Security Series No. 27-G, Physical Protection of Nuclear Material and Nuclear Facilities [63];
— IAEA Nuclear Security Series No. 10, Development, Use and Maintenance of the Design Basis Threat [64];
— IAEA Nuclear Security Series No. 25-G, Use of Nuclear Material Accounting and Control for Nuclear Security Purposes at Facilities [65];
— IAEA Nuclear Security Series No. 17, Computer Security at Nuclear Facilities [66].

The four KTs are described as follows and examples are provided in the Annex.

KT 7.1 Safeguards by design

Safeguards by design (SBD) can be explained as: "IAEA safeguards are a central part of international efforts to stem the spread of nuclear weapons. In implementing safeguards, the IAEA plays an independent verification role, which is essential for ensuring that States' safeguards obligations are fulfilled" [60]. They comprise an extensive set of technical measures by which the IAEA independently verifies the correctness and the completeness of the declarations made by Member States about their nuclear facilities, material and activities.

In general terms, IAEA safeguards activities are performed to verify the State's declarations about nuclear material quantities, locations and movements at a facility such as an NPP [60]. SBD refers to NPP design features that are incorporated at the reactor design stage to facilitate the implementation of these IAEA safeguards monitoring and verification activities. Verification consists of basically two types, verification of design information through on-site physical examination during the construction and subsequent phases of the facility's life cycle against the design, and verification of the nuclear material accountancy during fuelled operation.

While safeguards implementation potentially has a small impact on project cost and schedule when considered early in the design process, failure to do so can result in a much larger impact than necessary, both in construction and during operation. Section 3.2 of [60] gives details of SBD for each design phase, while Section 4 lists design features that assist in the implementation of safeguards. Reference [61] gives further guidance that can assist in evaluating a new design's readiness for safeguards implementation.

As a general rule, the more that an SMR's core and fuel management design deviates from traditional NPP practice, the more important SBD will be. This arises because the innovative designs, in addition to requiring the same generic safeguards considerations as other new NPPs, will generally also require an entirely new safeguards approach. This will necessitate additional analysis by the IAEA, and potentially new verification techniques and equipment involving time and often the R&D resources of the Member States (including the designer). Furthermore, during this process it may arise that certain design modifications can significantly increase the efficiency and/or effectiveness of the safeguards approach, so the more discussion there is with the IAEA (or other safeguards experts) at an early stage of development, the better.

As another general rule, since IAEA safeguards generally involve independent verification of nuclear material inventory and flow, the complexity of a safeguards approach will tend to increase with increasing inventory and flow, and with decreasing discreteness and distinctness of fuel items. Historically, reactor fuels have been relatively large and distinct (i.e. identifiable) objects that could be verified either visually or with standard radiation detection equipment in a straightforward manner. New SMR designs with pebble, slurry or liquid fuel will present a conceptually similar challenge to IAEA safeguards to that found historically in fuel fabrication and reprocessing facilities, where verification might require additional chemical and statistical analysis, or other inference techniques that rely upon IAEA installed or operator equipment. Some SMR designs involve factory sealed cores that may involve safeguards measures applied in the supplier State, which adds another level of complexity.

Clearly, SBD will play a significant role in meeting these safeguards challenges with a level of efficiency that minimizes both burden to the operator and resources of the IAEA.

How to assess technologies: The IAEA safeguards are embedded in legally binding agreements and provide the basis for the IAEA to implement effective verification. IAEA safeguards are applied to materials and facilities placed under IAEA safeguards by a State or States. Therefore, this KT will ideally not be a differentiator for well established NPP technologies that have operating units under safeguards elsewhere but could be an important consideration when evaluating evolutionary or revolutionary NPP concepts or designs. In the latter case, it would be highly desirable if the concept is covered under existing IAEA safeguards principles and procedures, or at least that it has been discussed with IAEA safeguards department experts.

Additional assessment subtopics may be added by the RTA team, such as, for example, the following subtopics supporting a specific assessment of the designs. Suggested questions to collect more information are also provided:

— Ease of design verification during construction:
  - Do construction plans include specific, agreed phases for IAEA safeguards design verifications?
  - Are relevant areas, such as shipping/receiving areas, fresh fuel storage, fuel transfer corridors, the core and spent fuel storage easily monitored?

— Consideration of IAEA safeguards equipment installation and power requirements:
  - What specific features are included in the design to facilitate installation of cameras, counting systems and radiation detectors, and their access for maintenance?
  - Does the design include adequate physical space, stable uninterruptible power and secure data transmission for IAEA safeguards equipment?
  - To what extent is the IAEA Safeguards department consulted in the facility design?

The following questions may support a collection of information relevant to this KT regarding the assessment of the SMR designs:

— How are IAEA safeguards incorporated in the SMR modular construction?

— Does the design allow for remote transmission of information?

KT 7.2 Special nuclear materials management

What are the special nuclear materials (SNMs): The IAEA safeguards verification of SNMs accountancy consists primarily of defect measurements on fresh or irradiated fuel at the reactor, where defect means missing or diverted. It can perform item counting and identification checks, or it can apply gross defect measurements to irradiated fuel when it is transferred during fuelled operation. Surveillance, containment and monitoring activities supplement the nuclear material accountancy measures by providing the means to detect undeclared access to, or movement of, nuclear material or safeguards equipment. Containment refers to the structural components that make undetected access difficult. Seals are tamper indicating devices used to detect tampering or unauthorized entry in containment and surveillance is the collection of optical or radiation information through human and instrument observation/monitoring [60].

How to assess technologies: IAEA safeguards are embedded in legally binding agreements and provide the basis for the IAEA to implement effective verification. IAEA safeguards are applied to materials and facilities placed under IAEA safeguards by a State or States. Therefore, this KT is not expected to be a differentiator for well established NPP technologies that have operating units under safeguards elsewhere but could be an important consideration when evaluating evolutionary or revolutionary NPP concepts or designs. In the latter case, it would be highly desirable if the concept is covered under existing IAEA safeguards principles and procedures, or at least that it has been discussed with IAEA safeguards department experts.

Additional assessment subtopics may be added by the RTA team, such as, for example, the following subtopics supporting a specific assessment of the designs. Suggested questions to collect more information are also provided:

- Ease of SNM verification during fuelled operation:
  - What implications arise from the fuel design on the complexity of implementing IAEA safeguards equipment and activities? Is major R&D required?
  - Does the design include adequate material balance areas, key measurement points and compartmentalized space?
  - Are provisions of cabling and penetrations included in the design to accommodate future safeguards equipment upgrades?
  - To what extent is the IAEA Safeguards department consulted in the facility operating procedures?
- Provision for remote monitoring of operating parameters and operational procedures:
  - What specific features are included in the design to facilitate desired continuous monitoring of reactor power levels and other important operating parameters needed to verify fuel loading burnup?
  - What specific features are included in the design to facilitate detection and monitoring of significant reactor operations that could affect SNM accounting and control?

The following questions may support a collection of information relevant to this KT regarding the assessment of SMR designs:

- What implications arise from sealed manufactured fuel on the complexity of implementing IAEA safeguards equipment and activities?
- What implications arise from the transport, handling and refuelling times of nuclear materials on the complexity of implementing IAEA safeguards activities?

KT 7.3 Physical protection of the nuclear power plant

This KT addresses how the physical protection measures differ among the assessed NPP designs. It is not expected to be a strong differentiator unless the security provisions are significantly deficient or will not be amended to conform.

Physical protection addresses different measures taken for the prevention and detection of, and response to, theft, sabotage, unauthorized access, illegal transfer or other malicious acts involving nuclear materials, other radioactive substances or their associated facilities. Although there may be differences in the details of the NPP physical protection and security plan and systems, it is expected that site security is design specific and will be mainly achieved by the responsible authorities.

Additional subtopics may be added by the RTA team, such as, for example, the following subtopics supporting a specific assessment of the designs in Phase 2 when confidential information can be obtained. Suggested questions to collect more information are also provided:

- Security plans: Evaluation of all physical security systems is performed under a confidential process, independent of the rest of the technology assessment:
  - Describe the security plans.
- Diversity and redundancy of security facilities:
  - Describe the levels of diversity and redundancy of security facilities.
- Security access to the buildings and related security facilities design against security related threats:
  - Describe the security access to the buildings and related security facilities design against security related threats;
  - What programmes and facilities for site security are provided?

  - What is the design basis threat(s)(DBTs)?
  - Is it relevant to the Member State's DBT for nuclear facilities?
- Integrated NPP access control system: to include in the general NPP design, such as but not limited to perimeter fences and roads, perimeter detection systems, closed circuit television control and recording systems, security dedicated lighting systems, supervisory security systems:
  - Describe the features for physical protection of plant systems;
  - How is an attempted malicious act managed by an integrated system of detection, delay and response?
  - How are the consequences of a malicious act mitigated?
  - What additional measures are taken to minimize insider threats?
  - Describe the dedicated security communication system with external support services such as but not limited to police, fire, emergency medical, regulatory and government agencies.

The following questions may support a collection of information relevant to this KT regarding the assessment of SMR designs:

- Describe additional security measures for land based (above ground and below ground) or floating based SMR NPP.
- What are the security measures for transporting more and smaller fuel assemblies? What are the security measures taken for remote SMR locations?
- Is there enough staff to implement the security plans?

KT 7.4 Cybersecurity protection of the nuclear power plant

Cybersecurity refers to the measures taken to protect computer based systems, networks and other digital systems that are critical for the safe and secure operation of the facility and for preventing theft, sabotage and other malicious acts.

Computer based systems refer to the computation, communication, instrumentation and control devices that make-up functional elements of the nuclear facility. This includes not only desktop computers, mainframe systems, servers, network devices, but also lower level components such as embedded systems and programmable logic controllers. IAEA Nuclear Security Series No. 17, Computer Security at Nuclear Facilities, provides more details on the specifics of the computer security designs in NPP. Computer based systems used for physical protection, nuclear safety, and nuclear material accountancy and control will ideally be protected against compromise (e.g. cyberattacks, manipulation, falsification) consistent with the threat assessment or design basis threat as outlined in IAEA Nuclear Security Series No. 13 on Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities [67].

The terms computer security and IT security are commonly used synonyms for cybersecurity. Thus, cybersecurity is basically concerned with all components that may be susceptible to electronic compromise of sensitive information. Cybersecurity is considered a subset of information security, which has the overarching role of taking the appropriate measures to ensure the confidentiality, integrity and availability of information. Although there may be differences in the details of the cybersecurity protection of a NPP and its security plan and systems, it is expected that cybersecurity will be achieved by the responsible authorities. This KT is not expected to be a strong differentiator unless the security provisions are significantly deficient or will not be amended to conform. This KT is the same for assessing large WCR or SMR designs. However, for the same level of power, more SMR systems need to be protected.

Suggested questions that may support assessment, in more detail specifically in Phase 2 when confidential information can be obtained, are as follows:

- What are the access control measures adopted for critical safety and security systems?
- What are the measures taken to ensure the confidentiality, integrity and availability of information?

— How are the critical safety and security systems completely isolated from the internet? How effective is the isolation?
— How are the support systems, and equipment and emergency preparedness functions protected from cyberattacks?
— What are the control measures for portables devices and media?
— What is the plan to periodically assess the vulnerability of critical safety and security systems and threats, including DBT, from cyberattacks?
— What additional measures are taken to minimize insider threats?

5.8. TECHNOLOGY READINESS

KE8 considers technology readiness of the NPP design at a high level, which could differentiate among the technologies under consideration, based on the following three KTs:

KT 8.1 Readiness of the SMR design;
KT 8.2 Licensing and/or certification status for the design;
KT 8.3 Language.

Importance rationale (Table 7): Verification of proven technology is important in a complex system for its long term, safe, economic and reliable operation. This KE has high importance and specifically may be a differentiator among the FOAK SMR designs.

Description: Technology readiness reflects a significant level of experience through operation of a certain component, system or entire NPP for a certain length of time, demonstrating the capabilities of those technologies. While there are several tools and frameworks available to perform a full technology readiness level assessment (e.g. DOE Guide 413.3-4A Technology Readiness Assessment Guide [68]), the RTA here is limited to the qualitative assessment over three KTs. The currently available large WCRs are ready for the market, as most of them have completed their design and licensing or are already under construction or in operation. Technology readiness, however, can be a significant differentiator among new SMR technology holders. Therefore, KT 8.1 applies to SMRs only, while the others apply to all NPP types.

Supporting publications:

— Nuclear Safety Infrastructure for a National Nuclear Power Programme Supported by the IAEA Fundamental Safety Principles, INSAG Series No. 22 [69];
— Licensing the First Nuclear Power Plant, INSAG Series No. 26 [70];
— Western European Regulators Association (WENRA) Reactor Safety Reference Levels, [26];
— Technology Readiness Assessment Guide, United States Department of Energy, DOE G 413.3-4A [68].

The three KTs are described as follows while examples are provided in the Annex.

KT 8.1 Readiness of the SMR design

This KT generally has high importance for new designs. There are three basic categories of the advanced reactor designs: proven, evolutionary or innovative, as reflected in the IAEA ARIS database [2].

In many advanced or innovative reactor designs, some systems or components may be highly developed and at an advanced or even proven technology level, while others may be less defined, designed or proven. Some systems or components may be proven in other industries (e.g. turbine) but not (yet) in an NPP. The RTA team at a minimum is to consider the main technology areas of an NPP: fuel, NSSS and BOP. Additional assessment subtopics may be added by the RTA team.

Large WCRs mainly use fuel with enriched uranium of below 5% and/or MOX. The SMRs' designs are also mainly based on fuel with enriched uranium of below 5% or MOX, but some may use fuel with enriched uranium approaching 20%. The alloys of fuel assemblies for large WCRs and SMRs can be considered similar. The geometry of fuel assemblies for large WCRs and SMRs can be similar, but with shorter length fuel assemblies for SMRs. There is a lot of experience regarding the design, testing and licensing of fuel assemblies for large WCRs that can support the fuel design in SMRs.

Most SMRs are designed as integral reactors in PWRs; the reactor core, steam generators, reactor coolant pumps, pressurizer and the core internals are integrated within the reactor pressure vessel presenting therefore important differences to large PWRs.

Steam generators of some PWR based SMRs are connected to reactor pressure vessels directly and main pipes are eliminated. In the PWR based SMRs integrating the pressurizer with the steam generator, significant primary piping is eliminated. The volume and weight of steam generators or core in PWR based SMRs are both very different when compared to large PWRs.

In some SMR designs, penetrations of the reactor vessel are eliminated by incorporating the control rod drive mechanism inside the pressure vessel, thus reducing rod ejection accidents. Integral reactor configuration and compact layout of system and components in many SMR designs almost eliminates the classic large loss of coolant accident. For some SMRs, small cores inside large vessels allow for an easy in-vessel retention strategy. In the SMR designs the safety systems rely more on passive safety features, which ensure safe shutdown and decay heat removal for an unlimited period without the need for external power, make-up water or operator's actions. Therefore, SMRs can have a longer grace period and coping time without operator intervention.

The containment of some SMRs is submerged in a water pool. This is safety related in preventing containment from exceeding the design pressure and temperature by cooling its outside surface. The large water pool provides long term passive cooling without an external heat sink. Some SMRs have steel containment located below the ground level, which reduces the risk of external events and improves the ability to withstand natural disasters. Buried or semi buried design for nuclear island building is one of the strengthening measures against extreme events.

The BOP design for large WCRs and SMRs can be similar. The steam turbo generator is like that of a fossil fired power plant. Compared to those of large WCRs, the SMRs and fossil fired power plants, the electrical capacities are more similar. Because of a larger potential market, many SMRs are also designed for different applications. Their turbines are designed to be capable of different types of extraction and are enabled to meet different levels of heating and industrial steam supply requirements. The number of system components is reduced in SMRs due to a common use of some of them. For example, the common crane can be replaced by a travelling crane, thus being used for both the reactor building and auxiliary building.

The following questions may support a collection of information relevant to this subtopic:

— To what extent does the fuel design have identical features as currently operating or licensed fuel designs?
— Which (how many) other reactors use the same or similar fuel?
— How many years of reactor operating experience have you had for this fuel design?
— What percentage of the detailed design is completed?
— What methodologies and criteria were/will be used to licence the new fuel design?
— What percentage of the procurement specification is completed?
— What improvements on previous and/or existing designs are expected?
— What are the established reactor technologies used?
— For components and systems that are newly designed, provide the rationale that supports the design decision?
— What percentage of the design meets the requirements of the AFCEN (L'Association française pour les règles de conception) or ASME (American Society of Mechanical Engineers) nuclear design codes?

— What percentage of the detailed design is completed?
— What percentage of the procurement specification is completed?
— How is the general experience (design, construction, manufacturing, operation) used in order to support evolution of the design for FOAK and the following reactors?
— For components and systems that are newly designed, provide the rationale and pedigree that supports the design decision.
— What percentage of the detailed design is completed?
— What percentage of the procurement specification is completed?

KT 8.2 Licensing and/or certification status for the design

This KT describes the status of licensing and regulatory frameworks in the vendor's country and the host country affecting the deployment ability of the design. Its importance is high because the licensing and regulatory frameworks can be different between the vendor's country and host country by chosen methodology (e.g. prescriptive, descriptive, risk informed, graded approach) and that may substantially affect the assessment.

As indicated in INSAG Series No. 22 [69] and INSAG Series No. 26 [70], it is expected that a new entrant country will use the reference plant concept for its first NPP unit. Under this approach, an important part of the NPP has the same design and safety features as a NPP already licensed by the regulatory body of the exporting country. This approach would facilitate the licensing process in the new entrant country.

Different large WCR designs are licensed in several countries and have a reference design. This is not the case for SMRs. However, some elements of the design or some of the equipment used in SMRs are already used in large WCRs. To facilitate the deployment of SMRs, common licensing (e.g. pre-licensing, design certification) for the same SMR in different countries is attempted and represents a widely shared aspiration. National regulatory requirements may also consider the inherent concept of standardization of SMRs to facilitate their implementation.

Portions of SMRs' design or some of the equipment are already compliant with regulatory requirements being used in large WCRs, and, for example, Western European Nuclear Regulators Association (WENRA) documents on safety objectives for large WCRs can be used for SMRs. The IAEA safety requirements for design, safety assessment, operation ([19, 21, 43]) are available for large WCRs and can be used for SMRs. Because of design simplifications, specific regulatory requirements for SMRs could be of less volume.

In order to develop an unbiased assessment among the technologies the following aspects that may also be defined as subtasks describe the scope of this KT:

— Regulatory requirements in the host country and the standards applied by the technology holder for the design, including licensing process and issues, recent or ongoing, both in the vendor's country and on the other exported sites, and the language of original licensing and/or certification documents;
— Regulations in the host country on radiation and safety for nuclear power plant siting;
— Licensing and/or certification status for the design, including past, current and anticipated licensing issues and resolutions;
— Compliance with regulatory requirements in the host country;
— Status of regulatory approval of the design in various countries.

The following questions may support a collection of information relevant to this KT:

— Does the design comply with (and to what extent) established local regulatory requirements?
— Does the design comply with regulations on radiation and safety related to nuclear power plant siting in the host country?

— Which regulatory agencies have approved/are approving (and/or certified) this design? What are the ongoing or recent major licensing issues being addressed?
— What is the licensing history of the design in the country of origin or other countries?
— What licensing has been completed for site, construction and operation in other countries?
— What are the plans and status for certification and licensing for this application?
— To what extent has standardization of equipment and components of the nuclear power plant been addressed in the licensing process?
— Which international or national regulations/guidance (including the IAEA safety standards) has the design been assessed against? Does the design make exceptions to any IAEA safety standards?
— Has the design been certified/reviewed by other national or international organizations?
— What are the licensing criteria and approaches that have been taken for this design? Prescriptive, deterministic arguments, probabilistic arguments?

KT 8.3 Language

The licensing and operation of the NPP will be done, most likely, in one of the local official or common languages. Therefore, it is of great benefit if the design information, licensing documents, and in particular all operating, training and maintenance procedures are available in the right language and system of units or can be produced from the vendor. Language during construction is a lesser concern, as most large construction projects deal with different languages, but a common language will ideally be chosen to facilitate information communication with the vendor. For large WCR design information, licensing documents, operating and maintenance procedures are available in different languages. This is not the case for SMRs. Most of the technical vocabulary used for large WCRs can be used also for SMRs.

The following questions may support a collection of information relevant to this KT:

— Does the vendor of technology have experience in foreign countries?
— Can the technology vendor provide existing and used documentation in the local language?

5.9. PROJECT DELIVERY

KE9 considers the ability of the technology holder to deliver the NPP as specified for schedule and cost, based on the following seven KTs:

KT 9.1 Owner/operator scope of supply;
KT 9.2 Supplier/technology holder issues;
KT 9.3 Project schedule capability;
KT 9.4 Technology transfer and technical support;
KT 9.5 Project contracting options;
KT 9.6 Services offered for the front end of fuel cycle (fresh fuel supply);
KT 9.7 Services offered for the back end of fuel cycle (spent fuel management).

Importance rationale (Table 7): This KE is more relevant for Phase 2 when details become available from the technology holders. The main scope is to assess the ability for all aspects of the NPP project including design, construction and operation to be delivered on the committed schedule and cost. The cost itself is considered in KE10.

Description: The assessment of the ability of the technology holder to deliver the NPP as specified for schedule and cost.

The following IAEA publications can provide further guidance on importance and weighting that could be appropriate to this KE:

- IAEA-TECDOC-1750 Alternative Contracting and Ownership Approaches for New Nuclear Power Plants [71];
- IAEA Nuclear Energy Series No. NP-T-2.7 Project Management in Nuclear Power Plant Construction: Guidelines and Experience [72];
- IAEA Nuclear Energy Series No. NG-T-3.9 Invitation and Evaluation of Bids for Nuclear Power Plants [5];
- IAEA Nuclear Energy Series No. NG-T-3.4 Industrial Involvement to Support a National Nuclear Power Programme [73];
- IAEA Nuclear Energy Series No. NP-T-3.21 Procurement Engineering and Supply Chain Guidelines in Support of Operation and Maintenance of Nuclear Facilities [74].

The seven KTs are described as follows and examples are provided in the Annex.

KT 9.1 Owner/operator scope of supply

This KT addresses the responsibilities of the owner/operator to ensure understanding of the obligations and opportunities associated with the successful progress and completion of the NPP. This includes the owner/operator requirements for design, construction and operational startup testing. The assessment of this KT focuses at determining therefore the degree of involvement of the owner/operator. The lack of clear definition of interfaces can lead to significant issues.

It is reasonable that the interface programme and integration between the owner/operator and the technology holder depend on the project size and its complexity. Technology holder expectations from the owner/operator are related to the owner/operator experience with nuclear technology, project size, plant siting, potential FOAK issues (if FOAK is chosen) and the complexity of design. SMRs, due to their inherently modular approach and generally smaller size (less volume and areas occupied), are easier to build and because of that the site preparation, infrastructure and site facilities, typically owner/operator scope, are simplified in comparison with traditional WCRs. On the other hand, SMRs FOAK impact has larger uncertainties and can introduce the need for additional assessment. It also may introduce local licensing risk requiring deeper involvement of the technology holder, which does not exist for standardized *N*th of a kind (NOAK) WCRs.

What may distinguish the technology holders are:

- Owner/operator requirements in design and construction for BOP, site preparation, infrastructure and site facilities, including simulator and other training facilities;
- Impact on owner/operator responsibility;
- Assistance of the technology holders to the owner/operator regarding construction and/or operation licensing documentation and communication to the regulatory bodies;
- Owner/operator oversight of engineering, procurement and construction.

The following questions may support a collection of information relevant to this KT:

- What are the technology holder expectations regarding owner/operator scope?
- What activities will the technology holder manage? What activities does the technology holder not plan to manage?
- What programme is proposed for interface and integration between the owner/operator and the technology holder? What is the human capacity building plan suggested by the technology holder to the owner?

— What owner/operator tasks would the technology holder prefer to assume and what are the proposed cost estimates?
— How does the technology holder propose to limit project risks due to the owner/operator programmes and interface?

KT 9.2 Supplier/technology holder issues

This KT may be of high importance as it addresses the strength of the relationship between the technology holder and its suppliers, including an assessment of the capabilities and history of the suppliers, the duration of the relationship, and any quality or schedule issues or advantages based upon current data or relevant experience record. This KT is of high impact especially when assessing different FOAK designs.

Regarding the potential supplier/technology holder issues, there are two key aspects, which can address the strength of the relationship between the technology holder and its suppliers:

— From standardization to modularity learning approach. Learning approaches for NPPs rely upon a technical concept that includes the supply of standardized components (highly standardized quality assurance and control) and their assembly and maintenance within the NPP site, with a reduction of investment and operating costs. The achievement of high standardization of NPP components is a necessary condition for both technologies, either WCRs or SMRs. However, the smaller size of units is an easier concept for a potential supplier not just to replicate in factory production but also to raise the factory made share in comparison with the site erected share of the value chain. Since systems are simpler and components are smaller, it is possible to manufacture preassembled structural or electromechanical modules in factories. This allows one to maintain the quality of products, satisfy the technology holder's schedule by completing pretesting in factories and thus to reap the learning economies.
— Mass production approach. For a given installed power, many more SMRs than large WCRs are required. Therefore, it is possible to have a large bulk ordering process of components from the technology holder to its suppliers (such as valves, indicators, etc.) for SMR technology in comparison with large WCR technology. This aspect will allow the SMR suppliers to exploit the economies of mass production and a more standardized procurement process in the future if suppliers are supposed to remain the same all along the completed series of NPPs.

The current relevant experience related to capability and history of suppliers of (more or less standardized) NOAK plants gives slight advantages to WCR technology with already known/proven near and long term suppliers for key components and parts and long term supply chain assurance. However, the FOAK plants, either SMRs or WCRs, cannot exploit mass production experiences for part components and technological solutions not used before.

The assessment of various technology holders' abilities to deliver the NPP as specified is based on a comparison of the technology holders' scope of supply, including programmes on quality, subcontractor relationships, personnel assignments, employee programmes, safety practices and record, and process and schedule controls. Some or all of the following issues being of relevance to the RTA team may be assessed as the sub-KTs:

— Responsibilities: related to the technology holder's scope of supply, including programmes on quality, subcontractor relationships and key personnel assignments;
— Experience: of key personnel;
— Supply: near and long term suppliers for key components and parts, and long term supply chain assurance;
— Warranties: corrective action programme; equipment commercial grade dedication; industrial safety programmes and achievement record;

— Quality control (QC) programmes;
— Risk: sharing and means of dispute resolution.

The following questions may support a collection of information relevant to this KT:

— What kind of delivery contracts would you require/offer?
— What supply chain arrangements have you used/will you use in the project?
— What are the obligations concerning bid invitation?
— What partnerships have been established or will be established to support this project?
— Who takes the risk with regard to the assurance of supply of components and parts?
— How will the architect/engineer/technology holder handle QA for lower level/domestic components?
— How will export control issues of sensitive technology be addressed?

KT 9.3 Project schedule capability

This KT is of medium importance. It describes the ability for all aspects of the NPP project including design, construction and operation to be delivered on the committed schedule. The scopes of the assessment will ideally include:

— Schedule for procurement of long lead time items and site preparation, with critical path and contingency;
— Integrated project schedule and experience base, including engineering, licensing, procurement, construction and startup;
— Scheduling tools and software for scheduling and analyses;
— Impact of local conditions (e.g. accommodation for labour laws, weather, transportation infrastructure).

The current integral and modular approach to the design of new nuclear reactors, either WCRs or SMRs, offers the unique possibility to exploit a simplification of the NPP construction because of the reduction of the type and number of components. Simplification of NPP and reduced number of active safety systems can decrease the time necessary for design, engineering, procurement and startup process.

The ability to convert the design of a NPP to a factory for fabrication of modules, shipment and installation in the field as complete assemblies has good potential for reducing the integrated project schedule. SMRs can take better advantage of this, since it is possible to have a greater percentage of factory made modules (presuming that SMRs are similar to WCRs in complexity) transported to decrease construction time in the field. However, the construction time is just a part of an integrated project schedule. Delivery contracts, which define the committed schedule, generally are not dependent on the size of NPP (e.g. the licensing process is typically on a critical path and riskier for FOAK design, whether SMRs or WCRs). Generally, project delays are incurred due to the project size, FOAK issues and the complexity of design. SMRs, due to their inherently modular approach, are easier to build and, because of their smaller size, the FOAK impact on cost escalation has a limited effect.

The following questions may support a collection of information relevant to this KT:

— Do you have a reference schedule based on other projects?
— Do you have a detailed schedule, month by month?
— Do you know the impact of potential events (weather situation) on the schedule?
— In the case of events modifying the schedule, what adaptations can you offer?
— What are the benefits for the programme of building multiple units on the same site?
— What kind of delivery contracts would you require/offer?
— What supply chain arrangements have you used/will you use in the project?
— What partnerships have been established or will be established to support this project?
— Who takes the risk with regard to the assurance of supply of components and parts?

— How will the architect/engineer/technology holder handle QA for lower level/domestic components?
— How will export control issues of sensitive technology be addressed?

KT 9.4 Technology transfer and technical support

This KT is of high importance for both large WCRs and SMRs. The assessment includes:

— Technology transfer of design features for related design, construction, and NPP operating and refurbishment requirements;
— Technical support available from comparable NPP operators, including industry groups enabling standardized NPP cooperation or shared operational and support experiences.

A fundamental precondition for successful industrial learning and technology transfer to the owner is a stable regulatory environment in the owner's country, allowing the standardized design and usage of licensing, operational and support experiences from the technology holder's regulatory framework or/and comparable NPP operators. This important portion of the learning offers a significant advantage for SMRs when, using a similar power comparison, a site with one large WCR is compared with a site with many SMRs. In terms of technology transfer, this is a direct function of the national policy of the Member State. The technical support is a key requirement to confirm long term reliable operation of the NPP.

The following questions may support a collection of information relevant to this KT:

— What are the technology transfer opportunities to be offered with this project and design?
— What constitutes the extent of the design basis and in what form will the design basis be transferred to the owner/operator?
— What technologies (software, design documents, design tools) will be provided to support the NPP design?
— What technical or operational support programmes will be available through owner/operator contacts?
— What support and technology are offered to support future required modifications to the design basis?
— What technical or operational support programmes will be available through technology holder contacts with partner suppliers?

KT 9.5 Project contracting options

This KT refers to the models that the technology holder can offer for the entire project contract. Financing is discussed in KE10.5.

The assessment among the designs will ideally address the contract options offered by the technology holder, and/or the government of the country of the technology holder, with regard to project participation, guarantees and type of construction contract offered, and technical support during commissioning, startup and operation.

The NPP can be contracted in different ways [75]. For example, at one extreme, a single contractor can be given complete technical responsibility to design, build and commission a complete NPP, handing it over to the owner only when it is running. At the other extreme, the owner can procure only the basic hardware of the NSSS from the reactor vendor, designing the rest of the NPP and buying all of the other equipment themselves.

As described in [75], the contracting options can be:

— Turnkey contract: A single contractor or a consortium of contractors takes the overall technical responsibility for the whole works: engineering, procurement, construction, commissioning. There are two types of turnkey contract:
  - Super turnkey contract: A single contract is placed covering the whole NPP. The prime technical responsibility for the success of the project and the design of the NPP is placed upon the contractor. It is particularly suitable for utilities with limitations in manpower resources and/or experience in the nuclear field.
  - Normal turnkey contract: A contract placed for a NPP where the utility supplies all peripheral items of the plant (10–20% of the NPP costs). It is usual for owners with nuclear experience or greater competence in conventional NPPs to wish to influence and approve the NPP design to a greater extent than for the super turnkey contracts.
  - Split package contract: Overall technical responsibility is divided between a relatively small number of contractors, each building a large section of the works.

— Multi-package contract: The owner or its architect engineer assumes overall responsibility for engineering the station, issuing many contracts. Each elementary contract can be limited to a part of the plant (e.g. nuclear steam supply system) or a part of the work to be done (e.g. engineering, procurement).

The following questions may support a collection of information relevant to this KT and will ideally be considered in more detail, specifically in Phase 2:

— What is the scope of the contract: Engineering — Procurement — Construction?
— What types of construction contracts are required/offered (e.g. turnkey with fixed price and dates, guarantees and penalty for time, budget, quality)?
— Are any government guarantees or undertakings required?
— Does the technology holder’s government offer any guarantees or undertakings?

KT 9.6 Services offered for the front end of the fuel cycle (fresh fuel supply)

This KT refers to the long term fuel supply policy and the reliability of technological support for the long term if fuel supply is part of the scope. It can have a high importance because it relates directly to the security of fuel supply. Fuel supply related financial issues are considered in KE10.3.

The assessment among the designs addresses the contract options offered by the technology holder, and the country of the technology holder, with regard to sourcing of the fuel, potential fuel supplier and obligations, fuel procurement process, fuel fabrication (including QA and QC by owner), fuel transport (to the owner location including design, V&V and testing of transport casks) and technological transfer about fuel limitations, operations and maintenance.

Fuel services include supply of fresh fuel but also other fuel cycle services including core management, de-fuelling/refuelling procedures and emergency refurbishment of fuel due to defects, if necessary, from the beginning of operation until the end of operations.

For SMR designs with enriched uranium fuel there are no big manufacturing or/and operational experiences that can be compared with large WCRs experiences. The SMRs’ fuel assemblies are sometimes shorter and therefore easier to handle when compared to large WCRs. Sometimes the design of SMR fuel elements is different in comparison with WCRs and requires different tools for manipulation; also, safety and operational limits could be different.

The following questions may support a collection of information relevant to this KT:

— What manufacturing, testing and operational experience do you have on the proposed design or on a similar design of fuel?
— Do you offer fuel cycle management? Do you offer training courses and computer codes necessary for fuel cycle management during plant life on-site?

- What are the warranties for fresh fuel delivered to the owner?
- What experience do you have with transport of the proposed fresh fuel used in this reactor or in similar reactors to the plant?
- What experience do you have with nuclear fuel storage and transport systems of fresh fuel (fuel casks) at NPPs?
- What experience do you have with emergency refurbishment of fuel elements (removal, cleaning, storage, etc.) if necessary?

KT 9.7 Services offered for the back end of the fuel cycle (spent fuel management)

This KT refers to spent fuel management as an important subject of the long term fuel supply policy and the reliability of technological support for the long term. The assessment of the designs addresses the contract options offered by the technology holder, and the country of the technology holder, with regard to ownership of the fuel including the spent fuel, management of the spent fuel on-site, transportation and final deposition. Spent fuel related financial issues are considered in KE10.4.

For the SMR designs with enriched uranium fuel there is no manufacture or/and operational specific experience with spent fuel. The SMRs fuel assemblies are sometimes shorter and therefore easier to handle when compared to large WCRs. Technology for safe storage of spent fuel is not different in comparison with WCRs.

The following questions may support a collection of information relevant to this KT:

- What experience do you have with manipulation of spent fuel from the reactor to the storage facility on-site using the proposed reactor?
- What experience do you have with storage of spent fuel on the site? How is cooling of spent fuel achieved?
- What is the limiting size of the spent fuel pool (how many spent fuel elements can be stored and cooled, or operating cycles covered)?
- Do you plan to transport spent fuel from the plant to the storage facility? When? How?
- What is the expected duration of spent fuel to be stored on the site before transfer to storage or refurbishment?
- What experience do you have with emergency refurbishment of damaged fuel elements (removal, cleaning, storage, etc.)?

5.10. ECONOMICS AND FINANCING

KE10 considers economics and financing, which could differentiate among the technologies under consideration, based on the following six KTs:

KT 10.1 Capital costs;
KT 10.2 Operations and maintenance (O&M) costs;
KT 10.3 Fuel costs;
KT 10.4 Spent fuel management costs;
KT 10.5 Decommissioning cost;
KT 10.6 Financing.

Importance rationale (Table 7): This KE has generally higher importance and it is more relevant for Phase 2 and is often not included at all in the RTA in Phase 1, because financial information is mostly proprietary, or because there is an economics team separate from the RTA team, who assesses the economical factors.

Each KT separately may have lower importance depending on the RTA team's assessment. The relative importance among the KTs may differ between the large WCRs and SMRs due to the size and duration of the overall nuclear power project.

In general, nuclear power costs are dominated by the following three categories of cost: project investment (typically about ~60% of the total cost), O&M (typically about ~20% of the total cost) and fuel and waste (typically about ~20% of the total cost). This is based on experience with hundreds of NPPs for electricity production. For non-electric or cogeneration plants, the adoption of a suitable business model will affect both the deployment and economics.

Description: This KE covers assessment of the designs based on six KTs that define the overall economics and financing of the design.

Special attention is drawn to the evaluation of construction and operational experience that can have a major impact on the capital and operational cost evaluation, and therefore the overall cost evaluation for an NPP. Two key topics that can be assessed from the operational experience and that are directly related to the NPP economics are the facility availability and capacity factors (KT 4.2) and plant life (KT 4.3).

The NEPIO or owner/operator will ideally carefully examine both the technology holder information and the industry information, when evaluating the bases for the claims of the technology holder. When doing so, it is critical to consider the geographical, regulatory and other characteristics that may affect these factors. Operational experience presented by the technology holders will ideally be validated against the environmental conditions applicable to the site and region for reactor deployment.

The following IAEA documents can provide further guidance on the importance and weighting that could be appropriate to this KE:

- IAEA Nuclear Energy Series No. NG-T-4.6 Managing the Financial Risk Associated with the Financing of New Nuclear Power Plant Projects [76];
- IAEA Nuclear Energy Series No. NG-T-4.2 Financing of New Nuclear Power Plants [77];
- Technical Reports Series No. 396 Economic Evaluation of Bids for Nuclear Power Plants [78].

Additionally, the following document provides general guidance on the cost estimation for NPPs: Cost Estimating Guidelines for Generation IV Nuclear Energy Systems [79].

The six KTs are described as follows and examples are provided in the Annex.

KT 10.1 Capital cost

This KT addresses the site specific NPP capital costs. The RTA team may differentiate various designs based on the identification of capital costs and cost impact factors, including material quantities, labour and equipment. The components of this KT to analyse are:

- Capital component of levelized cost of power, electricity, or specific capital cost (cost/kW);
- Comparison of material quantities (e.g. concrete volume) and footprint of the power plant;
- Impact of local labour and productivity, both for the direct and for the indirect components of the costs (such as but not limited to field supervision and fees of the architect engineering firm); the impact of these items is affected by the amount of localization during construction;
- Licensing costs;
- Additional costs typically not included in the vendor's estimate (e.g. the cost of additional transmission if needed);
- Plant design and costs, including the Fukushima Daiichi nuclear power plant related safety improvements;
- Ensuring that all necessary equipment is included in the cost estimate, or that there is no missing equipment (e.g. hydrogen recombiners);
- Assurance of reliable estimates of technology holder equipment prices.

Driven by different investment requirements, siting flexibility, grid connections and infrastructure restrictions, the economic factors affecting the competitiveness of large WCRs and SMRs are different. Also, given that the SMR technologies are still under development, the economic data based on actual experience are not available or do not exist, therefore detailed assessments among designs cannot be performed at the same level as for the large WCRs. It is assumed that SMRs may have an unfavourable size effect on their economics; however, the simplicity of SMR designs and modularization of the construction can bring a cost reduction and shorten the construction time, primarily due to savings in on-site labour.

SMRs have smaller equipment compared to large WCRs, which is expected to reduce transport of the infrastructures and thus the associated costs. A large part of the SMRs' components will be produced in the vendor's country. However, large factory assembled modules may not benefit from this type of saving.

Large WCRs are already licensed to be compliant with different utility requirements and regulations. The next reactors built will not bear these costs. On the contrary, SMRs are a new design. It is an opportunity to have a more standardized licensing, shared by different countries with reduced costs. Compared with large WCRs, the total amount of financing for SMRs is reduced due to a lesser upfront investment. This is expected to facilitate financing arrangements.

Compared with large WCRs, the construction time of an SMR is expected to be shorter and a larger part of the equipment is assembled in factories with less risk over the duration of operations. Lower risk in turn reduces the cost of financing. SMRs can be better adapted to the growing electricity demand than large WCRs, thus reducing the risk of sunk costs.

The following questions may support a collection of information relevant to this KT:

— What are the expected capital costs for the proposed facility (cost/kWe)? This cost is very often provided by different stakeholders (owner/operator, technology holder, etc.). Therefore, please provide the cost information according to a standard cost breakdown structure such as IAEA Coding of Accounts system for NPP, in order to clarify the perimeter of what, as a technology holder, you offer. Also, please provide the estimate of the capital cost for another component, such as owner's cost.
— What are the major material quantities, and do they align with the projection of expected capital costs?
— What are the labour cost estimates used in projecting the capital costs and the supporting rationale? How much localization is expected for this project?
— Have all the costs not included in the vendor's estimate been included (e.g. cost of additional grid connection and substation if included)?
— Have all the licensing costs for this NPP been considered?
— What is the expected capital cost of the NOAK unit versus the cost of the FOAK unit?
— A long construction time generates high interest costs during construction. How long is the expected construction time from first nuclear safety concrete date to the commercial operation date?
— What is the expected gain on the capital cost of building several units instead of one?
— What is the control cost management?
— Does the design comply (and to what extent) with any established user requirements?
— What other design requirements and standards are met?
— Provide the overall qualitative results of the existing compliance assessments regarding technical criteria related to fuel, NSSS, instrumentation and control (I&C) and BOP?

KT 10.2 Operating and maintenance cost

This KT addresses the site specific facility O&M costs. The assessed designs may differentiate based on the technology holder's prediction of the O&M cost and on the basis of the prediction. The O&M cost impact is secondary to the NPP capital costs; however, frequent maintenance can reduce production (thus lowering the capacity factor of the plant) and can be costly. In turn, lower capacity factors increase the overall unit cost of power (expressed as $/MWh).

The components of this KT to analyse are:

— Evaluation of projected O&M cost and staffing with comparisons to experience;
— Plant design features to reduce O&M cost;
— Impact of localization versus O&M contract;
— Opportunities and costs for shared spare parts pool;
— Reliance on passive design and redundant system trains to optimize operation and on-line maintenance;
— Optimized outage schedules based on historic equipment performance and real-time trending data.

The factors influencing the cost are:

— Annualized O&M cost (cost/kWh);
— Operations, maintenance, security, engineering, management, staff costs;
— Operations chemicals (feedstocks) and maintenance materials;
— Replacement equipment and spare parts;
— Utilities, supplies, miscellaneous consumables;
— Capital plant upgrades (not including financing costs);
— Taxes, insurance, regulatory costs;
— Contingency on annualized O&M costs.

Due to their simplifications, there are fewer inspections, tests and maintenance requirements in SMRs when compared to large WCRs. Some of the tests and maintenance are completed in factories. Some SMRs have compact steam generators. They can easily be removed and replaced. This design feature allows for their maintenance outside outage duration in a hot workshop on the site or in a factory. For the same level of production, SMRs can require more staff than large WCRs. With a fleet of SMRs, it will be easier to pool spare parts.

The following questions may support a collection of information relevant to this KT:

— What is the technology holder's estimate of the O&M cost advantage or penalty for the proposed NPP (cost/kWh) versus the O&M costs reported for today's fleet?
— How many people are needed to operate the NPP?
— How many people are needed to maintain the NPP?
— What is the average number of annual days required for maintenance? What is the number of maintenance days outside outages?
— What is the part of maintenance done by the operator?
— What is the part of maintenance done by specialized companies (refuelling)?

KT 10.3 Fuel cost

This KT addresses the nuclear fuel cost. Fuel cost impact is secondary to the NPP capital costs. Fuel costs are a key benefit to the selection of NPPs. When the NPP is in operation, fuel costs account for a high fraction of the variable operating costs. Therefore, the assessment will ideally identify those technical or contractual arrangements that can lower fuel costs for each technology option. Economic assessment of fuel costs may be performed by:

(a) Comparisons of fuel cost estimates provided by each technology holder;
(b) Comparisons of fuel costs provided in the contract, if fuel supply contracts are to be requested in the offerings;

(c) Fuel cost analyses, derived from prospective materials costs as developed in the technology holder proposals to meet the specifications in the request for proposal;
(d) Quantification of the fuel supply requirements, such as the amount of enrichment and natural uranium per kWh produced;
(e) Identifying additional development needed for non-oxide fuel forms, if applicable;
(f) Fuel cost evaluations as developed by consultants or the assessment team.

Contractual options will ideally be differentiated, including the use of alternative fuel technology holders. Security of fuel supply for the reactor technology will ideally be also assessed. Factors to consider include fuel costs projected with comparisons to applicable experience, competitive advantage between original equipment manufacturers and alternative suppliers/fabricators.

Nuclear materials can be similar for large WCRs and SMRs; consequently, in this case a part of the fuel cost would be similar. If the fuel assemblies are similar for large WCRs and SMRs, the fuel costs can be similar. Certain SMR designs require higher enrichment levels than large WCRs, because of the higher neutron leakage due to the smaller core size. In this case, this part of the fuel costs of SMRs may be higher compared to that of WRCs.

The following questions may support a collection of information relevant to this KT:

- What fuel supply arrangements would you offer/require?
- How do you plan to manage the security of fuel supply and possible volatility of fuel price until the end of the plant's life?
- Can the fuel be fabricated indigenously?
- What is the enrichment level required for this fuel, and is the necessary enrichment capacity commercially available?
- Is fabrication capacity readily available for non-oxide fuel forms if applicable?
- What is the number of fuel suppliers for this reactor?
- Do you provide services for fuel: examinations and repairs?

KT 10.4 Spent fuel management cost

This KT addresses the SNF management cost, including intermediate term on-site storage (wet or dry) and final disposal or fuel takeback costs. Spent fuel management may be a key consideration to the selection of NPPs, depending on the Member State's final disposal policy. During NPP operation spent fuel costs typically account for a small fraction of the operating costs. The assessment will ideally identify those technical and/or contractual arrangements that can lower spent fuel management costs for each technology option. Factors to consider include contractual arrangements for fuel supply and spent fuel disposal or takeback.

For large WCRs, solutions for the storage of spent fuel are available, which can be used for SMRs. Spent fuel from large WCRs and from SMRs can be stored in the same centralized pool or in the same dry storage facilities. In the case of more enriched fuel, or even other fuel forms (e.g. liquid metal, salt), in SMRs various back end options including storage, disposal and reprocessing can be more challenging and costly, primarily because of increased criticality concerns. In the case of new fuel assemblies design for SMRs and because of their shorter size, new casks with development costs will be necessary for transport and storage of spent fuel.

The following questions may support a collection of information relevant to this KT:

- Would you take back spent fuel, and if so, at what cost?
- What are the wet and dry storage solutions and costs for the spent fuel?
- Could you reprocess spent fuel? What is the cost of reprocessing?
- Would you take back spent fuel for long term storage and for geological disposal?
- In case you take back and reprocess spent fuel, will you keep nuclear waste?

KT 10.5 Decommissioning cost

This KT considers the NPP decommissioning cost, which is not expected to be a strong differentiator. Decommissioning costs are long term, a small component of overall costs, and will ideally not differ greatly between designs and vendors' offerings. The economic estimates of decommissioning costs can be developed; good practice in design will ideally incorporate features to minimize decommissioning costs and burdens. Decommissioning costs can be substantial in absolute terms; however, since they are collected during but expended following a 60-year plant lifetime, this implies a relatively small impact on levelized energy generation costs. Premature shutdowns can substantially affect the availability of funding for decommissioning. The assessment will ideally include:

— Major decommissioning expenditures start near the end of the facility's lifetime, but the funds for covering the expenses are generally accumulated during operation;
— Policies to manage and invest the decommissioning escrow fund during the operation and post-closure period before actual decommissioning begins;
— Experience gained in other projects on the actual decommissioning expenditures;
— National policies, industrial strategies and cost estimation models adopted or assumed for decommissioning projects vary widely.

SMRs require less materials and equipment, therefore decommissioning may have lower costs per unit. For the same level of capacity production, large WCRs compared to SMRs in several sites can have lower decommissioning costs. Large WCRs and SMRs will be built with the latest technologies. Digitalization will facilitate decommissioning.

The following questions may support a collection of information relevant to this KT:

— What is the projected decommissioning and decontamination funding requirement at the end of the NPP's lifetime (as a percentage of the total direct capital cost)?
— What principles of design for decommissioning are considered?
— What is the schedule for the decommissioning of the NPP?
— What is the expected duration for the decommissioning of the NPP?
— What is the decommissioning schedule imposed by regulations?
— What materials can be recycled?
— What is the quantity of waste resulting from decommissioning?
— What experience in costing and best practices can be gained from completed decommissioning projects?

KT 10.6 Financing

This KT considers the capital payment structure and project financing that can be a strong differentiator but is typically not known in Phase 1. It can have a high importance in Phase 2 because it relates directly to the project affordability and the related financial closure. Because of the high cost of construction, a difference in the cost of financing (discount rate) can have a large impact. In general, it might be easier to finance SMRs because the absolute construction costs of each module are lower, the construction time of each module is expected to be shorter, and cost overruns and delays during constructions are expected to be lower, thus reducing the construction risk of the project.

The capital payment structure and project financing options traditionally seem to be highly dependent on the plant size and complete nuclear energy programme of the owner. Some techno-economic analyses show that the average investment and operating costs per unit of electricity are decreasing with respect to increasing plant size. This result cannot be directly transferred into the investment analyses of SMRs versus WCRs, because it relies upon the clause 'other things being equal' and such analyses are too complex to be elaborated here. Such traditional approaches also do not consider that SMRs exhibit several benefits that are uniquely available to smaller innovative reactors and can only be replicated by WCRs to a limited extent. The most important factors are modularization, multiple units at a single site and new design strategies and solutions. SMRs, due to their inherently modular approach, are easier to build and, because of their smaller size, the FOAK impact on cost escalation has a limited effect.

The difference in amounts to invest and construction time between large WCRs and SMRs could be an opportunity to experiment more build-own-operate, build-own-operate-transfer, build-operate-transfer models.

The following questions may support a collection of information relevant to this KT:

- What relevant inputs can you provide for the risk analysis?
- What is the experience about financing of nuclear projects?
- What financing arrangements and terms are available from the technology holder?
- What level of export credits can you provide?
- Can you introduce commercial banks in the offer?
- Can you be one of the shareholders of the special purpose vehicle for the nuclear project? At what level?
- If you are a shareholder, do you agree with a contract for a difference model or a regulated asset base model?
- Regarding the NPP project, are you interested in the following models: build-own-operate, build-own-operate-transfer or build-operate-transfer?

6. CONCLUSION

This publication demonstrates how the IAEA RTA methodology is performed and how the process and results of this work enable national decision making for nuclear power planning and implementation. The preparation for, and application of, the RTA methodology and process itself creates an additional vehicle for capacity building in Member States through IAEA technology training, based on this publication. Specifically, the carefully developed examples provided in the Annex aim to guide the RTA practitioners and decision makers on how to support their RTA conclusions for an unbiased selection of a nuclear power technology.

The primary users of this publication are the NEPIO, utilities/operator organizations, governing organizations, or others who are or will be responsible for the process of selecting the NPP technology. Practitioners in the RTA team are technical experts involved in advising government or utility officials. Other technical experts in Member States, even regulators and other stakeholders involved in the NPP project, can benefit from this publication.

The decision makers for RTA and implementation are the ultimate users of the output from the work described in this publication. Accordingly, it is expected that reactor suppliers, architect engineers and constructors, and equipment manufacturers will also benefit from an understanding of how their reactor designs and technical proposals will be evaluated, judged and selected. IAEA Member States need to obtain reliable information that can be used to make these relevant comparisons between different NPP designs. The best source of data will ideally be that provided by the technology holder. The IAEA expects that technology holders may follow the approach given and develop a standard technical description of their product with an emphasis on addressing the key questions that are identified in this publication. These data, when provided for use by multiple countries, would support the performance of RTA in IAEA Member States.

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Annex

REACTOR TECHNOLOGY ASSESSMENT EXERCISES WITH EXAMPLES

This Annex presents examples on how to apply the IAEA RTA methodology in assessing various NPP designs. It also provides information that may need to be requested from the technology holder. Additional assessments may be necessary to have a global view of the offer.

Figure A–1 explains the steps to take in assessing the NPP designs:

1.1 Before assigning the importance values (in %) to ten KEs, the RTA team develops a rationale for their individual importance. Table 8 provides some initial ideas on the importance of the KEs if the national power programme considers deployment of a large WCR or SMR, without or with cogeneration applications.
1.2 When rationales are developed, the % importance can be assigned in summing to 100%.
2.1 For every KE, the rationales per KTs are to be well defined before assigning the weights.
2.2 The weights (in %) are assigned to every KT per KE based on the developed rationale. The sum per KT is 100%.
3.1 Develop rationale for every KT per KE to score NPP designs. The scoring scale is from 1 to 5, where 1 is the design that least meets the rationale for scoring while 5 is the best fit.
3.2 Provide the scores from 1 to 5 to every NPP design based on rationale for scoring.

The following RTA tables (A–1 to A–19) provide examples on how to exercise the RTA methodology. The illustrative examples are provided for a fictitious country Retasland (Refs [A–1] and [A–2]) and three NPP designs based on WCR or SMR technology. Each table may apply to a different set of reactor technologies. The RTA team may detail any KT further sub-KTs; such examples are also provided.

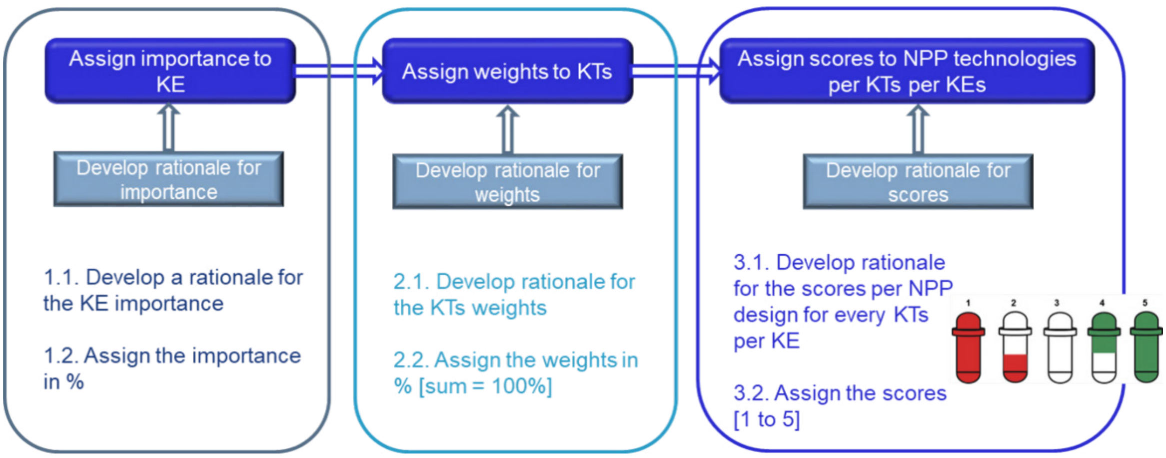

FIG. A–1. RTA matrix.

TABLE A–1. RTA MATRIX FOR KE1 AND LARGE WCRs

KE 1 Site and environment

Rationale for importance	**Importance**
The site and environment are of prime importance because they synergistically define the conditions or constraints that cause the NPP to be financially and technically attractive for Retasland, compared to the alternative approaches or in combination with them (e.g. gas, coal, hydro, renewables)	**HIGH (%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
1.1 Site seismicity	25	High expected peak ground acceleration of 0.12 g requires large seismic margin. DBE PGA has to be ≥0.2 g	5 (0.3g)	2 *(more info)*	4 (0.25g)	5: Seismic design ≥0.3 g 4: Seismic design is <0.3 g and >0.2 g 3: Seismic design = 0.2 g 2: Seismic design information is limited 1: Seismic design cannot withstand 0.12 g OR no information provided
1.2 Meteorology and hydrology	25	Unique wind conditions with seasonal variations in temperature, humidity and participation No flood risk	**Divided into two sub-KTs**			
Sub-KT 1.2.1 High winds	50	Strong winds blowing 6 month/ year	5	1 (*more info*)	2 (*more info*)	5: Designed against strong winds 4: Design includes some measures against strong winds 3: Design is yet to be proven in strong winds 2: Designed against other meteorological effects but not against the strong winds 1: Design does not consider measures against strong winds OR no information provided
Sub-KT 1.2.2 Seasonal variations in temperature, humidity and precipitation	50	Hot and humid summer, and winter with light snow	5	1 (*more info*)	2 (*more info*)	5: Designed against seasonal variations 4: Design includes some measures against seasonal variations 3: Design is yet to be proven in seasonal variations 2: Designed against other meteorological effects but not against seasonal variations 1: Design does not consider measures against seasonal variations OR no information provided

TABLE A–1. RTA MATRIX FOR KE1 AND LARGE WCRs (cont.)

KE 1 Site and environment

Rationale for importance	**Importance**
The site and environment are of prime importance because they synergistically define the conditions or constraints that cause the NPP to be financially and technically attractive for Retasland, compared to the alternative approaches or in combination with them (e.g. gas, coal, hydro, renewables)	**HIGH** **(%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
1.3 **Water resources**	5	Seawater used as a heat sink (2 km) Availability of seawater for heat sink	4	3 (*more info*)	5	5: Design fully adaptable to various water sources as a heat sink and incorporates features for emergency shutdown supply 4: Designed to use two water sources as a heat sink (i.e. river and seawater) 3: Designed for one source of water as a heat sink 2: Design considers a cooling water source but with limited information 1: No information provided
1.4 **Population**	5	Site is 5 km from a city with a population of 200 000	5	4	2	5: Required EPZ smaller than 5 km with high standard emergency response plan provided 4: Required EPZ close to 5 km with emergency response plan provided 3: Required EPZ between 6 and 10 km but with security measures in place to protect population and environs from radiation release with emergency response plan provided 2: Design considers exclusion zone and emergency response plan, but information provided is limited. 1: Required exclusion zone greater than 15 km OR no information provided

TABLE A–1. RTA MATRIX FOR KE1 AND LARGE WCRs (cont.)

KE 1 Site and environment

Rationale for importance	**Importance**
The site and environment are of prime importance because they synergistically define the conditions or constraints that cause the NPP to be financially and technically attractive for Retasland, compared to the alternative approaches or in combination with them (e.g. gas, coal, hydro, renewables)	**HIGH (%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
1.5 Site access for construction and operation	5	No archaeological sites No recreational centres Mountains located 50 km away from the site Transportation routes well developed with 10 m wide roads Bridges have a load limit of 1 000 tons, and a maximum height clearance of 6 m	5	4	3	5: Plant heavy components are transportable using the existing roads and within the bridge limit 4: Plant heavy components exceeds some limits, but alternative possibilities exist 3: Some plant heavy components exceed both road width limit and bridge weight or height limit. Infrastructure investment is required 2: Plant heavy components exceed limits, requiring substantial investment in new roads or bridge 1: No information provided
1.6 Site size	15	10 km × 10 km flat land available as site for all units planned by 2050 Footprint for the first 1 200 MWe nuclear power plant unit of <3 km^2 is desirable	1	4	5	5: Plant footprint including additional units <3 km^2 4: Plant footprint larger but <5 km^2 3: Plant footprint with additional units larger than 5 km^2 but design modifications can be made, and/or minor land purchases and allocations are possible 2: Plant footprint is with additional units importantly larger than the available site size with no availability to modify the design or expand the site size 1: Plant design requires significantly larger site size than available in approaching population centres OR information not provided

TABLE A–1. RTA MATRIX FOR KE1 AND LARGE WCRs (cont.)

KE 1 Site and environment

Rationale for importance	**Importance**
The site and environment are of prime importance because they synergistically define the conditions or constraints that cause the NPP to be financially and technically attractive for Retasland, compared to the alternative approaches or in combination with them (e.g. gas, coal, hydro, renewables)	**HIGH (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
1.7 Environmental and radiological impact	5	No requirements to relocate population centres No effect on natural habitats Strict requirements for seawater temperature used as a heat sink (max 35°C)	5	4	2 (*more info*)	5: Exhaust temperatures have no foreseeable effects on seawater temperature and aquatic life 4: Exhaust temperatures are at the required level, including record high temperature conditions with substantial margin 3: Exhaust temperatures meet the requirements, including the current record high temperatures 2: Exhaust temperatures render the plant inoperable only during record high temperatures 1: Exhaust temperatures render the plant inoperable for long periods of time OR no information is provided
1.8 External events	15	Commercial airline flight zone No sandstorms No public trespassing	4	3	1	5: Considers all noted external events including plant resistance to commercial airplane crash (design proven) 3: Contains features which offer some protection from noted external events 1: No systems designed indicating protection from noted external events OR no information provided

Note: info: information. A score of 1 for 'no information provided' is only to be given after contacting the vendor and receiving no/unacceptable response.

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

1.1 Question NPP2: The NPP design considers SSE and OBE ground motion but does not specify a numeric value. What is the site seismic level ground acceleration or PGA value considered in the design?

1.2 Justification NPP1: Designed against major meteorological and hydrological conditions including strong winds, hurricanes, tornadoes, flood and snow load (up to 0.6 kpa), and therefore scored high for L-Retasland site. Question NPP2: What meteorological and hydrological conditions are considered in the design and what is the maximum degree of tolerance? Question NPP3: What is the numerical value and provisions made for strong winds in the design?

1.3 Justification NPP1: Plant design can use both river and sea as a heat sink and therefore scored higher for L-Retasland site.

Question NPP2: Plant designed for the sea as a heat sink. In the absence of seawater, what other alternatives can be used as a heat sink?

Justification NPP3: Design fully adaptable to various water sources as a heat sink and incorporates features for emergency shutdown supply, therefore scored the highest for L-Retasland site.

1.5 Justification NPP1: Plant design major component weight is 330 t and the height is 3.85 m and therefore scored the highest for L-Retasland site.

Justification NPP2: Plant design major component weight is 480 t and the height is 5.2 m and therefore scored the second highest for L-Retasland.

Justification NPP3: Plant design major component weight is 700 t and the height is 5 m, therefore scored the third highest for L-Retasland.

1.7 Question NPP3: What is the outlet coolant temperature value?

TABLE A–2. RTA MATRIX FOR KE1 AND SMRs

KE 1 Site and environment

Rationale for importance	**Importance**
The site and environment are of prime importance because they synergistically define the conditions or constraints that cause the nuclear power plant to be financially and technically attractive for Retasland, compared to the alternative approaches or in combination with them (e.g. gas, coal, hydro, renewables)	**MEDIUM (%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
1.1 Site seismicity	15	High expected peak ground acceleration of 0.12 g requires large seismic margin DBE PGA has to be ≥0.2g	4 (0.2 g)	2 *(more info)*	3 (0.12 g)	5: Seismic design ≥0.3 g 4: Seismic design is <0.3 g and >0.2 g 3: Seismic design = 0.2 g 2: Seismic design information is limited 1: Seismic design cannot withstand 0.12 g OR no information provided
1.2 Meteorology and hydrology	10	Strong winds Hot and humid Floods with an average of 0.02 km (0.05 km highest)	3 (flood height 0.08 km)	4 (flood height 0.1 km)	4 (flood height 0.1 km)	5: Plant designed against strong winds and temperature effects and can withstand floods of height >0.1 km 4: Plant design considers strong winds and temperature effects and can withstand floods of max 0.1 km 3: Plant design considers winds and temperature effects and can withstand floods of max 0.08 km 2: Plant design considers winds and temperature effects with limited information on floods levels 1: Plant designed against floods of max 0.03 km and no operational experience in areas with extreme wind conditions OR no information provided

KE 1 Site and environment

Rationale for importance	Importance
The site and environment are of prime importance because they synergistically define the conditions or constraints that cause the nuclear power plant to be financially and technically attractive for Retasland, compared to the alternative approaches or in combination with them (e.g. gas, coal, hydro, renewables)	**MEDIUM (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
1.3 Water resources	15	Freshwater intake is distanced 5 km from the site and can be used as a heat sink	5	5	5	5: Design fully adaptable to various water sources as heat sink and incorporates features for emergency shutdown supply 4: Designed to use two water sources as a heat sink (i.e. river and seawater) 3: Designed for one source of water as a heat sink 2: Design considers a cooling water source but with limited information 1: No information provided
1.4 Population	10	Site is 10 km from a city with a population of 3 000	2 (more info)	5 (exclusion zone 2 km)	5 (exclusion zone 2.5 km)	5: Required exclusion zone smaller or equal to 3 km with high standard emergency response plan provided 4: Required exclusion zone equal to 3 km with emergency response plan provided 3: Required exclusion zone 5 km but with security measures in place to protect population and environs from radiation release with emergency response plan in place 2: Design considers exclusion zone and emergency response plan but information provided is limited 1: Required exclusion zone way greater than 3 km OR no information provided

TABLE A–2. RTA MATRIX FOR KE1 AND SMRs (cont.)

KE 1 Site and environment

Rationale for importance	**Importance**
The site and environment are of prime importance because they synergistically define the conditions or constraints that cause the nuclear power plant to be financially and technically attractive for Retasland, compared to the alternative approaches or in combination with them (e.g. gas, coal, hydro, renewables)	**MEDIUM** **(%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
1.5 **Site access for construction and operation**	10	Site in remote area Roads are 7–10 m wide with bridge load limit of 400 t and max height clearance of 5 m 300 MW capacity transmission line 3 km away No archaeological, recreational or tourist sites in the site vicinity	5 (largest plant component is turbine generator weighing 200 t)	5 (largest plant module weight is 250 t)	5 (largest plant module weight is 260 t)	5: Largest plant component less than 300 t can be transported to site without roads or bridges modifications 4: Minor roads/bridges modifications are required to transport largest plant equipment if weighted between 300–350 t 3: Modifications to roads and/or bridges are required to transport the heaviest plant component 2: Significant modifications to roads and/or bridges are required to transport the heaviest plant component 1: No information provided
1.6 **Site size**	15	Site size 5 km × 5 km	3	2	4	5: Plant footprint including additional units is within the site size 4: Plant footprint is larger but adjustable to the site size 3: Plant footprint with additional units is larger than the available site size but design modifications can be made, and/or minor land purchases and allocations are possible 2: Plant footprint with additional units is importantly larger than the available site size with no availability to modify the design or expand the site size 1: Plant design requires a significantly larger site size than available in approaching population centres OR information not provided

TABLE A–2. RTA MATRIX FOR KE1 AND SMRs (cont.)

KE 1 Site and environment

Rationale for importance	**Importance**
The site and environment are of prime importance because they synergistically define the conditions or constraints that cause the nuclear power plant to be financially and technically attractive for Retasland, compared to the alternative approaches or in combination with them (e.g. gas, coal, hydro, renewables)	**MEDIUM (%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
1.7 Environmental and radiological impact	15	No requirements to relocate population centres No effect on natural habitats Strict requirements for water body temperature used as a heat sink (max 27°C)	3	4	3	5: Exhaust temperatures have no foreseeable effects on river temperature and aquatic life 4: Exhaust temperatures are at the required level, including record high temperature conditions with substantial margin 3: Exhaust temperatures meet the requirements, including the current record high temperatures 2: Exhaust temperatures render the plant inoperable only during record highs 1: Exhaust temperatures render the plant inoperable for long periods of time, OR no information is provided
1.8 External events	10	No major air traffic lines over the site No public trespassing Sandstorms at 45 km/hr speed and 20 m height	5	3	3	5: Considers all noted external events including operational experience in sandstorm areas (design proven) 3: Contains features which offer some protection from noted external events 1: No systems designed indicating protection from noted external events OR no information provided

Note: info: information

TABLE A–3. RTA MATRIX FOR KE2 AND LARGE WCRs

KE 2 Fuel cycle

Rationale for importance	**Importance**
Fuel costs are small in comparison to capital costs of the nuclear power plant. However, the fuel, the fuel cycle and the in-plant management of fuel have a major impact on plant operation and operating costs. Besides, Retasland will import all the enriched uranium necessary and will store the spent fuel on the nuclear power plant site. Consequently, the decision regarding the fuel cycle is of prime importance	**HIGH (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
2.1 Fuel materials and components	10	The uranium is imported The capability of the fuel supply chain is important for security of supply	5	3	3	5: The technology can use enriched uranium, reprocessed uranium and MOX; fuel assemblies are made with broadly available materials 4: The technology can use enriched uranium, enriched reprocessed uranium and MOX after some modifications; alloys of fuel assemblies are made with broadly available materials
2.1 Fuel materials and components (cont.)						3: The technology can use enriched uranium and enriched reprocessed uranium; alloys of fuel assemblies are made with broadly available materials 2: The technology can easily use enriched uranium at different levels between 3 and 5% 1: The technology can use enriched uranium between 3 and 4%
2.2 Fuel product supply chain	10	The enriched uranium will be imported A diversified supply chain is important for security of supply	4	5	5	5: The technology can use the fuel product with a widely diversified supply chain 4: The technology can use the fuel product with a diversified supply chain 3: The technology can use the fuel product with an undiversified supply chain 2: The technology can use the fuel product with a specific supply chain 1: The technology can use the fuel product with a specific supply chain partially available

KE 2 Fuel cycle

Rationale for importance Fuel costs are small in comparison to capital costs of the nuclear power plant. However, the fuel, the fuel cycle and the in-plant management of fuel have a major impact on plant operation and operating costs. Besides, Retasland will import all the enriched uranium necessary and will store the spent fuel on the nuclear power plant site. Consequently, the decision regarding the fuel cycle is of prime importance	**Importance HIGH (%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
2.3 **Fuel unit fabrication**	20	Only one LWCR will be built by 2035 and four or five by 2050 Fabrication by several countries and suppliers aids in long term security of supply	5	4	3	5: Several plants and several vendors can provide fuel for this technology 4: Several vendors can provide fuel for this technology 3: Few plants and few vendors can provide fuel for this technology 2: Different vendors could provide fuel for this technology in the case of a long and large commitment of the country 1: Only one vendor can provide fuel for this technology
2.4 **Fuel operating experience**	15	Fuel operating experience brought by fuel vendors for many reactors for a long time provides many guarantees	5	4	3	5: The vendor has experience in many reactors for a long time 4: The vendor has indirect or partial experience in many reactors for a long time 3: The vendor has experience in a few reactors for a long time 2: The vendor has experience in a few reactors for a short time 1: The vendor has experience in only one reactor
2.5 **Refuelling outage**	15	The maximum planned (refuelling) outage is 20 days (only in spring) Frequency and duration of refuelling outages impact the level of production	3	4	2 (*more info*)	5: The vendor can provide training for outages; and refuelling outages can be chosen, are infrequent and short: ~15 days 4: Refuelling outages are infrequent and <20 days 3: The vendor can provide training for outages; refuelling outages are infrequent and >20 days 2: Refuelling outages are frequent and >25 days 1: Refuelling outages are frequent and >30 days

TABLE A–3. RTA MATRIX FOR KE2 AND LARGE WCRs (cont.)

KE 2 Fuel cycle

Rationale for importance
Fuel costs are small in comparison to capital costs of the nuclear power plant. However, the fuel, the fuel cycle and the in-plant management of fuel have a major impact on plant operation and operating costs. Besides, Retasland will import all the enriched uranium necessary and will store the spent fuel on the nuclear power plant site. Consequently, the decision regarding the fuel cycle is of prime importance

Importance HIGH (%)

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
2.6 Fuel flexibility by diversity of fuel used	10	Diversity of fuel used enhances security of supply, has an economic interest and is useful for back end fuel cycle	5 (*more info*)	3	4	5: Enriched uranium, enriched reprocessed uranium, MOX can be used; the vendor can transfer a large experience 4: Enriched uranium, enriched reprocessed uranium, MOX can be used 3: Limited amounts of MOX can be used and the vendor can transfer a large experience 2: Limited amounts of MOX can be used 1: Only enriched uranium can be used
2.7 Suitability to indigenous fuel fabrication	5	Indigenous fuel fabrication can strengthen security of supply and has economic benefits It is less important in the case of competitive fuel supply	3	2 (*more info*)	4	5: Indigenous fuel fabrication is achievable; the vendor can transfer technology 4: Indigenous fuel fabrication is a difficult option; the vendor can transfer technology 3: Indigenous fuel fabrication is a difficult option 2: Indigenous fuel fabrication is possible for some components 1: Indigenous fuel fabrication is difficult and very costly
2.8 Medium term spent fuel storage capacity	5	Spent fuel will be stored on the nuclear power plant site Solutions for medium term spent fuel storage is anticipated	3	5 (*more info*)	2	5: Wet or dry storage can be proposed for several decades 4: Wet or dry storage can be easily increased 3: Wet or dry storage can be increased 2: Dry storage can be increased 1: Increasing medium term spent fuel storage is difficult and very costly

TABLE A–3. RTA MATRIX FOR KE2 AND LARGE WCRs (cont.)

KE 2 Fuel cycle

Rationale for importance Fuel costs are small in comparison to capital costs of the nuclear power plant. However, the fuel, the fuel cycle and the in-plant management of fuel have a major impact on plant operation and operating costs. Besides, Retasland will import all the enriched uranium necessary and will store the spent fuel on the nuclear power plant site. Consequently, the decision regarding the fuel cycle is of prime importance	**Importance** **HIGH** **(%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
2.9 **Long term spent fuel storage**	10	There are no commitments in place for final disposal of the spent fuel Different solutions for long term spent fuel storage can be explored	3	2	5	5: The vendor can take back spent fuel, reprocess it and keep the waste 4: The vendor can take back spent fuel without reprocessing it 3: The vendor can take back spent fuel, reprocess it and return the waste 2: The vendor can propose experience for a local long term spent fuel storage 1: The vendor can propose experience for a local deep geological disposal

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

2.5 Question NPP3: What are the prospects of reducing the duration of stops to 20 days?

2.6 Question NPP1: Can you specify the experience of using enriched reprocessed uranium and MOX. How many years of experience do you have?

2.7 Question NPP2: What are the fuel components for which local manufacturing seems possible to you?

2.8 Question NPP2: Can you specify the duration of the safe and secure wet or dry storage?

TABLE A–4. RTA MATRIX FOR KE2 AND SMRs

KE 2 Fuel cycle

Rationale for importance	**Importance**
Fuel costs are small in comparison to capital costs of the nuclear power plant. However, the fuel, the fuel cycle and the in-plant management of fuel have a major impact on plant operation and operating costs. Besides, Retasland will import all the enriched uranium necessary and will store spent fuel on the nuclear power plant site. Consequently, the decision regarding fuel cycle is of prime importance	**HIGH (%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
2.1 Fuel materials and components	10	The capacity of the technology to use different fuel materials and components produced in several countries and provided by different suppliers is important for security of supply	5	2	4	5: The technology can use enriched uranium, enriched reprocessed uranium and MOX; alloys of fuel assemblies are made with broadly available materials 4: The technology can use enriched uranium and enriched reprocessed uranium; alloys of fuel assemblies are made with broadly available materials 3: The technology can use enriched uranium at different levels between 3 and 5% 2: The technology can use only enriched uranium >10% 1: The technology can use enriched uranium between 3 and 3.5%
2.2 Fuel product supply chain	10	A diversified supply chain with different mines, plants, means of transport and stock is important for security of supply	4	2	3	5: The technology can use a diversified fuel product supply chain 4: The technology can use a fuel product supply chain with adaptations 3: The technology can use a fuel product supply chain with many adaptations 2: The technology uses a specific fuel product supply chain 1: The technology can use a fuel product with a specific supply chain partially available

TABLE A–4. RTA MATRIX FOR KE2 AND SMRs (cont.)

KE 2 Fuel cycle

Rationale for importance
Fuel costs are small in comparison to capital costs of the nuclear power plant. However, the fuel, the fuel cycle and the in-plant management of fuel have a major impact on plant operation and operating costs. Besides, Retasland will import all the enriched uranium necessary and will store spent fuel on the nuclear power plant site. Consequently, the decision regarding fuel cycle is of prime importance

Importance HIGH (%)

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
2.3 Fuel unit fabrication	20	Fuel unit fabrication in several countries and owned by several suppliers is important for security of supply	4	1	3	5: Several plants and several vendors can provide fuel for this technology 4: Several vendors can provide fuel for this technology 3: Few plants and few vendors can provide fuel for this technology 2: Different vendors could provide fuel for this technology in the case of a long and large commitment of the country 1: Only one vendor can provide fuel for this technology
2.4 Fuel operating experience	15	Fuel operating experience brought by the vendors in many reactors for a long time provides many guarantees	5 (*more info*)	1	4	5: The vendor has experience with a similar fuel in a large number of reactors for a long time 4: The vendor has indirect or partial experience with a similar fuel in a large number of reactors for a long time 3: The vendor has experience with a similar fuel in a few reactors for a long time 2: The vendor has experience in a few reactors with a similar fuel for a short time 1: The vendor has no experience of this fuel

TABLE A–4. RTA MATRIX FOR KE2 AND SMRs (cont.)

KE 2 Fuel cycle

Rationale for importance	**Importance**
Fuel costs are small in comparison to capital costs of the nuclear power plant. However, the fuel, the fuel cycle and the in-plant management of fuel have a major impact on plant operation and operating costs. Besides, Retasland will import all the enriched uranium necessary and will store spent fuel on the nuclear power plant site. Consequently, the decision regarding fuel cycle is of prime importance	**HIGH (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
2.5 Refuelling outage	15	The country is interconnected and renewables are increasing. The maximum planned outage is 15 days (only in spring)	3 (*more info*)	5	4	5: The vendor can provide training for outages; and refuelling outages can be chosen and are infrequent: 10 years and short: 15 days 4: Refuelling outages are infrequent and short: 12 days 3: The vendor can provide training for outages; and refuelling outages are infrequent and long: 18 days 2: Refuelling outages are infrequent and long: 22 days 1: Refuelling outages are frequent and long: 22 days
2.6 Fuel flexibility by diversity of fuel used	80	Diversity of fuel used offers flexibility enhancing security of supply, having an economic interest and is useful for the back end fuel cycle	4	1 (*more info*)	3	5: Enriched uranium, enriched reprocessed uranium, MOX can be used in the reactor and the vendor can transfer a large experience 4: Enriched uranium, enriched reprocessed uranium, MOX can be used in the reactor 3: Enriched uranium, limited amounts of MOX can be used in the reactor 2: Limited amounts of MOX can be used in the reactor 1: Only enriched uranium can be used
2.7 Suitability to indigenous fuel fabrication	5	Indigenous fuel fabrication can strengthen the security of supply and bring economic benefits	4	2	3	5: Indigenous fuel fabrication is achievable; the vendor can transfer technology 4: Indigenous fuel fabrication is a difficult option; the vendor can transfer technology 3: Indigenous fuel fabrication is a difficult option 2: Indigenous fuel fabrication is possible for some components 1: Indigenous fuel fabrication is difficult and very costly

TABLE A–4. RTA MATRIX FOR KE2 AND SMRs (cont.)

KE 2 Fuel cycle

Rationale for importance Fuel costs are small in comparison to capital costs of the nuclear power plant. However, the fuel, the fuel cycle and the in-plant management of fuel have a major impact on plant operation and operating costs. Besides, Retasland will import all the enriched uranium necessary and will store spent fuel on the nuclear power plant site. Consequently, the decision regarding fuel cycle is of prime importance	**Importance** **HIGH** **(%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
2.8 **Medium term spent fuel storage capacity**	5	Spent fuel will be stored on the nuclear power plant site Solutions for medium term spent fuel storage are anticipated	5	3	3	5: Wet or dry storage can be proposed for several decades 4: Wet or dry storage can be easily increased 3: Wet or dry storage can be increased 2: Dry storage can be increased 1: Increasing medium term spent fuel storage is difficult and very costly
2.9 **Long term spent fuel storage**	10	Solutions for long term spent fuel storage are anticipated	3	2 (*more info*)	5	5: The vendor can take back spent fuel, reprocess it and keep the waste 4: The vendor can take back spent fuel without reprocessing it 3: The vendor can take back spent fuel, reprocess it and return the waste 2: The vendor can propose experience for a local long term spent fuel storage 1: The vendor can propose experience for a local deep geological disposal

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

2.4 Question NPP1: What guarantees can you provide on your experience with a similar fuel?
2.5 Question NPP1: What are the prospects of reducing the duration of stops to 20 days?
2.6 Question NPP2: What are the prospects for using fuels other than enriched uranium in the future?
2.9 Question NPP2: Can you explain your experience for a local long term spent fuel storage?

TABLE A–5. RTA MATRIX FOR KE3 AND LARGE WCRs

KE 3 Nuclear safety						
Rationale for importance Nuclear safety is expected to be included at the policy objectives level or the highest key element contribution level. It has the potential to be a strong differentiator. Retasland Nuclear Safety Commission bases licensing decisions on IAEA Safety Standards, so these are used in scoring where possible						**Importance** **HIGH** **(%)**
KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
3.1 Implementation of DiD philosophy	6	DiD is a broad-based concept. The main elements of DiD are inherent in other KEs and KTs	4	3	2	5: Well defined safety programme covering all five levels of DiD is well developed 3: Safety programme covering all five levels of DiD is in development 1: Safety programme does not cover all five levels of DiD or is basic
3.2 Safety design philosophy	2	Both active and passive designs approaches can potentially deliver adequate safety	2	4	5	5: Fully passive approach 3: Mixed active/passive approach 1: Fully active approach
3.3 Degree of diversity and redundancy	2	Overall level of safety will be assessed by DSA and PSA	5	4	3	5: High degree of diversity and redundancy 3: Medium degree of diversity and redundancy 1: Unacceptable degree of diversity and redundancy
3.4 Protection against internal and external hazards	8	Hazards can contribute significantly to overall plant safety	4	3	3	5: Good protection against hazards, well documented 3: Moderate protection against hazards or protection is not fully documented 1: Low protection against hazards or has not been demonstrated
3.5 Response to off-site power loss	2	This is just one postulated accident among many. Importance is already assessed in DSA	5	4	4	5: Nuclear power plant is robust against LOOP 3: Nuclear power plant has moderate protection against LOOP 1: Nuclear power plant is vulnerable to LOOP or protection has not been demonstrated

TABLE A–5. RTA MATRIX FOR KE3 AND LARGE WCRs (cont.)

KE 3 Nuclear safety

Rationale for importance	**Importance**
Nuclear safety is expected to be included at the policy objectives level or the highest key element contribution level. It has the potential to be a strong differentiator. Retasland Nuclear Safety Commission bases licensing decisions on IAEA Safety Standards, so these are used in scoring where possible	**HIGH (%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
3.6 Completeness of OLCs, SAR, PSA, O&EPs, SAMGs	10	Moderate weight as document completeness at RTA stage does not strongly impact safety	**Divided into five sub-KTs**			
Sub-KT 3.6.1 Safety analysis report	30	DSA part of SAR have to have results for key accidents	5	3	3	5: SAR is almost complete or has been prepared for another Member State 3: SAR covers full scope, but not in detail 1: SAR does not exist or is just an outline
Sub-KT 3.6.2 Operational limits and conditions	10	OLCs not expected to be complete	5	2	3	5: OLCs are almost complete or have been prepared for another Member State 3: OLCs cover full scope, but not in detail 1: OLCs do not exist or are just outlines
Sub-KT T 3.6.3 Probabilistic safety assessment	30	Level 1 PSA results have to be available for internal events	5	3	2	5: PSA almost complete or have been prepared for another Member State 3: PSA covers full scope, but not in detail 1: PSA does not exist or is just an outline
Sub-KT 3.6.4 Operating and emergency procedures	10	O&EPs not expected to be complete	5	1	2	5: O&EPs are almost complete or have been prepared for another Member State 3: O&EPs cover full scope, but not in detail 1: O&EPs do not exist or are just outlines
Sub-KT 3.6.5 Severe accident management guidelines	20	SAMGs not expected to be complete	5	4	1	5: SAMGs are almost complete or have been prepared for another Member State 3: SAMGs cover full scope, but not in detail 1: SAMGs do not exist or are just outlines

TABLE A–5. RTA MATRIX FOR KE3 AND LARGE WCRs (cont.)

KE 3 Nuclear safety

Rationale for importance	**Importance**
Nuclear safety is expected to be included at the policy objectives level or the highest key element contribution level. It has the potential to be a strong differentiator. Retasland Nuclear Safety Commission bases licensing decisions on IAEA Safety Standards, so these are used in scoring where possible	**HIGH (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
3.7 Results of deterministic safety analysis	18	DSA is a key predictor of overall nuclear power plant safety	5	3	4	5: DSA is advanced and gives acceptable results 3: Key examples of DSA are available and give acceptable results 1: Few examples of DSA are available or do not give confidence that results will be acceptable
3.8 Results of probabilistic safety assessment	18	PSA is a key predictor of overall nuclear power plant safety	4	3	3	5: Level 1 and 2 PSA give acceptable results 3: Key parts of PSA give acceptable results 1: PSA results do not give confidence that results will be acceptable
3.9 Mitigation of severe accidents	8	Main indicators are DSA and PSA	3	4	4	5: Design provides very low probability of severe accident or has strong mitigation measures 3: Design provides balance between severe accident prevention and mitigation 1: Severe accident prevention or mitigation has not been demonstrated
3.10 Operational expectations affecting safety	10	Becomes very important in late stages, or after RTA	4	4	1	5: Technology holder has mature, pre-operational programmes and supports research and data sharing 3: Technology holder has limited pre-operational programmes and limited research and data sharing 1: Technology holder has no pre-operational programmes and does not participate in research and data sharing

TABLE A–5. RTA MATRIX FOR KE3 AND LARGE WCRs (cont.)

KE 3 Nuclear safety

Rationale for importance	**Importance**
Nuclear safety is expected to be included at the policy objectives level or the highest key element contribution level. It has the potential to be a strong differentiator. Retasland Nuclear Safety Commission bases licensing decisions on IAEA Safety Standards, so these are used in scoring where possible	**HIGH (%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
3.11 Fuel storage facility safety	6	Spent fuel facility has large inventory of radioactive material, but protection is not complex	5	5	4	5: Fuel storage facility safety assessment shows large safety margins 3: Fuel storage facility safety assessment shows moderate safety margins 1: Fuel storage facility safety assessment is incomplete or shows low safety margins
3.12 Management system	10	Management system scope is small before procurement and construction begin	5	3	3	5: Technology holder has mature management system that integrates all elements of management 3: Technology holder has limited management system with only partial integration of management elements 1: Technology holder has no coherent management system

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

3.4 Justification NPP2: Robustness of the design is not clear at this stage.
3.5 Justification NPP1: Safety case presented to other MSs shows protection against LOOP.
3.5 Justification NPP3: Passive design has to perform well in LOOP but details not available.
3.6. Justification NPP2: OLCs part complete, but many details missing and will need updating when SAR is complete. DSA for key accidents complete, but full SAR incomplete. Level 1 PSA for internal events is complete, but full PSA incomplete.
3.6.2 Justification NPP3: DSA for key accidents complete, but full SAR incomplete.
3.7 Justification NPP1: Safety Report is accepted by other MSs with similar licensing rules.
3.8 Justification NPP1: Summary of PSA submitted to other MSs shows acceptable results for CDF and LERF.
3.9 Justification NPP1: In-vessel retention of molten core results show small margins to failure.
Justification NPP2: Design shows strong protection of containment function.
Justification NPP3: Very low CDF predicted by PSA so far.
3.10 Justification NPP1: Limited pre-operational programmes though similar NPPs are operating and can provide peer review and operational data.
Justification NPP2: Limited pre-operational programmes, though NPPs with similar designs and components are operating and can provide peer review and operational data.

Justification NPP3: No pre-operational programmes and unique design hampers setting up peer review and operational programmes.

3.12 Justification NPP1: Projects in other MSs give confidence in technology holder's management system.

TABLE A–6. RTA MATRIX FOR KE3 AND SMRs

KE 3 Nuclear safety

Rationale for importance	**Importance HIGH (%)**
Nuclear safety is expected to be included at the policy objectives level or the highest key element contribution level. It has the potential to be a strong differentiator	

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
3.1 Implementation of DiD philosophy	6	DiD is a broad-based concept. The main elements of DiD are inherent in other KEs and KTs	4	3	3	5: Well defined safety programme covering all 5 levels of DiD is well developed 3: Safety programme covering all 5 levels of DiD is in development 1: Safety programme does not cover all 5 levels of DiD or is basic
3.2 Safety design philosophy	2	Both active and passive designs approaches can potentially deliver adequate safety	3	5	4	5: Fully passive approach 3: Mixed active/passive approach 1: Fully active approach
3.3 Degree of diversity and redundancy	2	Overall level of safety will be assessed by DSA and PSA	3	4	3	5: High degree of diversity and redundancy 3: Medium degree of diversity and redundancy 1: Unacceptable degree of diversity and redundancy
3.4 Protection against internal and external hazards	8	Hazards can contribute significantly to overall plant safety	5	3	3	5: Good protection against hazards, well documented 3: Moderate protection against hazards or protection is not fully documented 1: Low protection against hazards or has not been demonstrated
3.5 Response to off-site power loss	2	This is just one postulated accident among many. Importance is already assessed in DSA	4	4	4	5: Nuclear power plant is robust against LOOP 3: Nuclear power plant has moderate protection against LOOP 1: Nuclear power plant is vulnerable to LOOP or protection has not been demonstrated
3.6 Completeness of OLCs, SAR, PSA, O&EPs, SAMGs	10	Moderate weight as document completeness at RTA stage does not strongly impact safety	**Divided into five sub-KTs**			

TABLE A–6. RTA MATRIX FOR KE3 AND SMRs (cont.)

KE 3 Nuclear safety **Rationale for importance** Nuclear safety is expected to be included at the policy objectives level or the highest key element contribution level. It has the potential to be a strong differentiator						**Importance HIGH (%)**
KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
Sub-KT 3.6.1 Safety analysis report	30	DSA part of SAR has to have results for key accidents	5	3	3	5: SAR is almost complete or has been prepared for another Member State 3: SAR covers full scope, but not in detail 1: SAR does not exist or is just an outline
Sub-KT 3.6.2 Operational limits and conditions	10	OLCs not expected to be complete	5	2	3	5: OLCs are almost complete or have been prepared for another Member State 3: OLCs cover full scope, but not in detail 1: OLCs do not exist or are just outlines
Sub-KT 3.6.3 Probabilistic safety assessment	30	Level 1 PSA results have to be available for internal events	5	3	2	5: PSA almost complete or has been prepared for another Member State 3: PSA covers full scope, but not in detail 1: PSA does not exist or is just an outline
Sub-KT 3.6.4 Operating and emergency procedures	10	O&EPs not expected to be complete	5	1	2	5: O&EPs are almost complete or have been prepared for another Member State 3: O&EPs cover full scope, but not in detail 1: O&EPs do not exist or are just outlines
Sub-KT 3.6.5 Severe accident management guidelines	20	SAMGs not expected to be complete	5	3	1	5: SAMGs are almost complete or have been prepared for another Member State 3: SAMGs cover full scope, but not in detail 1: SAMGs do not exist or are just outlines

TABLE A–6. RTA MATRIX FOR KE3 AND SMRs (cont.)

KE 3 Nuclear safety **Rationale for importance** Nuclear safety is expected to be included at the policy objectives level or the highest key element contribution level. It has the potential to be a strong differentiator						**Importance HIGH (%)**
KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
3.7 Results of deterministic safety analysis	18	DSA is a key predictor of overall nuclear power plant safety	5	4	2	5: DSA is advanced and gives acceptable results 3: Key examples of DSA are available and give acceptable results 1: Few examples of DSA are available or do not give confidence that results will be acceptable
3.8 Results of probabilistic safety assessment	18	PSA is a key predictor of overall nuclear power plant safety	4	3	Not Scored	5: Levels 1 and 2 PSA give acceptable results 3: Key parts of PSA give acceptable results 1: PSA results do not give confidence that results will be acceptable **Not Scored:** No criteria to replace CDF currently agreed by the international community
3.9 Mitigation of severe accidents	8	Main indicators are DSA and PSA	4	4	Not Scored	5: Design provides very low probability of severe accident or has strong mitigation measures 3: Design provides balance between severe accident prevention and mitigation 1: Severe accident prevention or mitigation has not been demonstrated **Not Scored**: Not clear if loss of coolant constitutes a severe accident
3.10 Operational expectations affecting safety	10	Becomes very important in late stages, or after RTA	4	4	1	5: Technology holder has mature, pre-operational programmes and supports research and data sharing 3: Technology holder has limited pre-operational programmes and limited research and data sharing 1: Technology holder has no pre-operational programmes and does not participate in research and data sharing

TABLE A–6. RTA MATRIX FOR KE3 AND SMRs (cont.)

KE 3 Nuclear safety **Rationale for importance** Nuclear safety is expected to be included at the policy objectives level or the highest key element contribution level. It has the potential to be a strong differentiator						**Importance HIGH (%)**
KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
3.11 **Fuel storage facility safety**	6	Spent fuel facility has large inventory of radioactive material, but protection is not complex	5	5	4	5: Fuel storage facility safety assessment shows large safety margins 3: Fuel storage facility safety assessment shows moderate safety margins 1: Fuel storage facility safety assessment is incomplete or shows low safety margins
3.12 **Management system**	10	Management system scope is small before procurement and construction begin	3	3	2	5: Technology holder has a mature management system that integrates all elements of management 3: Technology holder has a limited management system with only partial integration of management elements 1: Technology holder has no coherent management system

Not scored:

- NPP1 is an integral PWR with many design concepts and materials shared with large WCRs;
- NPP2 is a helium-cooled reactor with ceramic-coated fuel particles dispersed in a graphite moderator;
- NPP3 is a molten salt reactor with a secondary molten salt loop transferring heat to a steam generator.

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

3.4 Justification NPP1: Mixture of robust design and good segregation.

3.6.x Justification NPP2: OLCs part complete, but many details missing and will need updating when SAR is complete. DSA for key accidents complete, but full SAR incomplete. Level 1 PSA for internal events is complete, but full PSA incomplete. O&EPs in outline only. SAMGs available but verification incomplete.

3.7 Justification NPP1: Safety Report is accepted by other MSs with similar licensing rules.

3.8 Justification NPP2: Level 1 PSA is complete for internal events at power and shutdown. No seismic or fire PSA available.

3.9 Justification NPP1: In-vessel retention of molten core results show small margins to failure.
Justification NPP2: Design shows strong protection of containment function.
Justification NPP3: Not clear if loss of coolant constitutes a severe accident (molten core released into containment).

3.10 Justification NPP1: Limited pre-operational programmes, although similar components are used in operating NPPs and can provide peer review and operational data.

Justification NPP3: No pre-operational programmes and unique design hampers setting up peer review and operational programmes.

3.12 Justification NPP3: Technology holder's management system is extremely limited.

TABLE A–7. RTA MATRIX FOR KE4 AND LARGE WCRs

KE 4 Nuclear island design and performance

Rationale for importance
The nuclear island design and performance can be an important differentiator in all or some of the KTs

Importance HIGH (%)

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
4.1 Plant size	15	The desired electrical output is 1200 Mwe	4	4	2 (*more info*)	5: Within 2% and future up-rating possible 4: Within 10% and future up-rating possible 3: Within 15% 2: Between 1000 and 1400 MWe 1: too big or too small
4.2 Plant availability and capacity factors	5	Depends on operation process and practice than the design itself	5	5	5	5: >95% availability is realistic by operation 3: Design limits >95% availability 1: Design limits >90% availability
4.3 Plant lifetime	5	Low impact relative to other KTs	**Divided into two sub-KTs**			
Sub-KT 4.3.1 Overall plant (design) life	40	Mostly depend on operation and maintenance	3	3	3	5: More than 80 years can be achieved by operation and maintenance 3: Designed for more than 60 years operation 1: Designed for 40 to 60 years operation
Sub-KT 4.3.2 Major component refurbishment	60	Depends on design philosophy	3 (*more info*)	3 (*more info*)	3 (*more info*)	5: No refurbishment during entire plant life 3: Significant (well defined) refurbishment(s) 1: Significant, uncertain and/or very costly refurbishments
4.4 Standardization	10	Advantage from other operators with similar plants	3	5	1	5: More than five similar nuclear power plants are operating already or under construction 3: One to five similar nuclear power plants operating or under construction 1: FOAK
4.5 Simplification	10	Improves reliability and safety	4	5	2 (*more info*)	5: Simplified from the conventional WCRs 3: Same complexity as the conventional WCRs 1: More complicated than the conventional WCRs

TABLE A–7. RTA MATRIX FOR KE4 AND LARGE WCRs (cont.)

KE 4 Nuclear island design and performance

Rationale for importance
The nuclear island design and performance can be an important differentiator in all or some of the KTs

Importance HIGH (%)

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
4.6 Constructability	10	Important factor to reduce project risk (cost and delays)	5	3	3	5: Existence of strong evidence to demonstrate high constructability (track record/ experience, modular design in place, etc.) 3: Fair evidence 1: No evidence
4.7 Operability, inspectability, maintainability and reliability	10	Has to be factored in design	**Divided into two sub-KTs**			
Sub-KT 4.7.1 Operation	50	Good control room design	5	5	5	5: Excellent human–machine interface and control logic 3: Improved over present WCRs 1: Not improved or no information
Sub-KT 4.7.2 Maintenance	50	On-line or ‘smart’ maintenance	3 (*more info*)	1 (*more info*)	1 (*more info*)	5: Designed for mostly on-line maintenance and system testing 3: Improved over present WCRs 1: Not improved or no information
4.8 Manoeuvrability	15	Important factor for co-existence with renewables	4	4	1 (*more info*)	5: Capability of >50% in daily operation 3: >25% 1: <10%
4.9 Plant control and protection architecture	15	Important factor for licensability	5	5	5	5: Existence of strong evidence to demonstrate licensability of control system (licence in country of origin) 3: Fair evidence 1: No evidence
4.10 Radiation protection	5	Regulatory requirement, but not big technical differentiator	3	3	3	5: Radiation protection design features much better than regulatory requirement 3: Design features better than regulatory requirement 1: Design feature to meet regulatory requirement

TABLE A–7. RTA MATRIX FOR KE4 AND LARGE WCRs (cont.)

KE 4 Nuclear island design and performance						
Rationale for importance The nuclear island design and performance can be an important differentiator in all or some of the KTs						**Importance HIGH (%)**
KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

4.1 Question NPP3: Is there any design option to reduce its power to 1200 MWe without any economical penalty?
4.2 Justification NPP1: Please justify why OPEX suggests lower plan availability and capacity factor in the country of origin?
Justification NPP2&3: Please justify why high availability and capacity factor can be expected?
4.3.2 Question NPP1&2&3: What are major components/systems that are expected to be replaced or refurbished during its lifetime?
Justification NPP1&2&3: Only I&C systems are refurbished during plant life.
4.5 Question NPP3: Please provide evidence of how the design is simplified from the conventional WCR of the same kind?
Justification NPP1: Simplified by eliminating a major system and components.
Justification NPP2: Largely simplified with passive safety system.
4.6 Justification NPP2&3: Please explain why the construction delays (past experience) will not happen again in future projects?
4.7.2 Question NPP2&3: To what extent does the plant design accommodate on-line maintenance?
4.8 Question NPP3: What is the power reduction range for daily load following operation?
4.10 Question NPP1&2&3: Are there any radiation protection design features that are more advanced?

TABLE A–8. RTA MATRIX FOR KE4 AND SMRs

KE 4 Nuclear island design and performance						
Rationale for importance The nuclear island design and performance can be an important differentiator in all or some of the KTs and it is more pronounced for SMRs than for large WCRs						**Importance HIGH (%)**
KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
4.1 **Plant size**	20	The desired electrical output is 200 MWe, either one or more modules	3 (*more info*)	4	4	5: Within 2% and future up-rating possible 3: Within 5% less or 15% more 1: Too big or too small
4.2 **Plant availability and capacity factors**	5	Depends on operation process and practice than the design itself	5	5	3	5: >95% availability is realistic by operation 3: Design limits >90% availability 1: Design limits >85% availability
4.3 **Plant lifetime**	5	Low impact relative to other KTs	**Divided into two sub-KTs**			
Sub-KT 4.3.1 Overall plant (design) life	50	Mostly depends on operation and maintenance	3	3	3	5: More than 80 years can be achieved by operation and maintenance 3: Designed for more than 60 years operation 1: Designed for 40 to 60 years operation
Sub-KT 4.3.2 Major component refurbishment	50	Depends on design philosophy	(*more info*)	(*more info*)	(*more info*)	5: No refurbishment during entire plant life 3: Significant (well-defined) refurbishment(s) 1: Significant, uncertain and/or very costly refurbishments
4.4 **Standardization**	10	Advantage for future expansion plans	2	5	3	5: SMR plant is highly standardized with >50% of systems factory produced 3: Well standardized and modularized 1: FOAK or mainly on-site ‘stick-built’
4.5 **Simplification**	10	Improves reliability and safety	5	4	4	5: Largely simplified from the conventional WCRs 3: Same complexity as the conventional WCRs 1: More complicated than the conventional WCRs

TABLE A–8. RTA MATRIX FOR KE4 AND SMRs (cont.)

KE 4 Nuclear island design and performance **Rationale for importance** The nuclear island design and performance can be an important differentiator in all or some of the KTs and it is more pronounced for SMRs than for large WCRs	**Importance HIGH (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
4.6 Constructability	10	Important factor to reduce project risk (cost and delays)	5 (*more info*)	5 (*more info*)	4 (*more info*)	5: Existence of strong evidence for readiness of modular construction 3: Fair evidence 1: No evidence
4.7 Operability, inspectability, maintainability and reliability	20	Has to be factored in the design	**Divided into two sub-KTs**			
Sub-KT 4.7.1 Operation	50	Good control room design	5	5	5	5: Excellent human–machine interface and control logic, etc. 3: Improved over present WCRs 1: Not improved or no information
Sub-KT 4.7.2 Maintenance	50	On-line or ‘smart’ maintenance	Ask vend.	Ask vend.	Ask vend.	5: Designed for mostly on-line maintenance and system testing 4: Ease of maintenance by design 3: Improved over present WCRs 1: Not improved or no information
4.8 Manoeuvrability	5	Important factor for co-existence with renewables	(*more info*)	(*more info*)	(*more info*)	5: Capability of >50% in daily operation 3: >25% 1: <10%
4.9 Plant control and protection architecture	10	Important factor for licensability	(*more info*)	(*more info*)	(*more info*)	5: Existence of strong evidence to demonstrate licensability of control system (licence in country of origin, etc.) 3: Fair evidence 1: No evidence
4.10 Radiation protection	5	Regulatory requirement, but not big technical differentiator	3 (*more info*)	3 (*more info*)	3 (*more info*)	5: Radiation protection design features much better than regulatory requirement 3: Design features meet regulatory requirement 1: Minimum design features to meet regulatory requirement

TABLE A–8. RTA MATRIX FOR KE4 AND SMRs (cont.)

KE 4 Nuclear island design and performance						
Rationale for importance The nuclear island design and performance can be an important differentiator in all or some of the KTs and it is more pronounced for SMRs than for large WCRs						**Importance HIGH (%)**
KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**

Note: info: information; vend.: vendor

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

4.1 Question NPP1: Is there any design option to reduce its power to 200 MWe without any economical penalty?
4.2 Justification NPP1&2: Please justify why high availability and capacity factor can be expected?
4.3.2 Question NPP1&2&3: What are the major components/systems that are expected to be replaced or refurbished during its lifetime?
4.5.1 Justification NPP1: Simplified safety system and natural circulation.
4.6 Question NPP1&2&3: Please explain how the extensive modular construction method is applied and what is your experience in modular construction?
4.7.2 Question NPP1& 2&3: To what extent does the plant design accommodate on-line maintenance?
4.8 Question NPP1&2&3: What is the power reduction range for the daily load following operation?
4.9 Question NPP1&2&3: What is the plan to obtain the licence for the I&C digital system in the country?
4.10 Question NPP1&2&3: Are there any radiation protection design features that are more advanced than regulatory requirements?

TABLE A–9. RTA MATRIX FOR KE5 AND LARGE WCRs

KE 5 Balance of plant design and grid integration

Rationale for importance	**Importance**
Unique or challenging features of the grid arrangement for the balance of plant (BOP) interface in both initial and lifetime operation is critical to the plant's safe, economic and reliable operation. However, BOP performance is secondary to the nuclear reactor performance	**MEDIUM (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
5.1 Net thermal efficiency	30	Important characteristic for plant economy	3	3	5	5: More than 35% net thermal efficiency. 3: 30–35% 1: Less than 30%
5.2 Grid electrical code requirements	10	Not a differentiator since code requirements are to be met	3	2 (*more info*)	2 (*more info*)	5: Better grid response ability than code requirements 3: Able to comply with code requirements 1: Unable to comply with code requirements
5.3 Protection against external hazards	20	External hazards to BOP are secondary	3	2 (*more info*)	2 (*more info*)	5: Much better than the conventional WCRs 3: Same as the conventional WCRs 1: Worse than the conventional WCRs
5.4 Standardization of major components	30	Important for long term cost and component replacement availability	2	1 (*more info*)	2 (*more info*)	5: Majority of major components are standard ones 3: Majority of non-safety class major components are standard ones 1: Majority of major components are specially manufactured ones
5.5 Power requirements from the grid under normal operation	5	Not critical as long as grid code requirements are met	3	2 (*more info*)	2 (*more info*)	5: More robust against grid disturbance than the conventional WCRs 3: Same as the conventional WCRs 1: Worse than the conventional WCRs
5.6 Ability of the grid to accept added generating capacity	5	This is not a differentiator for technologies	3	3	2	5: Special characteristics make it easier for the grid to accept additional capacity 3: Same as the conventional WCRs 1: Special characteristics make it difficult for the grid to accept additional capacity

TABLE A–9. RTA MATRIX FOR KE5 AND LARGE WCRs (cont.)

KE 5 Balance of plant design and grid integration

Rationale for importance	**Importance**
Unique or challenging features of the grid arrangement for the balance of plant (BOP) interface in both initial and lifetime operation is critical to the plant's safe, economic and reliable operation. However, BOP performance is secondary to the nuclear reactor performance	**MEDIUM (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

5.2 Questions NPP2&3: Are there any advanced grid response capabilities, more than the grid code requires?

5.3 Questions NPP2&3: Are there any advanced protection capabilities in BOP against external hazards more than the conventional WCRs?

5.4 Question NPP2&3: What are the major components in safety and non-safety systems that are not standard products from the market?
Justification NPP1: Some of the major components are non-standard.

5.5 Questions NPP2&3: Are there any advanced design features to mitigate influence from grid disturbance compared with the conventional WCRs?

5.6 Justification NPP3: Extra large capacity may make it difficult.

TABLE A–10. RTA MATRIX FOR KE5 AND SMRs

KE 5 Balance of plant design and grid integration

Rationale for importance	**Importance**
Unique or challenging features of the grid arrangement for the balance of plant (BOP) interface in both initial and lifetime operation is critical to the plant's safe, economic and reliable operation. However, BOP performance is secondary to nuclear reactor performance. For this KE, there are many differences between WCRs and SMRs	**MEDIUM (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
5.1 Net thermal efficiency	35	Critical characteristic for plant economy	4 (34% gross)	3 (30% gross)	3 (30% gross)	5: More than 35% net thermal efficiency 3: More than 33% 1: Less than 30%
5.2 Grid electrical code requirements	5	Not a differentiator since code requirements are to be met	2 (*more info*)	1 (*more info*)	2 (*more info*)	5: Better grid response ability than code requirements 3: Able to comply with code requirements 1: Unable to comply with code requirements
5.3 Protection against external hazards	15	External hazards to BOP are secondary	3 (*more info*)	2 (*more info*)	2 (*more info*)	5: Much better than the conventional WCRs 3: Same as the conventional WCRs 1: Worse than the conventional WCRs
5.4 Standardization of major components	35	Important for long term cost and component replacement availability	3 (*more info*)	3 (*more info*)	1 (*more info*)	5: Majority of major components are standard ones 3: Majority of non-safety class major components are standard ones 1: Majority of major components are specially manufactured ones
5.5 Power requirements from the grid under normal operation	5	Not critical as long as grid code requirements are met	2 (*more info*)	3 (*more info*)	3 (*more info*)	5: More robust against grid disturbance than the conventional WCRs 3: Same as the conventional WCRs 1: Worse than the conventional WCRs
5.6 Ability of the grid to accept added generating capacity	5	This is not a differentiator for technologies	4	5	5	5: Special characteristics make it easier for the grid to accept additional capacity 3: Same as the conventional WCRs 1: Special characteristics make it difficult for the grid to accept additional capacity

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual scores:

5.2 Questions NPP1&2&3: Are there any advanced grid response capabilities, more than the grid code requires?

5.3 Questions NPP1&2&3: Are there any advanced protection capabilities in BOP against external hazards, more than the conventional WCRs?

5.4 Question NPP1&2&3: What are major components in safety and non-safety systems that are not standard products from the market?

5.5 Questions NPP1&2&3: Are there any advanced design features to mitigate influence from grid disturbance compared with the conventional WCRs?

5.6. Justification NPP2&3: Smaller capacity makes it easier.

TABLE A–11. RTA MATRIX FOR KE6 AND LARGE WCRs

KE 6 Balance of plant design for purposes other than electricity production

Rationale for importance	**Importance**
Large WCRs for Retasland are needed only for electricity production, but this may change in the future. Therefore, the overall importance of KE 6 in this case is low. An example here is evaluated based on the use of a large nuclear power plant for cogeneration of electricity (~75%) and district heating (~25%)	**LOW** **(%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
6.1 **Net thermal efficiency**	5	Net thermal efficiency is not very important in low-temperature district heating	3	4	2	5: Net thermal efficiency ≥40% 4: Net thermal efficiency ≥35% 3: Net thermal efficiency ≥30% 2: Net thermal efficiency ≤30% 1: Net thermal efficiency ≤25%
6.2 **Compatibility with local use requirements**	20	Local use requirements are to be met, even if some redesign is needed	2 (*more info*)	5	4	5: Nuclear power plant is compatible with the expected local use requirements over the lifetime of the nuclear power plant 4: Nuclear power plant is compatible with the current local use requirements and flexible for future upgrades 3: The local use requirements can be met 2: The local use requirements will only be met with additional input from other energy sources 1: No information on the nuclear power plant's ability to provide anything other than electricity production is provided
6.3 **Protection against external hazards**	20	Protection against external hazards is the safety criterion of the plant and is thus of high importance	3 (*more info*)	5	1	5: BOP greatly satisfies the highest standards in different national regulations 4: BOP satisfies the highest standards for the site 3: No regulatory standards are set yet, but the nuclear power plant seems to satisfy the standards on similar regions 2: BOP can only meet the required standards with additional upgrades 1: No information is provided

KE 6 Balance of plant design for purposes other than electricity production

Rationale for importance	Importance
Large WCRs for Retasland are needed only for electricity production, but this may change in the future. Therefore, the overall importance of KE 6 in this case is low. An example here is evaluated based on the use of a large nuclear power plant for cogeneration of electricity (~75%) and district heating (~25%)	LOW (%)

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
6.4 Standardization of major components	10	Standardized components increase safety and decrease costs	3 (*more info*)	5 (*more info*)	2	5: BOP is the NOAK from the technology holder and all components are fabricated following national standards of this technology holder 4: BOP is the NOAK from the technology holder, using different suppliers or outsourcing production 3: BOP is the FOAK; the technology holder plans standardization but does not seem to have any experience with this yet 2: BOP is a new design 1: No information is provided
6.5 Electrical power requirements	20	Electrical power needs for continued non-electric production (district heating) is a differentiator	3	2	2	5: Nuclear power plant is self-sufficient for operation and does not need shutdown for 24 hr grid interruption 4: Nuclear power plant can continue all operations for 12 hr 3: Nuclear power plant can continue operation, but no pumping for district heat system 2: Nuclear power plant relies on the grid for continued operation 1: No information provided

TABLE A–11. RTA MATRIX FOR KE6 AND LARGE WCRs (cont.)

KE 6 Balance of plant design for purposes other than electricity production

Rationale for importance	**Importance**
Large WCRs for Retasland are needed only for electricity production, but this may change in the future. Therefore, the overall importance of KE 6 in this case is low. An example here is evaluated based on the use of a large nuclear power plant for cogeneration of electricity (~75%) and district heating (~25%)	**LOW (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
6.6 Demand following and storage capabilities	10	BOP needs to accommodate heat use fluctuations on a daily and seasonal basis, with daily heat storage up to 50% capacity	4	2	1	5: Nuclear power plant can provide load following and BOP storage capabilities for the next 20 years to come. 3: Nuclear power plant can provide limited, slow load following BOP storage capabilities for current energy demand 2: Nuclear power plant cannot fully meet load following and storage capabilities for current energy demand 1: No information provided
6.7 Maximum output capacity (heat equivalent and quality)	5	Increased heating demand is expected over the next decades but could be augmented by other means It is thus not of highest importance here	4	5	4	5: Nuclear power plant capacity exceeds the demand and can provide the excess output to other users 4: Nuclear power plant just exceeds the demand 3: Nuclear power plant can supply a large share of the demanded output 2: Nuclear power plant cannot supply the required output 1: No information provided
6.8 Integrated energy systems	10	Integrated energy systems are strongly considered to allow optimized use of regional energy sources, gain public approval and increase overall grid stability	3	5	3	5: Nuclear power plant has already been successfully built and operated as a stand-alone and integrated system (direct coupling) 4: Nuclear power plant has already been built and indirectly coupled with other energy source 3: Nuclear power plant can theoretically be integrated with other energy sources 2: Nuclear power plant cannot be integrated with other energy sources 1: No information provided

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

6.2 Questions NPP1: Can the design be configured (easily) to provide 25% energy as heat at 125ºC?

6.3 Questions NPP1: What additional protection is needed for a 25% cogeneration BOP design?

6.4 Question NPP2&3: What are the major components in safety and non-safety systems that are not standard products from the market?
Justification NPP1: Technology holder is a start-up with a clever idea.
Justification NPP2: Technology holder has a long tradition and has built several similar plants over the past five years.

6.6 Justification NPP1: NPP shows adequate load following and storage capabilities.

6.8 Justification NPP1: Plans exist to integrate the NPP with a solar thermal power plant.
Justification NPP2: NPP exists as a stand-alone plant and as an integrated system with several large windfarms.

TABLE A–12. RTA MATRIX FOR KE6 AND SMRs

KE 6 Balance of plant design for purposes other than electricity production

Rationale for importance S/M Retasland requires 200 MW of electricity plus cogeneration of process heat and hydrogen. Therefore, the non-electric BOP capabilities are important and ranked MEDIUM	**Importance** **MEDIUM** **(%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
6.1 **Net thermal efficiency**	20	High net thermal efficiency implies high quality heat, which is important for hydrogen production and process heat	3	4	2	5: Net thermal efficiency ≥60% 4: Net thermal efficiency ≥45% 3: Net thermal efficiency ≥40% 2: Net thermal efficiency ≤40% 1: Net thermal efficiency ≤35%
6.2 **Compatibility with local use requirements**	10	Local use requirements are to be met, even if some redesign is needed	2 (*more info*)	5	1	5: Nuclear power plant is compatible with the expected local use requirements over the lifetime of the nuclear power plant 4: Nuclear power plant is compatible with the current local use requirements and flexible for future upgrades 3: The local use requirements can be met 2: The local use requirements will only be met with additional input from other energy sources 1: No information on the nuclear power plant's ability to provide anything other than electricity production is provided
6.3 **Protection against external hazards**	25	Protection against external hazards is a safety criterion of the plant and thus of high importance May need special consideration of the risk associated with the hydrogen plant	2 (*more info*)	5	4	5: BOP greatly satisfies the highest standards in different national regulations 4: BOP satisfies the highest standards for the site 3: No regulatory standards are set yet, but the nuclear power plant seems to satisfy the standards on similar regions 2: BOP can only meet the required standards with additional upgrades 1: No information is provided

TABLE A–12. RTA MATRIX FOR KE6 AND SMRs (cont.)

KE 6 Balance of plant design for purposes other than electricity production

Rationale for importance	Importance
S/M Retasland requires 200 MW of electricity plus cogeneration of process heat and hydrogen. Therefore, the non-electric BOP capabilities are important and ranked MEDIUM	**MEDIUM (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
6.4 Standardization of major components	10	Standardized components increase safety and decrease costs	3	5	4	5: BOP is the NOAK from the technology holder and all components are fabricated following national standards of this technology holder 4: BOP is the NOAK from the technology holder, using different suppliers or outsourcing production 3: BOP is the FOAK. Technology holder plans standardization but does not seem to have any experience with this yet 2: BOP is a new design 1: No information is provided
6.5 Electrical power requirements	15	Electrical power requirements are a differentiator for hydrogen production (incl. during grid interruption) and thus of high importance	4	4	2	5: Nuclear power plant is self-sufficient for operation and does not need shutdown for 24 hr grid interruption 4: Nuclear power plant can continue all operations for 12 hr 3: Nuclear power plant can continue operation, but no pumping for district heat system 2: Nuclear power plant relies on the grid for continued operation 1: No information provided
6.6 Demand following and storage capabilities	5	Cogeneration needs are fairly steady over a day and also seasonally, but this may change in the future as this is FOAK	4	2	3	5: Nuclear power plant can provide load following and BOP storage capabilities for the next 20 years to come 3: Nuclear power plant can provide limited, slow load following with BOP storage capabilities for current energy demand 2: Nuclear power plant cannot fully meet load following and storage capabilities for current energy demand 1: No information provided

TABLE A–12. RTA MATRIX FOR KE6 AND SMRs (cont.)

KE 6 Balance of plant design for purposes other than electricity production

Rationale for importance S/M Retasland requires 200 MW of electricity plus cogeneration of process heat and hydrogen. Therefore, the non-electric BOP capabilities are important and ranked MEDIUM	**Importance MEDIUM (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
6.7 Maximum output capacity (heat equivalent and quality)	5	Increased demand for hydrogen and process heat is expected over the next decades but could be augmented by other means It is thus not of highest importance here	4	5	2	5: Nuclear power plant capacity exceeds the demand and can provide the excess output to other users 4: Nuclear power plant just exceeds the demand 3: Nuclear power plant can supply a large share of the demanded output 2: Nuclear power plant cannot supply the required output. 1: No information provided
6.8 Integrated energy systems	10	Integrated energy systems allow optimized use of regional energy sources, gain public approval and increase overall grid stability	3	5	3	5: Nuclear power plant has already been successfully built and operated as a stand-alone and integrated system (direct coupling) 4: Nuclear power plant has already been built and indirectly coupled with another energy source 3: Nuclear power plant can theoretically be integrated with other energy sources 2: Nuclear power plant cannot be integrated with other energy sources 1: No information provided

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

6.2 Questions NPP1: Can the design be configured (easily) to provide heat to industrial processes?

6.3 Questions NPP1: What additional protection is needed for a BOP design with nearby hydrogen production, and any other associated facility, connected to the BOP?

6.8 Justification NPP1: Plans exist to integrate the NPP with a solar thermal power plant.
Justification NPP2: NPP exists as a stand-alone plant and as an integrated system with several large windfarms.

TABLE A–13. RTA MATRIX FOR KE7 AND LARGE WCRs

KE 7 Safeguards and protection

Rationale for importance	**Importance**
IAEA safeguards will be applied to materials and facilities by Member State or Member States. The large WCR designs may not differentiate from a safeguards point of view. Although there may be differences in the details of the security plan and systems, it is expected that site security will be achieved by the responsible authorities	**LOW (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
7.1 Safeguards by design	20	Consideration of safeguards agreements and their implementation is needed in the design phase	**Divided into two sub-KTs**			
Sub-KT 7.1.1 Ease of design verification during construction	60	Design verification is important to ensure compliance with IAEA treaties and conventions	5	2 (*more info*)	2 (*more info*)	5: Safeguards design information is fully provided 3: Safeguards design information is fairly provided 1: No information or inadequate information
Sub-KT 7.1.2 Consideration of IAEA safeguards equipment installation and power requirements	40	IAEA safeguards equipment are to be properly accommodated for in the design	4	5	2	5: Adequate accommodation of IAEA safeguards equipment in the design 3: Fair accommodation of IAEA safeguards equipment in the design 1: No information or inadequate accommodation
7.2 Special nuclear materials (SNMs) management	30	Provisions for SNM accounting and control are critical in safeguards	**Divided into two sub-KTs**			
Sub-KT 7.2.1 Ease of SNM verification during fuelled operation	50	Design features related to safeguards are important in SNM verification	4	4	3 (*more info*)	5: Adequate safeguards measures accounted for in the design 3: Fair safeguards measures accounted for in the design 1: No information or inadequate measures
Sub-KT 7.2.2 Provision for remote monitoring of operating parameters and operational procedures	50	Continuous and remote monitoring of reactor operations is desired in SNM accounting and control and provides a more complete and timely coverage of nuclear material movements at the facility	5	4	4	5: Adequate monitoring of reactor operations accounted for in the design 3: Fair monitoring of reactor operations accounted for in the design 1: No information or inadequate measures

TABLE A–13. RTA MATRIX FOR KE7 AND LARGE WCRs (cont.)

KE 7 Safeguards and protection

Rationale for importance	**Importance**
IAEA safeguards will be applied to materials and facilities by Member State or Member States. The large WCR designs may not differentiate from a safeguards point of view. Although there may be differences in the details of the security plan and systems, it is expected that site security will be achieved by the responsible authorities	**LOW** **(%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
7.3 Physical protection of the nuclear power plant	30	The physical protection of the nuclear power plant is to be thoroughly considered in the design phase	**Divided into two sub-KTs**			
Sub-KT 7.3.1 Security plans and facilities design	40	The security plans and facilities design are important in defending against security threats	2 (*more info*)	2 (*more info*)	5	5: The security plans and facilities design are comprehensive 3: The security plans and facilities design are fair 1: Inadequate information on the security plans and facilities design
Sub-KT 7.3.2 Prevention, detection and response measures	60	These measures are most critical in defining the security measures and their efficacy	2 (*more info*)	2 (*more info*)	4	5: The measures are thorough and well suited for the site 3: The measures are fair and somewhat suitable for the site 1: Inadequate information provided on the different measures
7.4 Cybersecurity protection of the nuclear power plant	20	Cybersecurity is critical for the safe and secure operation of the facility	3 (*more info*)	3 (*more info*)	5	5: The cybersecurity and information security plans are comprehensive 3: The cybersecurity and information security plans are adequate 1: No information or the cybersecurity and information security plans are inadequate

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

7.1.1 Justification NPP1: Information regarding compliance with the IAEA CSA and Additional Protocol were fully provided.
Question NPP2: What measures are taken to implement and comply with the provisions in the IAEA Additional Protocol?
Question NPP3: How are the receiving area and spent fuel storage monitored?

7.2.1 Information NPP3: Provide more information on the material balance area for the receiving and fresh fuel storage areas

7.3.1 Question NPP1: What is the DBT and how was it determined?

Question NPP2: How is the defence in depth approach demonstrated in the security facilities?

7.3.2 Question NPP1: How does the nearby city of L-Retasland site affect the mitigation of malicious acts?

Question NPP2: How are the access control systems integrated with one another?

Justification NPP3: The prevention and detection measures were thoroughly described. The response measures were briefly described. It is understood that the response measures will be taken by the responsible authorities.

7.4. Question NPP1: How is the access to critical systems restricted?

Question NPP2: How is the availability of information maintained with loss of power?

TABLE A–14. RTA MATRIX FOR KE7 AND SMRs

KE 7 Safeguards and protection

Rationale for importance	**Importance**
IAEA safeguards will be applied to materials and facilities placed under Agency safeguards by a Member State or Member States. This aspect maybe a differentiator among the SMRs, and especially for the FOAK. Although there may be differences in the details of the security plan and systems, it is expected that site security will be achieved by the responsible authorities	**LOW TO MEDIUM (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
7.1 Safeguards by design	20	Consideration of safeguards agreements and their implementation is needed in the design phase	**Divided into three sub-KTs**			
Sub-KT 7.1.1 Ease of design verification during construction	50	Design verification is important to ensure compliance with IAEA treaties and conventions	5	4	5	5: Safeguards design information is fully provided 3: Safeguards design information is fairly provided 1: No information or inadequate information
Sub-KT 7.1.2 Consideration of IAEA safeguards equipment installation and power requirements	30	IAEA safeguards equipment is to be properly accommodated in the design	4	5	4	5: Adequate accommodation of IAEA safeguards equipment in the design 3: Fair accommodation of IAEA safeguards equipment in the design 1: No information or inadequate accommodation
Sub-KT 7.1.3 SMR related issues in SBD	20	Additional measures and considerations taken for SMRs	3 (*more info*)	4 (*more info*)	3 (*more info*)	5: Adequate measures and considerations taken 3: Fair measures and considerations taken 1: No information or inadequate measures taken
7.2 Special nuclear materials (SNMs) management	30	Provisions for SNM accounting and control are critical in safeguards	**Divided into three sub-KTs**			
Sub-KT 7.2.1 Ease of SNM verification during fuelled operation	40	Design features related to safeguards are important in SNM verification	5	5	5	5: Adequate safeguards measures accounted for in the design 3: Fair safeguards measures accounted for in the design 1: No information or inadequate measures

TABLE A–14. RTA MATRIX FOR KE7 AND SMRs (cont.)

KE 7 Safeguards and protection

Rationale for importance	**Importance**
IAEA safeguards will be applied to materials and facilities placed under Agency safeguards by a Member State or Member States. This aspect maybe a differentiator among the SMRs, and especially for the FOAK. Although there may be differences in the details of the security plan and systems, it is expected that site security will be achieved by the responsible authorities	**LOW TO MEDIUM (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
Sub-KT 7.2.2 Provision for remote monitoring of operating parameters and operational procedures	40	Continuous and remote monitoring of reactor operations is desired in SNM accounting and control and provides a more complete and timely coverage of nuclear material movements at the facility	5	5	5	5: Adequate monitoring of reactor operations accounted for in the design 3: Fair monitoring of reactor operations accounted for in the design 1: No information or inadequate measures
Sub-KT 7.2.3 SMR related issues in SNM management	20	Additional measures and considerations taken for SMRs	4	3 (*more info*)	4	5: Adequate measures and considerations taken 3: Fair measures and considerations taken 1: No information or inadequate measures taken
7.3 Physical protection of the nuclear power plant	30	The physical protection of the nuclear power plant is to be thoroughly considered in the design phase	**Divided into three sub-KTs**			
Sub-KT 7.3.1 Security plans and facilities design	30	The security plans and facilities design are important in defending against security threats	5	4	5	5: The security plans and facilities design are comprehensive 3: The security plans and facilities design are fair 1: Inadequate information on the security plans and facilities design
Sub-KT 7.3.2 Prevention, detection and response measures	50	These measures are most critical in defining the security measures and their efficacy	4	5	5	5: The measures are thorough and well suited for the site 3: The measures are fair and somewhat suitable for the site 1: Inadequate information provided on the different measures

KE 7 Safeguards and protection

Rationale for importance	Importance
IAEA safeguards will be applied to materials and facilities placed under Agency safeguards by a Member State or Member States. This aspect maybe a differentiator among the SMRs, and especially for the FOAK. Although there may be differences in the details of the security plan and systems, it is expected that site security will be achieved by the responsible authorities	**LOW TO MEDIUM (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
Sub-KT 7.3.3 SMR related issues in physical protection	20	Additional measures and considerations taken for SMRs	3 (*more info*)	5	5	5: Adequate measures and considerations taken 3: Fair measures and considerations taken 1: No information or inadequate measures taken
7.4 Cybersecurity protection of the nuclear power plant	20	Cybersecurity is critical for the safe and secure operation of the facility	5	4	4	5: The cybersecurity and information security plans are comprehensive 3: The cybersecurity and information security plans are adequate 1: No information or the cybersecurity and information security plans are inadequate

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

7.1.3 Question NPP1: How are safeguards measures ensured during the modular construction?
Question NPP2: How is the fresh fuel transition monitored from the receiving area to the different units?
Question NPP3: How are safeguards measures ensured with future additional units on site?

7.2.3 Question NPP2: How is the continuity and synchronization of the operating parameters from the different units ensured?

7.3.3 Question NPP1: How does the remote location of S/M Retasland affect the physical protection measures? How does the weather of S/M Retasland site affect the physical protection measures?

TABLE A–15. RTA MATRIX FOR KE8 AND LARGE WCRs

KE 8 Technology readiness

Rationale for importance	**Importance LOW (%)**
Not a strong differentiator since all large WCRs under consideration are at a high level of technological readiness	

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
8.1 Readiness of the SMR design	0	**Not applicable for RTA for large WCRs**				
8.2 Licensing and/or certification status for the design	90	The proposed plants have to be licensable in the vendor's country	5	4	3	5: The design is licensed with recognized approaches 4: The design is licensed with different approaches 3: The design is licensed 2: Design licensing is ongoing 1: The design needs to be licensed
8.3 Language	10	It is a benefit if design information, licensing documents, and operating, training and maintenance procedures are available in the right language	5	3	1 (*more info*)	5: Technology owner and operator have the same language and documentation is abundant 4: Technology owner and operator have the same language 3: Documentation of the technology owner is translated and already used 2: Documentation of the technology owner is translated 1: Documentation is not translated

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual nuclear power plant scores:

8.1.1 Justification NPP1 & NPP2: Many assessment tests of this highly proven fuel design have been done.
Justification NPP3: Assessments and tests are necessary for this evolutionary fuel design.

8.2 Justification NPP1: The design is licensed in four countries with different recognized approaches.
Justification NPP2: The design is licensed in three countries with different approaches.
Justification NPP3: The design is licensed in one country.

8.3 Question NPP3: Have you planned to translate the documentation?

TABLE A–16. RTA MATRIX FOR KE8 AND SMRs

KE 8 Technology readiness

Rationale for importance Verification of the technological readiness of an SMR is important in order to secure its completion of the project on time and on budget for long term, safe, economic and reliable operation	**Importance HIGH (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
8.1 Readiness of the SMR design	60	There are three basic categories of advanced reactor design: proven, evolutionary or innovative Even if the category of ‘proven designs’ is not relevant for SMR so far, they can use proven components from other industries	**Divided into three sub-KTs**			
Sub-KT 8.1.1 Fuel	35	Fuel assembly is the first safety barrier	4 (*more info*)	2	3	5: This fuel design is proven 4: This fuel design is partially evolutionary 3: This fuel design is evolutionary 2: This fuel design is innovative 1: This fuel design is highly innovative
Sub-KT 8.1.2 NSSS	50	NSSS is the most important part of the nuclear power plant for safety and performance	4	2	3	5: This NSSS design is partially proven 4: This NSSS design is evolutionary 3: This NSSS is highly evolutionary 2: This NSSS design is innovative 1: This NSSS design is highly innovative
Sub-KT 8.1.3 BOP	15	A part of the design of the BOP is not particular to nuclear power plants.	4	1 (*more info*)	4	5: This BOP design is proven 4: This BOP design is partially proven 3: This BOP design is partially evolutionary 2: This BOP design is evolutionary 1: This BOP design is innovative

TABLE A–16. RTA MATRIX FOR KE8 AND SMRs (cont.)

KE 8 Technology readiness

Rationale for importance Verification of the technological readiness of an SMR is important in order to secure its completion of the project on time and on budget for long term, safe, economic and reliable operation	**Importance** **HIGH** **(%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
8.2 **Licensing and/or certification status for the design**	30	The proposed plants have to be licensable in the vendor's country	3	2	4	5: The design is licensed 4: Design licensing is ongoing 3: The design needs to be licensed 2: The evolutionary design needs to be licensed 1: The innovative design needs to be licensed
8.3 **Language**	10	Design and licensing documents, operating, training and maintenance procedures are to be understandable	3	2	5	5: Original documentation is in Retasland language 4: Full set of documents is translated and already used 3: Full set of documents is translated 2: Documentation is partially translated 1: Documentation is not translated

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

8.1.1 Questions NPP1: Is this fuel design also evolutionary for large WCRs?
If yes, can you also use the work done on large reactors?
Questions NPP2, NPP3: Can you clarify what is a similar fuel design? Similar pellets, rods, grids?

8.1.2 Justification NPP1: Confirmatory tests are necessary.
Justification NPP2: Substantial R&D, feasibility tests and a prototype/demonstration plant are required.
Justification NPP3: Many engineering actions and many confirmatory tests are necessary.

8.1.3 Question NPP2: What are the radical conceptual changes in design approaches or system configuration of the innovative design in comparison with the existing design?

TABLE A–17. RTA MATRIX FOR KE9 AND LARGE WCRs and SMRs

KE 9 Project delivery

Rationale for importance	Importance
This KE is very relevant in Phase 2 (both L-Retasland and S/M Retasland) and details have to be available from the technology holders or need to be obtained. Importance ranking would be lower (even zero) during Phase 1. This is a major national project of high priority and visibility	**MEDIUM** **(%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
9.1 Owner/operator scope of supply	10	Nuclear power plant will be supplied in its entirety Site preparation only by owner	5	4	3	5: Scope well defined with cost estimates and project risk analysis 4: Project risks are not evaluated 3: Interfaces descriptions are missing 2: Cost estimates are missing 1: Owner/operator scope is not defined
9.2 Supplier/technology holder issues	25	Good suppliers/ technology holder relationship is crucial Well established supply chain desired	4	3	3	5: Relationship suppliers/holder well defined 4: Supply chain is described but experiences description is missing 3: Supply chain and qualification not described 2: Warranties are missing, QA and QC missing 1: No descriptions or no supply chain exists
9.3 Project schedule capability	20	This is a major national project of high priority and visibility Reliable project management is crucial	5	2	3	5: Project schedule well defined; experienced vendor with good track record 4: Project schedule well defined; no history with significant delay 3: Project schedule well defined, significant delays observed in the past 2: High level schedule only 1: Project schedule is not defined
9.4 Technology transfer and technological support	5	Technology transfer is not of high importance for the first nuclear power plant but is desired for follow up units	5	4	4	5: Technology transfer well defined 4: Technology transfer defined, but for software and design tool not defined 3: Transfer only for BOP 2: Transfer only for construction 1: Technology transfer is not offered

TABLE A–17. RTA MATRIX FOR KE9 AND LARGE WCRs and SMRs (cont.)

KE 9 Project delivery

Rationale for importance	**Importance**
This KE is very relevant in Phase 2 (both L-Retasland and S/M Retasland) and details have to be available from the technology holders or need to be obtained. Importance ranking would be lower (even zero) during Phase 1. This is a major national project of high priority and visibility	**MEDIUM** **(%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
9.5 **Project contracting option**	10	Multiple project contracting options can be considered	4	5	5	5: All contracting options offered 4: Experience with several contracting options 3: Acceptable contracting option is achievable 2: No flexibility for different models 1: No information about contracting option available
9.6 **Services offered for the front end of fuel cycle (fresh fuel supply)**	15	Reliable fuel supply, ideally from diverse suppliers	4	4	1	5: Fuel services well defined 4: No warranties for fresh fuel 3: No evidence of manufacturing, testing and operational experience for proposed fuel 2: No fuel cycle services (including core management) offered 1: No information or long term fuel supply services not offered
9.7 **Services offered for the back end of fuel cycle (spent fuel management)**	15	Spent fuel on-site is needed for the entire nuclear power plant life Final disposal not yet decided	2	2	2	5: Spent fuel services well defined, including take-back 4: Spent fuel capacity is sufficient to end of life (EOL) 3: Limited size spent fuel storage facility (not to the plant EOL); need for dry storage included as project option 2: Limited size spent fuel storage facility (not to the plant EOL); need for dry storage not included in contract 1: No information available or service for spent fuel management is not offered

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

9.1 Justification: NPP2 missed describing possible project risks in detail but NPP3 did not describe interfaces with owner at all.

9.2 Justification: NPP1 missed describing supplier's experiences but NPP2 and NPP3 did not provide information about existing supply chain and qualification of suppliers.

9.3 Justification: NPP2 did not provide cost estimates in detail while NPP3 did not describe the subcontractor's interfaces during the project.

9.4 Justification: NPP2 and NPP3 did not offer either the technology transfer of design tools or necessary software.

9.5 Justification: NPP1 did not offer any penalty for possible project delay time and/or extra-budgetary expenses.

9.6 Justification: NPP3 did not offer long term fuel supply services until EOL of plant. NPP1 and 2 did not offer warranties for fresh fuel delivery. Both plants have to justify warranties and additional emergency refurbishment if fresh fuel has some discrepancies.

9.7 Justification: All three plants offer the spent fuel pool for only 15 years of full power operations. The technology holder has to describe how the plant will manage spent fuel until plant EOL and who will take responsibility for this management.

TABLE A–18. RTA MATRIX FOR KE10 AND LARGE WCRs

KE 10 Economics and financing

Rationale for importance The nuclear island design and performance can be an important differentiator in all or some of the KTs and it is more pronounced for SMRs than for large WCRs	**Importance** **HIGH** **(%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
10.1 **Capital costs**	33	Capital costs are the main component of nuclear energy LCOE Upfront investments are high	4 (*more info*)	2	3	5: Capital costs are highly optimized, clearly estimated and controlled (identified and experienced supply chain, experienced feedback) 4: Capital costs are optimized, clearly estimated and controlled (identified and experienced supply chain, experienced feedback) 3: Capital costs are clearly estimated and controlled (identified and experienced supply chain, experienced feedback) 2: Capital costs are clearly estimated 1: Capital costs are not clearly estimated and with no experienced feedback
10.2 **O&M costs**	22	Complex operation and frequent maintenance can reduce capacity factor and be costly	4 (*more info*)	3 (*more info*)	2	5: O&M are simplified; maintenance is not frequent; their costs are optimized and moderate 4: O&M are simplified; their costs are optimized and moderate 3: O&M are simplified; their costs are optimized 2: O&M are costly 1: O&M are complex and with high costs
10.3 **Fuel costs**	11	Fuel costs are a relatively small part of the cost of nuclear energy	4 (*more info*)	3	2	5: Very competitive fuel is broadly available for this reactor 4: Competitive fuel is broadly available for this reactor 3: Competitive fuel is available for this reactor 2: Competitive fuel is not widely available for this reactor 1: Competitive fuel is little or not available for this reactor

TABLE A–18. RTA MATRIX FOR KE10 AND LARGE WCRs (cont.)

KE 10 Economics and financing

Rationale for importance The nuclear island design and performance can be an important differentiator in all or some of the KTs and it is more pronounced for SMRs than for large WCRs	**Importance HIGH (%)**

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
10.4 Spent fuel management costs	7	Spent fuel management costs are a small part of the cost of nuclear energy	5	2	4	5: Long wet or dry storage solutions available with moderate costs; reprocessing is assessed 4: Long wet or dry storage solutions available with moderate costs; reprocessing is partially assessed 3: Long wet or dry storage solutions available with costs assessed; reprocessing is not assessed 2: Long wet or dry storage solutions available with costs not assessed; reprocessing is not assessed 1: Storage solutions costly; reprocessing not assessed
10.5 Decommissioning costs	10	Decommissioning costs are not the most important Their impact is reduced if they are well anticipated and provisioned	5	4	3	5: Decommissioning works are evaluated, and costs are moderate and clearly estimated 4: Decommissioning works are evaluated, and costs are moderate 3: Decommissioning works are evaluated, and costs are not assessed 2: Decommissioning works are complex, and costs are not assessed 1: Decommissioning works are complex and high costs are clearly estimated

TABLE A–18. RTA MATRIX FOR KE10 AND LARGE WCRs (cont.)

KE 10 Economics and financing

Rationale for importance	**Importance**
The nuclear island design and performance can be an important differentiator in all or some of the KTs and it is more pronounced for SMRs than for large WCRs	**HIGH (%)**

KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
10.6 Financing	17	Because of the high cost of construction, a difference in the cost of financing can have a large impact The numbers and quality of shareholders and lenders could depend on the technology holder	4 (*more info*)	1	3	5: Many shareholders and many lenders with guarantees are available for a competitive financing 4: Shareholders and lenders with guarantees are available for a competitive financing 3: Shareholders and lenders are available for the financing 2: Few shareholders and lenders are available for the financing 1: Only national shareholders and lenders are available for the financing

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

10.1 Question NPP1: Can you explain how you optimize the capital cost?
10.2 Question NPP1 and NPP2: Can you explain how you optimize the operation and maintenance costs?
10.3 Question NPP1: How can you guarantee fuel competitiveness?
10.4 Question NPP3: Do you plan to study and demonstrate the possibility of reprocessing the spent fuel?
10.6 Question NPP1 and NPP3: Can you identify the shareholders and lenders? How can they be involved in the project?

TABLE A–19. RTA MATRIX FOR KE10 AND SMRs

KE 10 Economics and financing **Rationale for importance** The nuclear island design and performance can be an important differentiator in all or some of the KTs and it is more pronounced for SMRs than for large WCRs						**Importance HIGH (%)**
KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
10.1 Capital costs	28	Capital costs are the main cost of nuclear energy Upfront investment is less in comparison with large WCRs	4	2	3	5: Capital costs are highly optimized, clearly estimated and controlled (identified and experienced supply chain, experienced feedback) 4: Capital costs are optimized, clearly estimated and controlled (identified and experienced supply chain, experienced feedback) 3: Capital costs are clearly estimated and controlled (identified and experienced supply chain, experienced feedback) 2: Capital costs are estimated but with low experienced feedback 1: Capital costs are not clearly estimated and with no experienced feedback
10.2 O&M costs	25	Complex operation and frequent maintenance can reduce the capacity factor and be costly	4	2	3	5: O&M are simplified; maintenance is not frequent; their costs are optimized and moderate 4: O&M are simplified; their costs are optimized and moderate 3: O&M are simplified; their costs are optimized 2: O&M are costly 1: O&M are complex and with high costs
10.3 Fuel costs	18	Fuel costs are a small part of the cost of nuclear energy	4	3	5	5: Very competitive fuel is broadly available for this reactor 4: Competitive fuel is broadly available for this reactor 3: Competitive fuel is available for this reactor 2: Competitive fuel is not widely available for this reactor 1: Competitive fuel is scarce or not available for this reactor

TABLE A–19. RTA MATRIX FOR KE10 AND SMRs (cont.)

KE 10 Economics and financing

Rationale for importance
The nuclear island design and performance can be an important differentiator in all or some of the KTs and it is more pronounced for SMRs than for large WCRs

Importance HIGH (%)

KTs	%	Rationale for weights	NPP1 score	NPP2 score	NPP3 score	Rationale for scores
10.4 Spent fuel management costs	7	Spent fuel management costs are a small part of the cost of nuclear energy	5	3	4	5: Long wet or dry storage solutions available with moderate costs; reprocessing is assessed 4: Long wet or dry storage solutions available with moderate costs; reprocessing is partially assessed 3: Long wet or dry storage solutions available with costs assessed; reprocessing is not assessed 2: Long wet or dry storage solutions available with costs not assessed; reprocessing is not assessed 1: Storage solutions costly; reprocessing not assessed
10.5 Decommissioning costs	10	Decommissioning costs are not the most important factor Their impact is reduced if they are well anticipated and provisioned	4	3 *(more info)*	5	5: Decommissioning works are evaluated and costs are moderate and clearly estimated 4: Decommissioning works are evaluated and costs are moderate 3: Decommissioning works are evaluated and costs are not assessed 2: Decommissioning works are complex and costs are not assessed 1: Decommissioning works are complex and high costs are clearly estimated

TABLE A–19. RTA MATRIX FOR KE10 AND SMRs (cont.)

KE 10 Economics and financing **Rationale for importance** The nuclear island design and performance can be an important differentiator in all or some of the KTs and it is more pronounced for SMRs than for large WCRs						**Importance HIGH (%)**
KTs	**%**	**Rationale for weights**	**NPP1 score**	**NPP2 score**	**NPP3 score**	**Rationale for scores**
10.6 Financing	12	Because of the high costs of construction, a difference in the cost of financing can have a large impact The numbers and quality of shareholders and lenders could depend on the technology holder	5 *(more info)*	2	4 *(more info)*	5: Many shareholders and many lenders with guarantees are available for a competitive financing 4: Shareholders and lenders with guarantees are available for a competitive financing 3: Shareholders and lenders are available for the financing 2: Few shareholders and lenders are available for the financing 1: Only national shareholders and lenders are available for the financing

Note: info: information

Example questions to the vendors for *more info* to evaluate KTs/STs and example justifications to support individual NPP scores:

10.5 Question NPP2: Why are decommissioning costs not assessed? What is the regulator's requirement on this point?

10.6 Question NPP1 and NPP3: Can you explain and detail the guarantees? Export credits?

ANNEX REFERENCES

[A–1] INTERNTIONAL ATOMIC ENERGY AGENCY, Reactor Power Technology Assessment Case Study: L-Retasland, internal report, IAEA, Vienna, 2020.

[A–2] INTERNTIONAL ATOMIC ENERGY AGENCY, Reactor Power Technology Assessment Case Study: S/M-Retasland, internal report, IAEA, Vienna, 2020.

ABBREVIATIONS

AC	alternating current (frequency)
AHP	analytic hierarchy process
ALARA	as low as reasonably achievable
ALWR	advanced light water reactor
ARIS	Advanced Reactors Information System
BIS	bid invitation specification
BOP	balance of plant
CDF	core damage frequency
CUC	common user considerations
DBE	design basis earthquake
DBH	design basis hazards
DBT	design basis threat
DiD	defence in depth
DOE	Department of Energy (United States of America)
DSA	deterministic safety analysis
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
EUR	European Utility Requirements
FOAK	first of a kind
FS	feasibility study
HALEU	high assay low enriched uranium
HTGR	high temperature gas cooled reactor
I&C	instrumentation and control
IT	information technology
KE	key element
KT	key topic
LERF	large early release frequency
LWR	light water reactor
MOX	mixed oxide
NEPIO	nuclear energy programme implementing organization
NOAK	Nth of a kind
NPP	nuclear power plant
NSSS	nuclear steam supply system
O&EP	operating and emergency procedure
O&M	operation and maintenance
OBE	operating basis earthquake
OLCs	operational limits and conditions
OPEX	operational experience
PFS	pre-feasibility study
PGA	peak ground acceleration
PRIS	Power Reactor Information System
PSA	probabilistic safety assessment
PWR	pressurized water cooled reactor
QA	quality assurance
QC	quality control
RTA	reactor technology assessment
SAMG	severe accident management guideline

SAR	safety analysis report
SBD	safeguards by design
SMART	simple multi-attribute rating technique
SMRs	small and medium sized or modular reactors
SNF	spent nuclear fuel
SNM	special nuclear material
ST	subtopic
UHS	ultimate heat sink
URD	utilities requirements document
V&V	verification and validation
WCR	water cooled reactor

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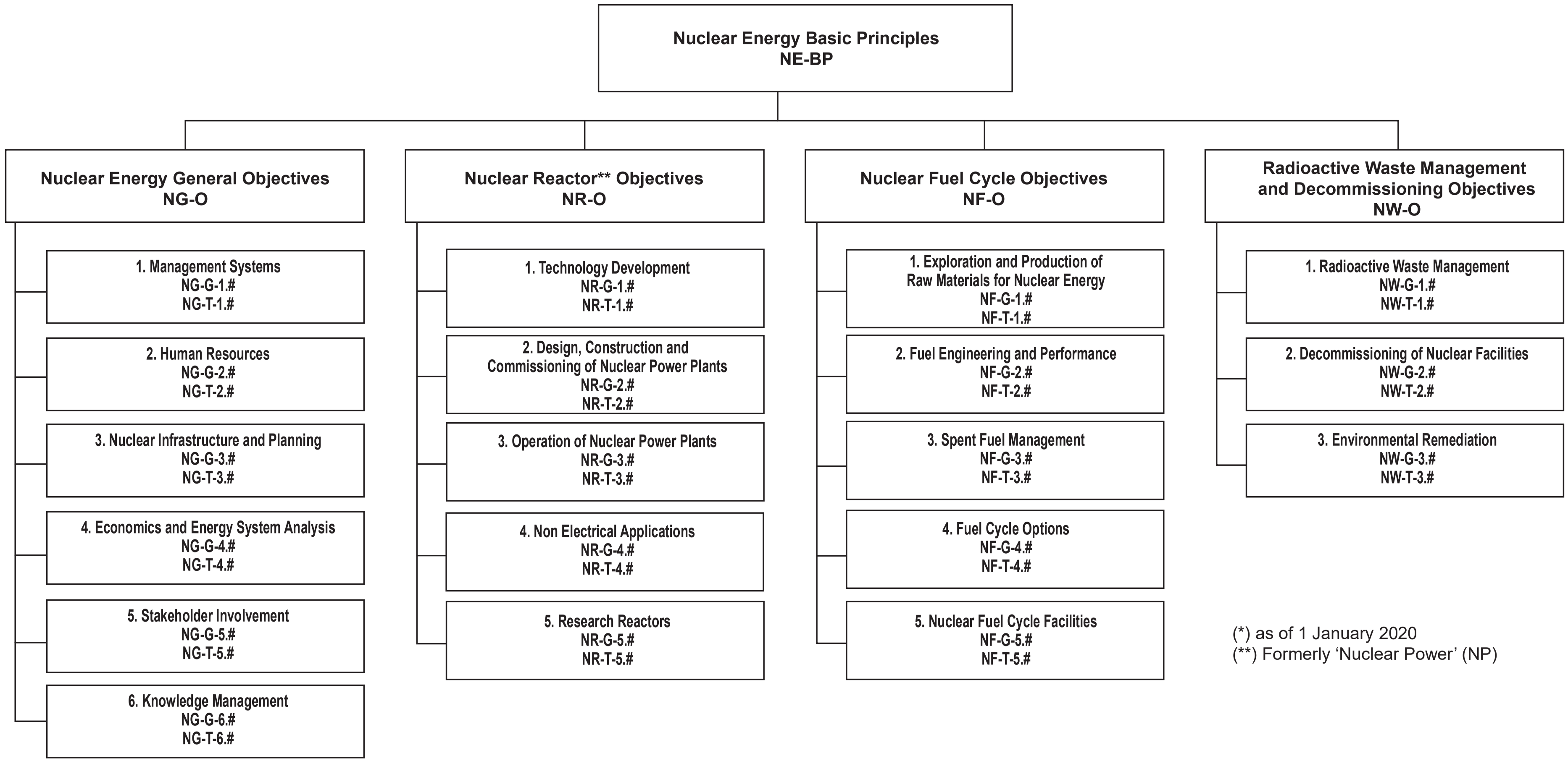

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Key

BP: Basic Principles
O: Objectives
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Examples

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NR-T-5.4: Nuclear Reactors (**NR**), Technical Report (**T**), Research Reactors (topic **5**), **#4**
NF-T-3.6: Nuclear Fuel (**NF**), Technical Report (**T**), Spent Fuel Management (topic **3**), **#6**
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