BGP-GW-GL-200  
Revision B

**High Level Risk Assessment of the 26 identified laws, regulations, standards and guides, proposed by KNPP Newbuilds for 2023 Front-End Engineering and Design (FEED) study for the AP1000**

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RECORD OF CHANGES

| Revision | Author | Description | Completed |
| --- | --- | --- | --- |
| A | Hlib Kryvonishchenko | Initial Document Issue | See PRIME |
| B | Hlib Kryvonishchenko | Inclusion of the results of the assessment APP-GW-G0R-007 Revision A, reference [41] to sections 2.11 SSG-39 Design of Instrumentation and Control Systems for Nuclear Power Plants, and 3.2.1 Risks and topics for further consideration. | See PRIME |

OPEN ITEMS

| Item | Description | Status |
| --- | --- | --- |
| N/A | No open items |  |

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LIST OF ACRONYMS AND ABBREVIATIONS

|  |  |
| --- | --- |
| AC | Alternate current |
| ADS | Automatic Depressurization System |
| ALARA | As Low As Reasonably Possible |
| ANS | American Nuclear Society |
| ANSI | American National Standards Institute |
| ASME | American Society of Mechanical Engineers |
| BGP | Bulgaria Standard for AP1000 plants |
| BDB | beyond design basis |
| BNRA | Bulgarian Nuclear Regulatory Agency |
| BWR | Boiling Water Reactor |
| CDF | Core Damage Frequency |
| CFR | Code of Federal Regulations |
| COM | Compliance |
| COM-B | Compliant with planned update for Bulgaria Units |
| COM-P | Compliant with planned update for new European AP1000 plant projects |
| CSDRS | Certified Seismic Design Response Spectra |
| CVS | Control Volume System |
| CWO | Compliance with Objective |
| DAS | Diverse Actuation System |
| DBA | Design Basis Accident |
| DC | Direct Current |
| DCD | Design Control Document |
| DiD | Defense-in-Depth |
| EAB | Exclusion Area Boundary |
| EOP | Emergency Operating Procedures |
| EP | External Party |
| EQ | Equipment Qualification |
| EU | European Union |
| FEED | Front End Engineering Development |
| FSAR | Final Safety Analysis Report |
| GDA | Generic Design Assessment |
| GSR | General Safety Requirements |
| HCLPF | High-Confidence, Low Probability of Failure |
| IAEA | International Atomic Energy Agency |
| IDS | Class 1E DC and UPS Power System |
| IRSWST | Inside-Containment Refueling Water Storage Tank |
| ISFSI | Independent Spent Fuel Storage Installations |
| KNPP | Kozloduy Nuclear Power Plant |
| KNPP-NB | KNPP Newbuilds. A public limited company registered in the Republic of Bulgaria |
| LOCA | Loss of Coolant Accident |
| LPZ | Low Population Zone |
| LRF | Large Release Frequency |
| MCR | Main Control Room |
| NAP | Not Applicable |
| NAR | Not a Requirement |
| NAS | Not Assessable |
| NFPA | National Fire Protection Association |
| NOC | Non-Compliance |
| NPP | Nuclear Power Plant |
| NRC | Nuclear Regulatory Commission |
| OBE | Operating Basis Earthquake |
| OLCs | Operational Limits and Conditions |
| OSHA | Occupational Safety and Health Administration |
| OWN | Owner |
| PGA | Peak Ground Acceleration |
| PHWR | Pressurized Heavy Water Reactor |
| PIE | Postulated Initiating Events |
| POS | Project or Site-specific Scope |
| PPE | Personal Protection Equipment |
| PRA | Probabilistic Risk Assessment |
| PSA | Probabilistic Safety Assessment |
| PSAR | Preliminary Safety Analysis Report |
| PSS | Primary Sampling System |
| PTP | Plant Transportation Package |
| PWR | Pressurized Water Reactor |
| QA | Quality Assurance |
| QMS | Quality Management System |
| RAW | Radioactive Waste |
| RMS | Radiation Monitoring System |
| SAMDA | Severe Accident Mitigation Design Alternatives |
| SAMG | Severe Accident Management Guidelines |
| SAR | Safety Analysis Report |
| SB EOP | Symptom Based Emergency Operating Procedures |
| SC | Seismic Category |
| SCR | Supplementary Control Room |
| SERAW | Bulgarian State Enterprise for “Radioactive Waste” |
| SF | Safety Fundamentals |
| SFP | Spent Fuel Pool |
| SNF | Spent Nuclear Fuel |
| SRTF | Site Specific Radwaste Treatment Facility |
| SSC | Structures, Systems, Components |
| SSE | Safe Shutdown Earthquake |
| SSG | Specific Safety Guide |
| SSR | Specific Safety Requirements |
| UFSAR | Updated Final Safety Analysis Report |
| UK | United Kingdom |
| US | United States |
| WEC | Westinghouse Electric Company |
| WENRA | Western European Nuclear Regulators’ Association |

# 

# INTRODUCTION

## Overview and purpose of document

This document provides a high level assessment of risk categorization of the 26 documents that were prioritized by KNPP New Builds to be analyzed as part of the Task 2 Described in ANNEX I “Scope of Work” to the 2023 Contract for “Front End Engineering and Design Work Agreement” between Kozloduy NPP-Newbuilds, PLC and Westinghouse Energy Systems LLC Bulgarian Branch for Works Supporting an AP1000 Nuclear Facility at Kozloduy, Bulgaria.

The Documents include the English Translations of the Bulgarian Acts and Regulations that were provided by KNPP-NB as contract Input I.02.01 containing the “Applicable regulations, guides, codes and standards that are planned to be invoked by KNPP Newbuilds, translated into English” by the Letter **L.KNP\_WEC\_230003** [27].

The purpose of this task during FEED is the identification of potential design risks and priorities for assessment work to be performed in a future project phase.

### Document Structure

To facilitate the discussion of future risk for the project, this report is structured as follows:

**Section 1:** Introductory part which covers:

1.1 The Purpose, Overview and structure of the document

1.2 Some details and description of the AP1000 design which is the one to be assessed for potential design changes or additional analyses

1.3 Definition of the Risk Categories to be Applied to each Document

1.4 Definition of the compliance assessment labels used to classify each of the analyzed requirements/articles/items in the supporting individual supporting reports [31] to [56]

**Section 2** of this Document is divided in 26 subsections (2.1 to 2.26) each one of these subsections contains the High Level Risk Assessment Summary for each of the 26 documents (Act, Regulation, IAEA Safety Requirements or Safety Guides).

Each of this subsection contains:

1. The results of the high level assessment. A summary in which a justified risk category is assigned to each document in agreement to the contract definition, as explained in section 1.3 of this document.
2. Second is a Roadmap to facilitate the location of important topics that have needed specific discussions in each one of the compliance assessment reports included in Attachment 1 to this document. These reports are references [31] to [56] that contain the assessments of each of the documents [1] to [26] to be analyzed.
3. A discussion of Non Compliances with the document if there were any.
4. A list of the potential Risk Identified for each document marked by the customer as mandatory for compliance, as identified in Annex I page I-5 of the Contract [27].

**Section 3** contains:

3.1 The List Summarizing risks classes for all the documents. See **Table 3‑1**.

3.2 Main risks and topics for further discussions identified in the compliance assessments.

**Section 4** contains the references for this report including the individual supporting reports [31] to [56] included in Attachment 1 to this document. Note, however, that it does not contain the list of references used in each of these independent compliance assessment reports.

## Standard AP1000 plant as basis for the assessments

The standard AP1000 plant design is the design documented in APP-GW-GL-700 Rev. 19, “AP1000 Plant Design Control Document” [28], herein referred to the as the Design Control Document (DCD [28]). This standard AP1000 plant design is used as the basis for the assessment as it provides one consistent reference document with sufficient documentation to demonstrate the safety approach for the AP1000 plant design, analysis, and licensing basis in the United States.

The Reference Plant design for future AP1000 units is the Vogtle AP1000 design, which have publicly available information of their FSAR available in reference [29]. There are design and licensing updates for the Vogtle Units design that have occurred since the issuance of the DCD [28]. However, the DCD [28] provides documentation that supports the safety approach of the AP1000 plant design for the Reference Plant and future AP1000 plant designs for the purpose of demonstrating compliance or assessing the potential risks for the project related with Bulgaria regulations and invoked regulations, guides, codes and standards, that could potentially be used in the project at a later stage.

During the risk assessment process performed in this document, the current Vogtle 3 and 4 AP1000 plant design and licensing basis has been reviewed within the context of the Bulgaria regulation requirements.

If there has been a significant deviation from the design, methodology, or safety analysis approach since the time of the DCD [28] documentation that would impact compliance/risks with the Bulgaria regulation, this would be identified in the assessment of that particular article of the regulation.

A probabilistic safety assessment (PSA) consists of a systematic and comprehensive evaluation of the risks. This exercise is referred to as ‘probabilistic risk assessment’ in the U.S. regulatory terminology. These two names are equivalent. The design Probabilistic Risk Assessment (PRA) for the standard AP1000 plant is documented in APP-GW-GL-022 Rev. 8 [30]. The design PRA [30] provides a basis to demonstrate the PRA approach and methodology implemented in informing the standard AP1000 plant design documented in the DCD [28]. For the Reference Plant, there is a Vogtle Unit 3 & 4 specific PRA that has been developed implementing updates to the AP1000 plant design from the standard AP1000 plant design to the Vogtle Units design and site-specific aspects of the Vogtle 3 & 4 AP1000 units, based on the design PRA. The design PRA [5] is used in this compliance assessment to provide consistent documentation to the DCD [28] and because it adequately demonstrates the PRA methodology and PRA basis for the overall AP1000 plant design. For the Bulgaria AP1000 plant project, the updated Vogtle Unit 3&4 PRA will be the basis for the future PRA as one of Vogtle Units will be the Reference Plant.

This High Level Risk Assessment contains a preliminary compliance/risk assessment performed primarily to the standard AP1000 plant design documented in APP-GW-GL-700 Rev. 19, with review of any Reference Plant (Vogtle 4) licensing basis design changes (as of the date of this document) that would impact the assessment of compliance.

A final reconcilation and compliance assessment will be required to be performed after the preliminary decision on the design and licensing basis for the Bulgaria AP1000 plant project and any new analyses that may need to be performed for the Bulgaria AP1000 to address final compliance with the Regulations.

## Regulation risk assessment categories

In agreement with the purpose of this task during FEED is the identification of potential design risks and priorities for assessment work to be performed in a future project phase.

Each of the High Level Risk Assessments has been assigned to a specific risk category.

The following risk categories are assigned:

1. Low risk: Compliance with regulation, codes, and standards is expected to be demonstrated without requiring a design change or requiring new design analyses.
2. Medium risk: Compliance with regulation, codes, and standards is expected to be demonstrated without requiring a design change but requiring new design analyses.
3. High risk: Compliance with regulation, codes, and standards is expected to require a design change and potentially new design analyses.

The category assigned to each regulation, code or standard will be justified, and any identified gaps will be discussed.

## Compliance assesment classification label

Even though the scope of this Task 2 does not include a compliance matrix assessment for each regulation, code, and standard nor a line-by-line compliance matrix at this stage, some parts are checked line by line in section.

**Table 1-1** presents special compliance classification labels that were used for the assessments where a line-by-line compliance matrix has been done.

**Table 1‑1: Compliance classification labelling**

|  |  |  |
| --- | --- | --- |
| **Classification Label** | **Compliance Assessment Label** | **Meaning** |
| COM | Compliance | The design meets the requirement as stated. |
| COM-B1 | Compliant with planned update for Bulgaria Units | Full compliance with design modification or analysis/licensing scope planned for Bulgaria AP1000 plant project. |
| COM-P2 | Compliant with planned update for new European AP1000 plant projects | Full compliance with project-specific design modification or analysis/licensing scope planned for future European AP1000 plant project |
| CWO | Compliance with Objective | The design meets the objective of the requirement, although there may be differences in approach or terminology. |
| EP | External Party | This requirement is the responsibility of an External Party i.e. Government, Regulatory Body etc. |
| NOC | Non-Compliance | The design does not meet the requirement. |
| NAS3 | Not Assessable | Not Assessable: The requirement is not currently assessable (e.g., unclear requirement, insufficient design maturity, different methodology application than standard AP1000 plant, site-specific feature). |
| NAP, N/A | Not Applicable | The requirement is not applicable to the technology. |
| NAR, N/R | Not a Requirement | This is a statement, not a requirement. |
| OWN, OR | Owner | This requirement is the responsibility of the Owner and/or Licensee. |
| POS | Project or Site-specific Scope | This requirement requires project or site-specific scope to be performed to meet the requirement.  The responsibility party for this scope will be defined in a project-specific DOR; the scope could be the responsibility of the designer, Owner, or other third party. |

Note1: Labelling specific to assessments of Bulgarian regulations, codes and standards.

Note2: Labelling specific to *“IAEA Specific Safety Guide No SSG-61 Format and content of the safety analysis report for nuclear power plants”.*

Note3: Labelling specific to assessments of Bulgarian regulations, codes and standards as well as *“IAEA Specific Safety Guide No. SSG-61 Format and content of the safety analysis report for nuclear power plants”, “IAEA Specific Safety Guide No.* SSG-63 *Design of Fuel Handling and Storage Systems For Nuclear Power Plants”, “IAEA Specific Safety Guide No, SSG-64 Protection Against Internal Hazards in the Design of Nuclear Power Plants”.*

# HIGH LEVEL RISK ASSESMENTS Summaries

This section provides summaries for each of the high level risk assessments.

Regulations, Codes and Standards marked by the customer as “obligatory to comply with” have additional “identified potential risks” subsection.

All identified articles, paragraphs and requirements classified as CWO, POS, COM-P and COM-B are highlighted for each chapter or section of the Regulation, Code or Standard where compliance matrix has been used. Each compliance class mentioned in this report is followed by the number of the item that it refers to and its high level description. For further details on each item and justification provided by Westinghouse, one should refer to the relevant compliance assessment in the Attachment 1.

## \*ACT ON THE SAFE USE OF NUCLEAR ENERGY

Following summary reflects results of the assessment conducted against reference [1] as part of the compliance assessment report [31] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [31] indicates that compliance with the requirements presented in Bulgarian “Act on the Safe Use of Nuclear Energy” [1] is expected to be fully demonstrated via compliance with the requirements “as stated”. Since no non-compliances have been identified and site-specific scope items are not expected to result in significant design changes or additional design analyses, Bulgarian “Energy Act” has been assigned to a “**Low Risk**” category.

Potential risks associated with the Bulgarian “Act on the Safe Use of Nuclear Energy” are presented in “Identified Potential Risks To Be Addressed In Bulgaria Project” subsection.

### Non Compliance

No “Non Compliances” have been identified in the assessment [31].

### Identified Potential Risks to Be Addressed In Bulgaria Project

The assessments conducted in Section 2 of the compliance assessment report [31] indicate that the following project specific items need to be considered for future assessment in the Bulgaria AP1000 project:

* **Chapter 7 (Physical Protection):** The Design Basis Threat will need to be reviewed to determine AP1000 plant compliance.
* **Chapter 9 (Application of Safeguards):** The current AP1000 plant design and general infrastructure is expected to support the addition of technical safeguards, as approved by Euratom and IAEA, to fully measure, monitor and record nuclear material subject to safeguards both via on-site inspections and remote monitoring; however, these systems are not currently provided in the standard design.

## \*REGULATION ON RADIATION PROTECTION

Following summary reflects results of the assessment conducted against reference [2] as part of the compliance assessment report [32] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [32] indicates that compliance with the requirements presented in Bulgarian “Regulation on Radiation Protection” [2] is expected to be demonstrated but will require some additional site-specific analyses. Since no non-compliances have been identified and no design changes are anticipated, Bulgarian “Regulation on Radiation Protection” has been assigned to a “**Medium Risk**” category.

Risks associated with the Bulgarian “Regulation on Radiation Protection” are presented in “Identified Potential Risks to Be Addressed In Bulgaria Project” subsection.

These risks are related to:

* Differences in Regulatory framework between Bulgaria and the United States. **These differences are analyzed more detailed in subsection 1.4 and 3.2 of the compliance assessment [32]**, and might in some specific cases require some additional analyses, but are not expected to lead to any major design changes,
* Some minor design changes might be requested for some plant and effluent monitors, however since they are considered as minor, we maintain the classification as medium risk.

The AP1000 plant is designed with administrative programs and procedures to maximize the incorporation of good engineering practices and lessons learned to accomplish ALARA objectives. This has been reviewed and verified by US-NRC Reviews of Standard Design Certification, the beginning of the Operation at Vogtle Unit 3, China Operating Plants, as well as by the Generic Design Assessment in the United Kingdom (UK).

Thus, AP1000 design should be able to accommodate Bulgarian Regulation, with some specific dose analyses that need to be performed for compliance in the early stages of the project, and some minor changes in some radiation and effluent monitors.

### Roadmap of Items Specifically Discussed in the Regulation Assessment

#### Chapter 1 (General Provisions)

**Chapter 1** of the reference document [2] is explanatory.

Setting out specific safety requirements for the activities related to the operation of nuclear power plants, research reactors, radioactive waste management facilities and spent nuclear fuel management facilities as well as the activities related to transport of radioactive substances is the responsibility of external party.

#### Chapter 2 (Radiation Protection System)

**Chapter 2** of the reference document [2] is comprised of five sections.

Section I (General principles of radiation protection) and Section III (Dose limits for apprentices and students) are predominantly responsibility of the Owner or/and External Party, while Section IV (Protection of pregnant and breastfeeding workers) entirely falls under responsibility of the Owner or/and External Party.

Requirements that fall under responsibility of Westinghouse are either met “as stated” or are expected to be met as a result of “COM-B” scope.

“Compliant with planned update for Bulgaria Units” (COM-B) items were identified in three Sections.

Section II (Dose limits for occupational and public exposure) contains COM-B items related to:

* **(Article 11 (1))** Requirement for limit on the effective dose for any occupationally exposed worker to be 20 mSv in any single year.
* **(Article 11 (2))** Limits on equivalent doses, additional to the effective dose for any occupationally exposed worker.
* **(Article 13 (2))** Limits on equivalent doses, additional to the effective dose for a member of the public.

Section III (Dose limits for apprentices and students) contains a COM-B item related to:

* **(Article 15 (2))** Limits on equivalent doses, additional to the effective dose for apprentices and students aged between 16 and 18 years who, in the course of their studies, are obliged to work with ionising radiation.

Section V (Assessment of the effective and equivalent doses from internal and external exposure) contains COM-B items related to:

* **(Article 18 (1))** Using the values, interdependencies and measurement units, as well as the radiation and tissue weighting factors in the evaluation of the effective and equivalent doses of external and internal exposure.
* **(Article 18 (2))** Taking into accountthe physico-chemical or other characteristics of the sources of ionising radiation in the assessment of the doses for a given exposure situation or exposed individual.
* **(Article 18 (6))** Secondary (derived) limits for radiation control purposes, protection planning and dose estimation for occupationally exposed workers and members of the public in planned exposure situations.

*Refer to* ***Subsection 2.3*** *of the compliance assessment report [32] for more details.*

#### Chapter 6 (Radiation Protection in Occupational Exposure)

**Chapter 6** of the reference document [2] falls under scope of responsibility of the Owner and/or External party.

Section III (Controlled areas) and Section IV (Surveillance areas) are entirely responsibility of the Owner.

Section XII (Radiation protection of outside workers) and Section XIII (Control of exposure to radon in workplaces) are predominantly responsibility of the Owner.

Section I (Operational radiation protection), Section V (Categorization of exposed workers), Section VI (Individual monitoring in occupational exposure), Section VII (Radiation monitoring programs), Section VIII (Recording and reporting radiation monitoring and individual dosimetric monitoring results), Section IX (Medical surveillance of occupationally exposed workers), Section X (Specially authorized exposures) as well as Section XI (Emergency occupational exposure) are responsibility of the Owner and/or External Party.

Section II (Arrangements in workplaces) is predominantly not applicable to technology. However AP1000 radiation zones are explained.

One COM-B item was identified in Section III (Controlled Areas) and is related to:

* **(Article 56 (2))** Radiological parameters that, depending on the case, shall be measured, assessed and included to the radiological monitoring.

*Refer to* ***Subsection 2.4*** *of the compliance assessment report [32] for more details.*

#### Chapter 7 (Radiation Protection of Members of the Public In Planned Exposure Situations)

Requirements in **Chapter 7** of the reference document [2] are responsibility of the Owner and/or External party.

Several COM-B items were identified in Section I and are related to:

* **(Article 98 (3))** Determining representative individuals on the basis of performed studies and taking into account the actual exposure pathways leading to internal and external exposure (For the purposes of a realistic assessment of the doses to members of the public and for comparison with dose constraints).
* **(Article 100 (1))** obligation for the undertakings authorised for activities involving discharges of effluents into the environment to appropriately monitor and/or evaluate the quantity and activity of the radioactive airborne or liquid discharges into the environment in normal operation of the respective nuclear facilities and facilities with sources of ionising radiation.

*Refer to* ***Subsection 2.5*** *of the compliance assessment report [32] for more details.*

#### Chapter 8 (Radiation Protection of The Public in Emergency Exposure Situations)

Requirements in **Chapter 8** of the reference document [2] are responsibility of the Owner and/or External party.

One COM-B items was identified in **Chapter 8** and is related to:

* **(Article 103 (1))** Setting the reference levels for exposure of members of the public, expressed in effective doses, in the range of 20 to 100 mSv (acute or annual) for emergency exposure situations.

*Refer to* ***Subsection 2.6*** *of the compliance assessment report [32] for more details.*

#### Chapter 11 (Requirements for The Design and Operation Of Nuclear Facilities And Facilities With Sources Of Ionizing Radiation)

Majority of the requirements in **Chapter 11** of the reference document [2] are responsibility of the Owner and/or External Party.

Requirements that fall under responsibility of Westinghouse are either met “as stated” or via compliance with their objectives. One requirement was assigned to a “COM-B” scope and one is not assessable.

COM-B item was identified in **Chapter 11** andis related to:

* **(Article 135 (1))** Subjects that shall be provided for the purposes of radiation protection, in the process of designing of nuclear facilities and facilities with sources of ionising radiation, as well as in the choice of technologies, structures, systems and components.

Several “Compliant with objective” (CWO) items were identified in **Chapter 11** and are related to:

* **(Article 140 (1))** Determining the requirements to the design of the systems for ventilation, cleaning of dust, aerosols and gases, sewage and water supply by the design and construction norms and rules applicable for industrial undertakings.
* **(Article 140 (2))** The specific requirements for the systems for ventilation, cleaning of dust, aerosols and gases, sewage and water supply, related to the radiation protection in nuclear facilities and facilities with sources of ionising radiation.

*Refer to* ***Subsection 2.7*** *of the compliance assessment report [32] for more details.*

### Non Compliance

No “Non Compliances” have been identified in the assessment [32].

### Identified Potential Risks To Be Addressed In Bulgaria Project

The assessment conducted in **Section 2** of the compliance assessment report [32] indicates that the following articles and sub-articles need to be considered in the risk assessment:

Articles related to different Occupational dose as explained in subsection **1.4** of compliance assessment report [32]:

* **(Article 11)** Occupational Exposure for Workers,
* **(Article 15)** Dose Limits for apprentices and students (16 to 18 years). This is conservatively considered a COM-B,
* **(Article 135 (1))** 1. Compliance with dose limits and dose constraints for occupationally exposed workers and members of the public. As explained in section 1.4 of the compliance assessment report [32], current AP1000 design is capable of fulfilling this requirement with appropriate radiation programs in place, even though these limits differ to the ones used in 10 CFR 20. Thus conservatively, this is considered as COM-B since at some point there could be potentially the need to make additional analyses.

Other articles based in differences between US-NRC radiation protection regulatory framework and Bulgarian Regulation for Radiation Protection:

* **(Article 13)** Dose Limitation for Member for the Public. This is conservatively considered as a COM-B,
* **(Article 18 (1))** Evaluation of the effective and equivalent doses of external and internal exposure, the values, interdependencies, and measurement units, as well as the radiation and tissue weighting factors,
* **(Article 18 (2))** Assessment of the doses for a given exposure situation or exposed individual, the physico-chemical or other characteristics of the sources of ionizing radiation,
* **(Article 18 (6))** Secondary (derived) limits for radiation control purposes, protection planning and dose estimation for occupationally exposed workers and members of the public in planned exposure situations, set out in Annex No. 2 of the Bulgarian “Regulation on Radiation Protection” reference [2].
* **(Article 103 (1))** For emergency exposure situations might require additional updated accidental analyses to support Emergency Planning.

Articles related to radiation monitoring in the plant and premises:

* **(Article 56 (2))** The radiological monitoring shall include measurement and assessment of the radiological parameters relevant for the workrooms. This could require changes on existing monitors setpoint and more unlikely the addition of some extra monitors.

Articles related to effluent monitoring and normal operation doses to the public constraint compliance:

* **(Article 98 (3))** Realistic assessment of the doses to members of the public and for comparison with dose constraints due to releases. A specific calculation will be needed for releases in normal operation, even though some previous feasibility studies of compliance were performed previously,
* **(Article 100)** Appropriately monitor and/or evaluate the quantity and activity of the radioactive airborne or liquid discharges into the environment in normal operation. AP1000 Effluent monitoring for Bulgaria shall agree with local requirements, thus most likely it should conform with Recommendation 2004/2/Euratom which provides guidance to EU countries on the reporting of discharges of radioactive nuclides.

## \*REGULATION ON ENSURING THE SAFETY OF NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [3] as part of the compliance assessment report [33] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [33] indicates that compliance with the requirements presented in Bulgarian “Regulation on Radiation Protection” [3] is expected to be demonstrated but will require some additional site-specific analyses. Since no non-compliances have been identified and no design changes are anticipated, Bulgarian “Regulation on Radiation Protection” has been assigned to a “**Medium Risk**” category.

Risks associated with the Bulgarian “Regulation on Radiation Protection” are presented in “Identified Potential Risks To Be Addressed In Bulgaria Project” subsection.

### Roadmap of Items Specifically Discussed in the Regulation Assessment

#### Chapter 1 (General provisions)

All design related requirements presented in **Chapter 1** of the reference document [3] are either met by Westinghouse “as stated” or will require some additional “COM-B” scope.

Part of the requirements are responsibility of the Owner.

“Compliant with planned update for Bulgaria Units” (COM-B) items were identified in **Chapter 1** and are related to:

* **(Article 2 (1) 1)** Provision that requires the radiation impact of an NPP in all operational states to be maintained below the prescribed dose limits for internal and external exposure of the personnel and the public, and to be kept as low as reasonably achievable.
* **(Article 3 (1) 2)** Ensuring the use of the defence-in-depth concept by **(2)** selection of a suitable site and combining a conservative design with appropriate engineering solutions to provide diversity, redundancy, and safety margins, mainly through the use of:  
  a) design, technology and materials of high quality and reliability,  
  b) systems and design parameters that control and limit the reactor installation operation,  
  c) appropriate combination of inherent and engineered safety features.
* **(Article 4 (3) 2-4)** Requirement stating that the safety objectives for accidents without nuclear fuel melting are to prevent fuel damage through engineering and administrative provisions while demonstrating that:

**(2)** such accidents will not cause any radiological impact outside of the NPP site or will not require iodine prophylaxis, sheltering or evacuation as protection measures for the public;

**(3)** release of radioactive substances from all sources of ionizing radiation has been minimized to the extent practicable.

**(4)** at the stages of site selection and design, measures have been taken to decrease the impact of external events, hazards or malicious acts.

*Refer to* ***Subsection 4.2*** *of the compliance assessment report [33] for more details.*

#### Chapter 3 (Site characteristics)

Site is already approved, however further considerations may be needed to confirm that site-specific characteristics are covered by the AP1000 design. Owner will be responsible for providing the required input data for such considerations.

Part of the requirements fall entirely under the Owner’s scope of responsibility.

As a result, most of the requirements in **Chapter 3** of the reference document [3] were identified to be not applicable, requiring additional “COM-B” scope or being responsibility of the Owner.

COM-B items were identified in two sections.

“Section I: General Requirements” contains COM-B items related to:

* **(Article 28)** Requirement to assess and document the characteristics of potential NPP sites and of the selected site as an integral part of the overall NPP safety analysis.
* **(Article 30 (2))** Requirement to take account of the impact of the existing nuclear facilities when the site under consideration for a new NPP is in close proximity to the site of an existing NPP.

“Section II: Studies of Natural and Human Induced Factors for Site Selection” contains COM-B items related to:

* **(Article 33 (5))** Requirement stating thatthe impact parameters on an NPP and the respective probabilities shall be determined for events induced by:

1. explosions and fires, releases of explosive, inflammable, toxic and corrosive gases and substances from industrial facilities, ground and water conveyance facilities;

2. a modern passenger aircraft crash;

3. floods, including those related to reaching the waterfront as a result of breaking of dams located upstream of the NPP site;

4. accidents of a water vessel along water routes and in harbour zones occurring together with explosions and fires, releases of dangerous chemicals, provided the NPP is situated within their range of impact;

5. electromagnetic emissions (fields);

6. external fires (forest areas, peateries, flammable liquids);

7. deformations and other factors arising on developing underground resource deposits, carrying out excavation works, including tunnel construction, mines and quarries exploitation and their emergency destruction;

8. water level fluctuations of the NPP water supply source.

*Refer to* ***Subsection 4.3*** *of the compliance assessment report [33] for more details.*

#### Chapter 4 (Defence in depth and design basis)

Majority of the requirements presented in **Chapter 4** of the reference document [3] are met by Westinghouse “as stated”. Some requirements are met via compliance with their objectives while several will require an additional “COM-B” scope.

Operation, maintenance and testing related topics in this Article are also requirements for the Owner.

“Compliant with objective” (CWO) items were identified in two sections.

“Section I: Defence in Depth Implementation in the Design” contains a CWO item related to:

* **(Article 41 (4))** Requirement for the systems and means for prevention of accidents with nuclear fuel melting to be independent of systems and means specially designed to perform safety functions in case of a postulated severe accident to such an extent as not to impede the implementation of these functions.

“Section II: Design Basis” contains CWO items related to:

* **(Article 49 (2))** Requirement stating that the selection of multiple failure events shall consider as follows:

1. a postulated common cause failure or inefficiency of all trains of a safety system which performs a required safety function in the conditions of an anticipated operational occurrence or a postulated initiating event;

2. a postulated common cause failure of a safety system or a safety important system performing a main safety function in normal operational mode.

* **(Article 54 (2))** Requirement stating that the following shall be specified for each safety class:

1. the appropriate standards and rules for design, manufacturing, installation and inspection;

2. the degree of redundancy, the need for emergency power supply, and qualification for operation under specific adverse environmental conditions;

3. the state of operability or inoperability of SSCs that is considered in the deterministic safety analysis;

4. the applicable quality requirements.

* **(Article 54 (3))** Requirement stating that the NPP design shall preclude the interference of individual SSCs important to safety, and shall ensure that a failure of an SSC of one safety class shall not cause a failure of an SSC of a higher safety class. The auxiliary systems, supporting SSCs important to safety, shall be assigned to the same safety class.
* **(Article 54 (4))** Requirement stating that the design shall consider an appropriate isolating device which shall be classified in a higher safety class for the cases of connecting SSCs of different safety classes, or of safety classified SSCs to SSCs that are not safety related.
* **(Article 57 (1))** Requirement stating that the design of SSCs and other technical features intended to perform safety functions during severe accidents shall implement the following principle requirements:

1. SSCs performing safety functions at other defence-in-depth levels shall be independent to the extent practicable;

2. safety classification, seismic qualification and environmental qualification for the duration of the accident throughout which they are required to remain functional;

COM-B items were identified in “Section II: Design Basis” and are related to:

* **(Article 43)** Requirement stating that in all operating states of the nuclear facilities on the NPP site, the annual individual effective dose resulting from internal and external exposure of the public caused by the impact of all nuclear facilities on-site shall be maintained as low as possible and shall not exceed 0,15 mSv.
* **(Article 44 (2))** Requirement for the safety assessment under para. 1 to confirm that the mean value of the nuclear fuel melting frequency is less than 10-5 a year for a nuclear power unit based on the consideration of all operating states and all types of initiating events and hazards.
* **(Article 49 (6))** Requirement for the design basis to cover possible combinations of single events, including internal and external hazards that can result in anticipated operational occurrences and accidents without fuel melting.

*Refer to* ***Subsection 4.4*** *of the compliance assessment report [33] for more details.*

#### Chapter 5 (Safety assessments)

Most of the requirements presented in **Chapter 5** of the reference document [3] are met by Westinghouse “as stated”. One requirement was met by demonstrating compliance with its objective.

Numerous requirements were identified as “not assessable” or “COM-B” mainly because safety assessment may require further consideration of site-specific phenomena’s, characteristics, and hazards to ensure nuclear safety and further considerations may be needed to consider site-specific data in the PSA analysis.

Part of the requirements falls under responsibility of the Owner.

One CWO item was identified in “Section IV: Analysis of External Events and Hazards” and is related to:

* **(Article 84 (2))** Requirement stating that extreme events and phenomena, which are more severe than the design-basis ones, but cannot be practically eliminated, shall be identified and analysed with a realistic approach in order to practically identify the possible improvements related to them. The assessment process shall take into account the following aspects:

1. determination of the event impact parameters in case of which the main safety functions performance cannot be ensured;

2. demonstration of a sufficient margin until cliff-edge effects.

3. identification of the most robust means of ensuring the main safety functions;

4. the possibility that the event may cause multiple failures in the safety systems and/or their supporting systems and may simultaneously threaten multiple power units on the same site, site infrastructure, regional infrastructure and external supplies;

5. provision of sufficient resources for multi-unit NPP sites, taking into account the use of plant common equipment or services;

6. conducting field checks and walkdowns.

COM-B items were identified in two sections.

“Section III: Probabilistic Safety Analysis” contains COM-B items related to:

* **(Article 79 (1))** Requirement stating that for the application of an integrated approach to the NPP safety assessment, a Probabilistic Safety Analysis (PSA) shall be performed that systematically identifies all factors which have a significant contribution to safety and the radiation risk to the population and the environment. The PSA shall be conducted at the following levels:  
  1. Probabilistic Safety Analysis Level 1 – it identifies the accident initiating events, and the accident sequences, and assesses the nuclear fuel damage frequency;  
  2. Probabilistic Safety Analysis Level 2 – it identifies possible radioactive substances release pathways into the environment and assesses the large radioactive releases frequency;  
  3. Probabilistic Safety Analysis Level 3 – it assesses the risk to human health and other social risks such as soil, water and food contamination by radioactive substances, and it is implemented by a decision of the Chairperson of the Nuclear Regulatory Agency
* **(Article 79 (2))** Requirement for implementation of probabilistic safety analysis in order to achieve the following objectives:  
  1. perform a systematic analysis of the compliance with the main safety objectives and criteria, assess the frequency of occurrence of severe fuel damage and large radioactive releases into the environment, and determine the risk to the population;  
  2. prove, where possible, a sufficient margin until cliff-edge effects occur.
* **(Article 79 (3))** Requirement for the scope of PSA to include:  
  1. significant sources of radioactivity (nuclear fuel in the reactor core and the spent fuel pool) and all operational states of the power unit (including operation at full power, low power and shutdown state);  
  2. all significant initiating events, internal hazards (such as internal fires and floods) and external events and hazards (such as seismic impacts and extreme weather conditions) identified on the basis of appropriate selection criteria;  
  3. all functional dependencies resulting from spatial distribution and other possible causes for common cause failures;  
  4. realistically modelled behaviour of the power unit, taking into account the actions of the operating personnel in accordance with the operating and emergency procedures and justified time for the performance of the functions of the systems;  
  5. human error analysis, taking into account the factors which can influence the performance of the personnel in all operational states and accident conditions.  
  6. sensitivity analysis of results and uncertainty assessment.
* **(Article 79 (4))** Requirement for the probabilistic safety analysis to be performed using unit-specific data and in accordance with a contemporary proven methodology. Requirement for the data, methodology and results of the analysis to be documented in a traceable way and kept up-to-date in accordance with the operating organisation's management system.
* **(Article 80 (5))** Requirement for the external events analysis to take into account the impact of the external hazard on the reliability of the buildings and civil structures, the robustness of the systems and components and the possibilities for human action under such conditions.

“Section IV: Analysis of External Events and Hazards” contains COM-B items related to:

* **(Article 81)** Requirement stating that to assess the effectiveness and adequacy of the NPP protection against external events, the design shall consider and assess all sources of hazard that may affect safety originating from:  
  1. natural phenomena, processes and factors characteristic of the site and the surrounding area;  
  2. hazards induced by human activity.
* **(Article 83 (4))** Requirement stating that based on the assessment under para. 3 of the reference document [3], the events of natural origin considered in the design shall be grouped for analysis in the following categories:  
  1. design-basis events that include single events of natural origin and combinations of causal or unrelated phenomena and processes whose frequency of occurrence is at least 10-4 per year. Where it is not possible to determine the frequency of occurrence with an acceptable confidence level, the design-basis event shall be selected and justified in such a way as to ensure an equivalent level of safety;  
  2. extreme events which are identified, assessed and analysed in order to define the margins to cliff-edge effects.
* **(Article 83 (6))** Requirement stating that based on the deterministic analysis of design-basis events, a robust concept of protection shall be developed that shall provide for conservative implementation of the main safety functions for all direct and possible indirect effects of the design-basis event. The concept of protection shall:  
  1. use moderate conservatism to ensure the design safety margin;  
  2. be based primarily on passive measures, where applicable;  
  3. ensure that accident management measures remain effective during and after the design-basis event;  
  4. take into account the evolution of the event over time and the possibility of forecasting;  
  5. ensure that procedures and means for checking the state of the NPP are in place along with alarms and annunciation during and after the design-basis event which shall be enacted upon the occurrence of predefined limit parameters;  
  6. consider the fact that the event may cause multiple failures in the safety systems and/or their supporting systems and may simultaneously threaten multiple power units at the same site, site infrastructure, regional infrastructure and external supplies;  
  7. ensure that sufficient resources are available on multi-unit NPP sites, taking into account the use of plant common equipment (including mobile) and maintenance;  
  8. not adversely affect protection against design-basis events of other origin.
* **(Article 83 (7))** Requirement for the structures, systems and components identified as part of the concept of protection against design-basis events to be assigned to a safety class and qualified for the conditions and impacts of the relevant natural phenomena and hazards. To qualify for seismic impacts, SSCs are subdivided into seismic categories according to their functions for ensuring safety during and after an earthquake. Requirement stating that in terms of Safe Shut-down Earthquake (SSE), the minimum acceleration for seismic assurance shall be 1 m/s2 at the natural terrain elevation, while the response spectrum shall be at least equal to the corresponding response spectrum for conventional buildings construction.
* **(Article 85 (1))** Sources of hazard to be taken into account while carrying out determination of the external events of technogenic origin, characteristic of the NPP site and the siting area.
* **(Article 86 (1))** Requirement stating that in pursuance of the safety objective under Article 4, para. 3of the reference document [3], the NPP design shall take into account the consequences of a modern passenger airplane crash (as a technogenic event in the category of extreme events) and that he analysis shall demonstrate the assurance of the main safety functions that render and maintain the NPP in a safe state.

*Refer to* ***Subsection 4.5*** *of the compliance assessment report [33] for more details.*

#### Chapter 6 (Requirements for design of NPP and plant systems)

Large majority of the requirements presented in **Chapter 6** of the reference document [3] are met by Westinghouse “as stated”. Several requirements were met via compliance with their objectives and one was identified as “COM-B”.

Part of the requirements in “ Section I: General Requirements” is responsibility of the Owner.

“Section X: District Heating System” of **Chapter 6** is not applicable to the AP1000 Standard Design.

CWO items were identified in three sections.

“Section IV: Instrumentation and Control Systems” contains CWO items related to:

* **(Article 112 (3))** Requirement stating that instrumentation and control systems for the hardware and software intended for multiple failure protection and for containment protection in accidents with fuel melting, shall be separated and independent from the other instrumentation and control systems as far as practicable. Active system components shall be redundant.
* **(Article 114 (3))** Requirement for the SCR to be designed to protect the personnel in all conditions resulting from internal and external events and in accident conditions.

“Section VII: Structures and Systems Performing Confinement Safety Function” contains a CWO item related to:

* **(Article 130 (1))** Requirement stating that to ensure reliable isolation of the containment during accidents, each line that penetrates the containment (whether part of the reactor coolant pressure boundary or directly connected to the containment atmosphere) shall be reliably isolated by at least two isolation valves having independent automatic control, arranged in a series and located as close as practicable to the containment structure outside and inside.

“Section VIII: Supporting Safety Systems” contains a CWO item related to:

* **(Article 137 (4))** Requirement stating that the fulfilment of supporting functions shall have priority over the supporting systems own protections, if this will not aggravate the safety consequences. The design shall specify the uninterruptible own protections of the components of the supporting safety systems.

One COM-B item was identified “Section VI: System for Core Cooling and Heat Transfer to an Ultimate Heat Sink” and is related to:

* **(Article 125 (3))** Requirement for the choice of main and alternative ultimate heat sink and the design of heat transfer systems to be made with account taken of the site-specific natural phenomena and man-induced events with the purpose of ensuring the performance of the function under conditions of extreme external events.

*Refer to* ***Subsection 4.6*** *of the compliance assessment report [33] for more details.*

#### Chapter 7 (Construction and commissioning)

Requirements in “Section I: General Requirements” of **Chapter 7** of the reference document [3] are met by Westinghouse “as stated”.

Part of the requirements in “ Section II: Commissioning Program” belong to “COM-B” scope as detailed commissioning program, commissioning, written procedures, test program and commissioning activities will be specified more in detail during construction phase.

Compliance with objective (CWO) was assigned to one of the Articles as the minimum conditions for initial fuel loading are specified in DCD 14.2.7.1 and will be specified more in detail during construction phase.

Part of the requirements in both Sections of **Chapter 7** are addressed to the Owner.

CWO items were identified in “Section II: Commissioning Program” and are related to:

* **(Article 173 (1))** Requirement stating that before the initial core loading with nuclear fuel: SSCs important to safety and required at this stage shall be tested and their availability shall be confirmed; tests to determine the characteristics of the reactor coolant pressure boundary shall be carried out; biological shielding effectiveness shall be tested; radiation monitoring shall be carried out at the premises, site, precautionary action zone and surveillance zone.
* **(Article 173 (2))** Requirement stating that before reaching initial criticality of the reactor installation, functional tests of SSCs important to safety shall be carried out to confirm the fulfilment of the design functions and the compliance with the design characteristics.

COM-B items were identified in “Section II: Commissioning Program” and are related to:

* **(Article 168 (2))** Requirement for the tests conducted under the commissioning programme not to lead to operational states and accident conditions that have not been analysed in the interim safety analysis report.
* **(Article 169 (1))** Requirement for the NPP commissioning to be performed in sequential stages, for which separate programmes are required to be developed. Requirement for the implementation of each stage to be preceded by an evaluation of the results from the previous stage and a confirmation that the objectives set and design requirements have been met.
* **(Article 169 (2))** Requirement for the programme for each stage to describe:  
  1. The sequence, timing and logical connections between the activities at the stage;  
  2. The initial and final status at the respective stage;  
  3. The organisation for implementation and the required personnel;  
  4. The preconditions for implementation of the tests;  
  5. The requirements on the technological preparation and provision of power sources and operating fluids;  
  6. The criteria for acceptance and an assessment of their fulfilment;  
  7. The conditions for transition to the next stage.
* **(Article 169 (3))** Requirement for the programmes for each NPP commissioning stage to contain a time schedule and a list of procedures to be followed during the tests.
* **(Article 170 (1))** Requirement stating that the tests shall be carried out in accordance with written procedures that shall have as a minimum requirements on:

1. Introduction and withdrawal of temporary modifications necessary to conduct the testing;

2. Verification that the prerequisites and the preconditions to conduct the tests are fulfilled;

3. Measurement equipment used and its calibration;

4. Limits and conditions to conduct the testing;

5. Test results recording means;

6. Acceptance criteria for the results and their allowable ranges;

7. Clear and unambiguous instructions on the test performance;

8. Clear rules to follow when the acceptance criteria are not met;

9. Plant recovery to normal state following the test performance.

* **(Article 170 (2))** Requirement for the developing, approving, modification, distribution and storage of the test procedures and the reporting documents containing the results to be in compliance with the management system.
* **(Article 171)** Requirement for the test results approval to be arranged in a way that the following objectives are met:  
  1. The NPP behaviour has been compared with the design requirements;  
  2. Sufficient data on the revaluation of the design bases are ensured, in case the unit behaviour deviates from the expectations;  
  3. Demonstrate that the NPP as tested allows to proceed with the next stage of commissioning or with the next test.
* **(Article 172 (1))** Requirement for the commissioning activities to be carried out in compliance with the commissioning programme, testing procedures, operation, maintenance, surveillance and inspections.
* **(Article 172 (2))** Requirement for the applicability and quality of the operating procedures to be confirmed (validation and verification) during the commissioning process.
* **(Article 173 (3))** Requirement stating that The transition from one power level to another shall be performed after successful neutron physics test (experiments) of the reactor installation and completion of all construction and assembly works.
* **(Article 173 (4))** Requirement to perform trial-testing operation as a commissioning stage for evolutionary NPPs.
* **(Article 174)** Requirement for a nuclear power plant unit, which is in a process of commissioning, to be physically isolated from other units that are in operation or under construction at the same site.

*Refer to* ***Subsection 4.7*** *of the compliance assessment report [33] for more details.*

#### Chapter 8 (Operations)

Majority of the requirements presented in **Chapter 8** of the reference document [3] are responsibility of the Owner or will be responsibility of the Owner during operation.

COM-B scope items were identified in **Chapter 8** mainly due to the fact that operating procedures and instructions, emergency procedures, water chemistry programme, necessary maintenance, test, surveillance, and inspection programmes, SAMGs and EOPs are prepared during the construction phase. In addition, one requirements was assigned to COM-B as AP1000 effluent monitoring for Bulgaria shall be in agreement with local requirements, thus most likely it should conform with Recommendation 2004/2/Euratom which provides guidance to EU countries on the reporting of discharges of radioactive nuclides.

The rest of the requirements are met by Westinghouse “as stated” and one requirement is met via compliance with its objective.

One CWO item was identified in “ Section II: Conduct of Operations” and is related to:

* **(Article 211 (1))** Requirement stating that habitability and good working conditions (lighting, noise, radiation level, temperature, means of communication) shall be maintained in the MCR and SCR during all operational states and accident conditions.

COM-B items were identified in four sections.

“Section I: Operational Safety Management” contains COM-B items related to:

* **(Article 188 (2))** Requirement for the level of detail of procedures and instructions to be based on their intended purpose. Requirement for the guidance to be clear and concise, verified, and validated.
* **(Article 188 (3))** Requirement stating that procedures, instructions, and aids shall be clearly identified, discernible in respect of intended function, and readily accessible at the MCR and other control rooms, if necessary.
* **(Article 189 (1))** Requirement for the operating procedures and instructions for normal operation to be prepared based on the design and engineering documentation, operating limits and conditions, and results of plant commissioning.
* **(Article 189 (2))** Requirement stating that Operators’ emergency response actions for all the operational states shall be stipulated in Emergency Procedures and Severe Accident Management Guidelines (SAMGs).
* **(Article 190 (1))** Requirement for the Emergency Procedures to cover the design basis accidents and scenarios where a significant fuel damage at the core or at the spent fuel pool can be prevented. Requirement stating that the Emergency Procedures shall be symptom-based (SB EOPs) and compatible with the Alarm Instructions and SAMGs.
* **(Article 190 (2))** Requirement stating that the Design Basis Accident Emergency Procedures shall prescribe how to bring the plant to a stable safe condition, while the Emergency Procedures for the scenarios where a significant nuclear fuel damage can be prevented shall prescribe how to restore or compensate for lost safety functions as well as how to prevent nuclear fuel damage at the core or at the spent fuel pool (SFP).
* **(Article 190 (3))** Requirement for the set of SB EOPs to include:  
  1. State diagnostic procedures;  
  2. Procedures for optimal recovery in the event of transients and design basis accidents;  
  3. Condition monitoring procedures and procedures for restoration of safety functions, such as subcriticality, core cooling, residual heat removal, coolant inventory, integrity of the reactor coolant pressure boundary and containment integrity;  
  4. Procedures for transition to severe accident management.
* **(Article 191 (1))** Requirement stating that the Severe Accident Management Guidelines shall result in mitigation of consequences of severe accidents when the personnel actions, including the actions prescribed in the SB EOPs, have not been successful in preventing core damage or fuel damage at the SFP.
* **(Article 191 (2))** Requirement stating that the Severe Accident Management Guidelines and SB EOPs shall:  
  1. Provide for the management of accidents impacting both the reactor and spent fuel pool and consider the possible interaction between the reactor and SFP;  
  2. Consider the capabilities for one of one powerunit to support another power unit on the NPP site without compromising its own safety;  
  3. Ensure their implementation, even when all the on-site nuclear facilities are in accident conditions, and take into account the dependencies among the systems and common resources;  
  4. Consider the expected conditions on the NPP site, including the radiological situation resulting from the accidents which they are intended for as well as the initiating event or external hazard that may have caused.
* **(Article 192 (1))** Requirement stating that the Severe Accident Management Guidelines shall be based on strategies for management of the scenarios resulting from the analysis of the weaknesses and capabilities of the nuclear power unit in the event of a severe accident, and the possible measures for management including for containment protection. The SAMGs shall consider as a priority the operation of the equipment and instruments qualified for the relevant conditions
* **(Article 192 (2))** Requirement stating that for the preparation of SAMGs and SB EOPs, unit-specific data shall be used. The effectiveness of the operators’ actions shall be analytically validated using verified computer codes and unit-specific calculation models. The results of the analysis shall be documented and used as a technical basis for the procedures.
* **(Article 203 (2))** Requirement for the operating procedures concerning reactor start-up, power operation, shutdown, and core refuelling to include safeguards and controls necessary to ensure nuclear fuel integrity and compliance with the operating limits and conditions throughout the entire period of nuclear fuel operation.
* **(Article 203 (3))** Requirement stating that the procedures governing the operations involving nuclear fuel and core components shall ensure controlled handling of fresh and irradiated nuclear fuel, adequate storage, transfer or transport preparation. Requirement stating that a person having the required experience, knowledge, and qualification shall be assigned in charge of the supervision and conduct of the operations involving nuclear fuel.
* **Article 206 (2))** Requirement for EOPs preparation on Responsibilities and actions of the personnel in case of fire shall be stipulated in a fire fighting strategy and emergency procedures that are to be learnt in the course of emergency exercises. This strategy shall cover each area where an internal fire may affect SSCs important to safety as well as the radioactive material protection.

“Section II: Conduct of Operations” contains COM-B items related to:

* **(Article 213 (1))** Requirement stating that to maintain water chemistry within the established margins, limit the ingress of chemical contaminants, and limit the radiological factors during the commissioning and operation of the plant, a programme for water chemistry and radiochemistry control shall be implemented.
* **(Article 213 (2))** Requirement for this programme for water chemistry and radiochemistry control to cover monitoring and data processing systems which, along with the laboratory analyses, shall provide for the measurement and recording of chemical and radiochemical parameters.

“Section III: Maintenance, Tests, Surveillance, and Inspections. Ageing Management” contains COM-B items related to:

* **(Article 217 (1))** Requirement stating that during commissioning and operation, maintenance, test, surveillance, and inspection programmes shall be prepared and implemented to ensure compliance of the operability, reliability, and functionality of SSCs important to safety with the design criteria throughout the entire period of plant operation. Those programmes shall consider the operating limits and conditions and shall be revised in order to reflect operating experience.
* **(Article 217 (2))** Requirement for the programmes for predictive, preventive, and corrective maintenance to cover activities for control of degradation processes, prevention of failures, and restoration of the operability and reliability of SSCs important to safety.
* **(Article 218 (3))** Requirement for the in-service inspection to be performed at intervals determined on the basis of detection of each degradation of the most loaded component before it results in a failure.
* **(Article 226 (1))** Requirement stating that surveillance measures to verify containment integrity shall cover:

1. Containment leak rate tests;

2. Individual tests of leak-tight penetrations and sealing devices, such as airlocks, doors, and isolation valves which are part of the containment, to determine their leak-tightness and, if necessary, their operability;

3. Integrity inspections of containment metal liner and prestressing tendons.

* **(Article 227 (1))** Requirement stating that the instruments and equipment intended for tests and studies shall be qualified and metrologically checked before use. Record keeping and validity of metrological assurance shall comply with the management system.
* **(Article 227 (2))** Requirement for the methods and procedures, instruments and personnel to be used for the reactor coolant system in-service inspection to be qualified.

“Section IV: Radiation Protection During Operations” contains COM-B items related to:

* **(Article 234 (2))** Requirement stating that the limits on liquid and gaseous radioactive discharges shall be monitored at the sources of those discharges and verified through environmental measurements. Three types of measurements can be used for source monitoring:  
  1. Online discharge monitoring;  
  2. Continuous sampling and laboratory measurements of sample activity;  
  3. Periodical sampling and laboratory measurements of sample activity.

*Refer to* ***Subsection 4.8*** *of the compliance assessment report [33] for more details.*

### Non Compliance

No “Non Compliances” have been identified in the assessment [33].

### Identified Potential Risks To Be Addressed In Bulgaria Project

**Articles 41 (4), 57 (1) and 112 (3)** belong to CWO category and introduce risks related to **defense-in-depth and SSCs independency.**

There are SSCs which are used both during transients and during accidents with fuel melting, such as Passive Containment Cooling System. However, operation of these systems during transients does not affect the operation during accidents with fuel melting.

Due to the passive features with high reliability in AP1000 design, it is considered that objective of this requirement is met. This is confirmed by deterministic and probabilistic safety analyses (DCD sections 15 and 19).

No design changes to AP1000 standard design or additional analyses are expected.

**Article 49 (2)** belongs to CWO category and introduces risks related to **multiple failures.**

Common cause failures are considered in AP1000 design by introducing diversified safety functions and SSCs to strengthen the defense-in-depth concept, e.g., diverse means for core cooling, diversity in automatic depressurization and diversity in actuation system.

Common cause analysis is included in the AP1000 plant PRA as stated in the AP1000 plant DCD Section 19.29. The PRA was used to define where and to what degree diversity needed to be incorporated into the AP1000 plant SSCs.

No design changes to AP1000 standard design or additional analyses are expected.

**Article 52 (2-4)** belongs to CWO category and introduces risks related to **safety classification**.

Safety classification principles for AP1000 design is presented in DCD section 3.2. The classification system provides a means of identifying the extent to which structures, systems, and components are related to safety-related and seismic requirements. The classification of SSCs is slightly different than required in this article, e.g., safety class C includes components, which provides safety support functions to Class A, B and C SSCs.

The AP1000 classification system provides a means of identifying the extent to which structures, systems, and components are related to safety-related and seismic requirements. The classification system provides an easily recognizable means of identifying the extent to which structures, systems, and components are related to ANS nuclear safety classification, NRC quality groups, ASME Code, Section III classification, seismic category, and other applicable industry standards, as shown in DCD table 3.2-3.

No design changes to AP1000 standard design or additional analyses are expected.

**Article 84 (2)** belongs to CWO category and introduces risks related to **site-specific hazards and extreme events**.

External hazards are considered in AP1000 design, e.g., in DCD section 3.7 (seismic), flooding (section 3.4), wind and tornado loadings (section 3.3) and malevolent aircraft impact (appendix 19F).

No design changes to AP1000 standard design are expected, however site-specific hazards may require further consideration/analysis in the safety assessment.

**Articles 114 (3) and 211 (1)** belong to CWO category and introduce risks related to **remote shutdown station**. The remote shutdown workstation is provided for control of the plant in the case of an evacuation of the main control room (mainly due to fire). The remote shutdown station is not needed for other internal and external events.

Additionally, a secondary diverse actuation is in a diverse spatially separated location (not in the same zone of the plant) to actuate key safety functions such as ADS Stage 4 actuation, IRWST injection and containment recirculation actuation. The secondary DAS panel is powered by an independent local battery. The secondary DAS panel is located sufficiently far from the Main Control Room (MCR) and Remote Shutdown Workstation), its location has been selected as to provide additional protection so that it is very unlikely that it could be affected by internal events such as fire, internal flooding, or external events such as flooding (thus providing additional protection from these events).

This is considered complying with the objective, hence, no design changes to AP1000 standard design is expected. This topic is proposed to be follow-up to ensure this approach and discard potential additional analyses or design change. At this stage this is considered a Medium Risk.

**Article 130 (1)** belongs to COM-B category and introduces risks related to **containment isolation for 4 instrument lines**.

Containment Isolation related to four instrumentation lines is demonstrated according to Regulatory Guide 1.11. and alternate criteria.

No design changes to AP1000 standard design or additional analyses are expected.

**Article 137 (4)** belongs to COM-B category and introduces risks related to **support functions own protections**.

Due to the passive features of the AP1000 design, the number of supporting safety systems needed for accident management is limited, since main safety functions are performed by passive systems. Thus, active components are not classified as safety related to perform safety functions (they are included in the Defense in Depth type of Systems, thus not safety-related), for the previous reason the need to preclude their own protections is not as relevant and is to be analyzed only for Probabilistic Analyses.

**Article 173 (1-2)** belongs to COM-B category and introduces risks related to **commissioning tests**.

Commissioning tests will be provided as presented in DCD section 14. Test program will cover all necessary tests but with different test structure.

No design changes to AP1000 standard design or additional analyses are expected.

**Articles** **2 (1), 4 (3), 43** belong to COM-B category and introduce risks related to **radiation impact.**

Additional Analyses Needed.

DCD chapter 12 provides principles to ensure that radiation impact is kept as low as reasonably achievable. This has been assessed in reference [7] of the compliance assessment report [33] as COM-B since it is recognized that new analyses need to be performed. Due to comprehensive consideration of design features to limit radiation exposures (see e.g., DCD section 12), it is expected that there is no need for design changes. However, this needs to be confirmed with the analyses.

**Articles** **3 (1), 4 (3), 28** belong to COM-B category and introduce risks related to **site compatibility.**

Additional Analyses Needed.

The compatibility of the site and site conditions are studied in report KZG-GW-GL-100 (See reference [9] of the compliance assessment report [33]) and related to seismic and geotechnics in report KZG-GE01-X7R-001 (See reference [10] of the compliance assessment report [33]). Full compatibility of the proposed units with the conditions on site will need to be demonstrated taking into consideration the recommendations in these reports.

**Articles** **30 (2), 174** belong to COM-B category and introduce risks related to **multiple NPP’s on site.**

Additional Analyses Needed.

Selected site includes existing nuclear facilities. This may need further considerations in the design and safety analyses.

**Articles** **33 (5), 85, 86** belong to COM-B category and introduce risks **related to aircraft crash.**

Westinghouse has performed a rigorous assessment of the AP1000 plant design to demonstrate that the plant’s design features, and functional capabilities provide inherent protection against the effects of an aircraft impact, this assessment count with regulatory approvals that will need to be validated by BNRA.

**Articles** **44 (2), 49 (6), 79 (1-4)** belong to COM-B category and introduce risks related to **Probabilistic Risk Analysis (PRA).**

Additional Analyses Needed.

External hazards are considered in the AP1000 design by structural design solutions and passive design features. Similarly, spent fuel pool design is designed so that structural integrity is always confirmed and spent fuel pool cooling is always ensured. For these reasons, it is expected that there are no design changes needed in AP1000.

Probabilistic safety analyses are documented in DCD section 19, and it confirms that AP1000 design is robust against different initiating events and hazards with high safety margin. However, these topics may require further development needs in the PRA models and analyses. In addition, site-specific data may require adjustments in the PRA models.

**Articles** **33(5), 80 (5), 81, 83 (4, 6-7), 125 (3)** belong to COM-B category and introduce risks related to **site specific hazards and conditions.**

Additional Analyses Needed.

Each site has its specific characteristics, and it shall be confirmed that all hazards are considered in the design. Due to robust design solutions for AP1000, it is expected that there will be no major design changes needed due to site-specific characteristics and hazards. However, this may require new analyses to confirm that there are no hazards which could affect the safety of the AP1000.

See recommendations on references [9] and [10] of the compliance assessment report [33].

**Articles** **168 (1), 169 (1-3), 170 (1-2), 171, 172 (1-2), 173 (3-4), 217 (1-3), 218 (3), 221 (1-2), 226 (1), 227 (1-2)** belong to COM-B category and introduce risks related to **inspection, maintenance, testing.**

Specific/Additional Documentation might need to be produced.

AP1000 plant meets the design related to taking into consideration inspection and maintenance programs. Nonetheless the licensee shall prepare and implement documented programs of maintenance, testing, surveillance, and inspection of SSCs important to safety.

Westinghouse will provide input to the designer for maintenance, testing, surveillance, and inspection developed for AP1000 plant SSCs for the Owner to develop their programs.

The initial test program is described in DCD chapter 14. Detailed commissioning programme will be specified more in detail during the construction phase.

**Articles 188 (2-3), 189 (1-2), 190 (1-3), 191 (1-2), 192 (1-2), 203 (2-3), 206 (2)** belong to COM-B category and introduce risks related to **operating procedures, emergency procedures & SAMGs.**

Specific/Additional Documentation might need to be produced. No design Changes to Standard Design expected.

Emergency procedures and SAMGs are prepared during the construction phase. Standard AP1000 Procedures will be used as input.

**Articles** **213 (1-2)** belongs to COM-B category and introduces risks related to **chemistry program**.

Specific/Additional Documentation might need to be produced. No design Changes to Standard Design expected.

The chemistry program shall be prepared during the construction phase. It will be based on AP1000 standard Chemistry Manual, procedures, and specifications.

**Articles** **234 (2)** belongs to COM-B category and introduces risks related to **effluent monitoring.**

AP1000 plant is designed in a way that all liquid and gaseous release points are monitored using continuous radiation monitoring system RMS. AP1000 Effluent monitoring for Bulgaria shall be in agreement with local requirements, thus most likely it should conform with Recommendation 2004/2/Euratom which provides guidance to EU countries on the reporting of discharges of radioactive nuclides.

**Other topics of interest in the assessment:**

**Article 139 (2):** Fire response class A1 or A2 is required. This has been Classified as “NAS” (Not Assessable) as this might need to be studied to understand its potential impact. At this moment it is understood as low Risk since AP1000 Fire protection Approach is expected to be maintained.

The AP1000 design is protected against internal and external fires, see e.g., DCD Appendix 9A, where fire protection analysis is performed. For that reason, it is expected that there is no need for changes regarding fire protection. However, it needs to be understood if currently applied fire response class ratings comply with classes A1 and A2.

**Article 147 (4)** specifies that “The design shall specify the way for management of large quantities of liquid RAW generated in accident conditions”. This has been classified as owner requirement, since it does not affect AP1000 Standard Design. However, the extension/scope of this requirement could potentially add some additional scope that is currently not considered. If additional scope is requested, it needs to be fully understood.

## \*REGULATION ON ENSURING THE SAFETY IN SPENT FUEL MANAGEMENT

Following summary reflects results of the assessment conducted against reference [4] as part of the compliance assessment report [34] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [34] indicates that compliance with the requirements presented in Bulgarian “Regulation on Ensuring the Safety of Spent Nuclear Fuel Management” [4] is expected to be demonstrated but will require some additional site-specific analyses. Since no non-compliances have been identified and no design changes are anticipated, Bulgarian “Regulation on Ensuring the Safety of Spent Nuclear Fuel Management” has been assigned to a “**Medium Risk**” category.

Risks associated with the Bulgarian “Regulation on Ensuring the Safety of Spent Nuclear Fuel Management” are presented in “Identified Potential Risks To Be Addressed In Bulgaria Project” subsection.

In general, the AP1000 NPP provides highly reliable systems, structures, and components for the safe storage, handling, and cooling of nuclear fuel as explained in **Subsection 1.4** of the compliance assessment report [34].

### Roadmap of Items Specifically Discussed in the Regulation Assessment

#### Chapter 1 (Scope of application)

Article 1 of **Chapter 1** of the reference document [4] consists solely of explanatory statements.

Rest of the **Chapter 1** is responsibility of Owner and/or External Party.

#### Chapter 2 (Safety principles and criteria in spent nuclear fuel management)

The Owner will be responsible for meeting the statutory limits through appropriate Radiation Protection Program for the referenced plants.

Organizational measures, non-design measures to protect the public, personnel, and the environment are also the responsibility of the Owner.

Parts of the requirements that address plant design in **Chapter 2** of the reference document [4] are aligned with AP1000 Standard Design.

Development of a plant-specific SAR (and all its mandatory content) is the responsibility of the Owner, however Westinghouse provides the inputs to the Owner as appropriate and contracted, on AP1000 design, construction and operation items based on experience as the plant designer.

Requirements in **Chapter 2** which need an additional assessment or analysis to be fully met are expressed as COM-B items.

“Compliant with planned update for Bulgaria Units” (COM-B) items were identified in **Chapter 2** and are related to:

* **(Article 4)** **(1)** The annual individual effective dose , in all operational states, for the public from internal and external exposure due to the effects of liquid and gaseous releases to the environment from all on-site nuclear facilities; **(2)** The annual individual effective dose from internal and external exposure of the population on the border between the precautionary action zone and beyond; **(3)** The limit value of Caesium-137 releases to the atmosphere, that do not impose long-term restrictions in the use of soil and water in the urgent protective action planning zone in case of severe accidents; **(4)** Limit for frequency for large radioactive releases to the environment for which it is necessary to take urgent protective measures for the public.
* **(Article 6)** Establishing compliance with the principles, the achievement of the objectives and the fulfilment of the safety criteria by the SAR.

*Refer to* ***Subsection 2.3*** *of the compliance assessment report [34] for more details.*

#### Chapter 3 (Nuclear safety)

Majority of the design related requirements in **Chapter 3** of the reference document [4] are met by Westinghouse either “as stated” or via compliance with their objectives. Some of the requirements are expected to be met as a result of “COM-B” scope.

The Owner is responsible for ensuring the control of the performance of safety functions in all SNF management activities during normal operation of the facilities and in the event of design basis accidents.

Owner shall control the burnup according to the design.

Organizational measures for SNF management are the responsibility of the Owner.

Site characterization and selection is the responsibility of the Owner. The AP1000 plant license applicant is to demonstrate that the site meets the generic site characteristics and design parameters specified in DCD Chapter 2 or appropriate site-specific design considerations are incorporated to accommodate any site-specific deviation from the generic site parameters assumed for the standard plant design.

Owner will be responsible for assessing the impact of the SNF management facilities on the population (present and future) and on the environment in the site selection. The AP1000 plant is designed with administrative programs and procedures to maximize the incorporation of good engineering practices and lessons learned to accomplish ALARA objectives for the population and minimize the impact on the environment. Fuel handling activities stand for only a small portion of the overall dose.

The Owner is responsible for ensuring suitability of the site for SNF management facilities, taking into account the specific characteristics of the site, and development and implementation of an operational radiation protection program that, in conjunction with the AP1000 NPP plant ALARA design features, will support compliance with operational dose limits.

Providing a site-specific Preliminary Safety Analysis Report (PSAR) and all its associated mandatory content is the responsibility of the Owner. However, Westinghouse can, as contracted and consistent with Westinghouse’s role as the plant designer, provide appropriate inputs to the Owner in terms of AP1000 plant design, construction and operation to support site licensing.

The AP1000 plant design for SNF storage and handling includes provisions for the mitigation of BDBA and their possible consequences. Organizational and non-design measures to limit such consequences, are the responsibility of the Owner. Site-specific analyses of accidents and events external to the nuclear plant that are potential hazards within the site vicinity are to be performed by the Owner as well.

The owner will be responsible for implementing procedures for refueling and outage planning, including accounting for fuel assemblies.

The Owner is responsible for addressing site-specific information related to security, contingency, and guards training plans. The development of the Physical Security Plan, the Training and Qualification Plan, and the Safeguards Contingency Plan is also the Owner’s responsibility.

Handling and dry storage of the loaded SNF casks is outside of AP1000 standard design and is the responsibility of the Owner, however the AP1000 NPP is designed to support utilization of multiple commercially-available spent fuel storage container designs in order to provide the operator flexibility in their spent fuel management program.

The design of SNF casks (including any neutron absorbers potentially credited in their design) is the responsibility of the Owner.

The AP1000 plant design of Fuel Storage and Handling Area does have all the specified functions except of drying and cooling of the transport casks, as the dry storage and design of the casks is the responsibility of the Owner. However, AP1000 plant design provides areas for cask loading and cask washdown, respectively named Cask Loading Pit and Cask Washdown Pit, next to the SFP. Additionally, cask handling crane is provided in the standard AP1000 plant design. The AP1000 NPP is designed to support utilization of multiple commercially-available spent fuel storage container designs in order to provide the operator flexibility in their spent fuel management program.

During plant operation, the inspection of the spent fuel is by the Owner. The Owner will be responsible to take the actions to address discovery of damaged fuel in the spent fuel pool or dry cask storage. During the refueling process, fuel is inspected for damage prior to reinsertion in the reactor or in the Spent Fuel Pool. Storage or transport cask for dry storage usually include damaged fuel cans designs that allow the insertion of damaged fuels inside them, thus allowing sealed dry storage but this is a requirement of the Owner.

Spent fuel transport and storage cask/packages are to be selected by the Owner and are not part of the standard design. Usually available designs prevent and accident with the uncontrolled release. On the other hand, fuel handling accident (for a single assembly) is considered a DBA.

The AP1000 plant is designed to support utilization of multiple commercially-available spent fuel storage cask designs in order to provide the operator flexibility in their spent fuel management program. The stated requirements for the plant transportation package (PTP) are, in the experience of Westinghouse, typical spent fuel storage system requirements that would be complied with by credible licensed designs.

Ensuring the statutory requirements for the dose limits are met is the responsibility of the Owner. The AP1000 plant’s spent fuel handling equipment is designed and constructed to ensure the subcriticality of the SNF and temperature of the SFP are maintained. Additionally, the design and conduct of fuel handling operations is such that personnel exposures can be maintained ALARA in coordination with site-specific operational and radiation protection programs.

Section II (General requirements to ensuring nuclear safety) contains COM-B items related to:

* **(Article 16 (4))** Requirement for the design of the SNF management facilities to include a section on decommissioning.
* **(Article 23) (1)** Designing, construction and installation of the structures, systems and components of the safety significant facilities taking into account the natural phenomena characteristic of the site area, such as earthquakes, hurricanes and floods, the possibility of externally caused fires, and the possibility of external events and factors of technogenic origin occurring on or near the site. (2) Assumption of quantitative values of the parameters of processes, phenomena and factors of natural and technogenic origin. **(2)** Assuming quantitative values of the parameters of processes, phenomena and factors of natural and technogenic origin in design in accordance with the regulations.
* **(Article 30 (2))** Requirement for the design to include a plan for evacuation from buildings and facilities, characteristics of evacuation routes, and location of signaling devices.
* **(Article 34)** Ensuring the fulfilment of the requirements related to the control, storage, on-site transportation and physical protection of nuclear material arising from the obligations of the Republic of Bulgaria under international treaties ratified, promulgated and in force for the Republic of Bulgaria.

“Compliant with objective” (CWO) items were identified in two sections.

Section I (Safety functions) contains CWO items related to:

* **(Article 14 (2))** Requirement for the system of physical barriers shall include a minimum of two barriers for all SNF management activities.

Section II (General requirements to ensuring nuclear safety) contains CWO items related to:

* **(Article 16 (1)) Requirement** for the design of the SNF management facilities to contain a preliminary SAR for the storage, on-site transportation and handling of SNF during normal operation and in design basis and beyond design basis accidents. After the construction of the facilities – updating SAR in accordance with the current status of the facility.
* **(Article 24 (3))** Topics to be ensured by design.
* **(Article 36)** Means of residual heat removal from SNF in order to prevent fuel rod leaks, damages and release of radioactive substances.
* **(Article 37 (2))** Storing spent nuclear fuel under water under continuous monitoring of water parameters such as temperature, level, activity, medium acidity (pH), electrical conductivity, chemical composition, and hydrogen content, as required.
* **(Article 39)** Requirement for all piping to be located in the upper part of the pools to provide a regulated water level above the fuel in the event of a possible rupture. The pools shall be drained with submersible type of pumps. In normal operation, no electrical power shall be provided to these pumps.
* **(Article 41)** Designing the pool water make-up system to have a bigger flow rate than the one of the pool water purification system.
* **(Article 47)** Storing leaking assemblies in specially designed canisters, which shall withstand the temperature and pressure resulting from residual heat removal from the assemblies and be resistant to chemical reactions between the fuel and its cladding with the working medium of the canister.
* **(Article 59)** Requirements to the electric motors of the transport and handling equipment mechanisms, the failure of which can lead to an accident.
* **(Article 60 (2))** Requirement for the design to provide for automatic stopping of the assemblies handling devices in the event of an earthquake with an intensity exceeding the values specified in the design.
* **(Article 67)** Requirement for the equipment for SNF handling to be fire resistant.

*Refer to* ***Subsection 2.4*** *of the compliance assessment report [34] for more details.*

#### Chapter 4 (Operation and decommissioning of spent nuclear fuel management facilities)

Majority of the design related requirements in **Chapter 4** of the reference document [4] are met by Westinghouse either “as stated” or via compliance with their objectives. Some of the requirements are expected to be met as a result of “COM-B” scope.

The compliance of operations on SNF and related systems with the applicable Bulgarian regulation is the responsibility of the operating personnel, thus of the Owner.

Requirements on the operating personnel, management staff and plant’s procedures are the responsibility of the Owner.

The design basis for items critical to operational safety are documented at the design stage for use as an input to the plant Owner which is responsible for creating the operating procedures and technical specifications for safe operation.

During plant operation, the inspection of the spent fuel is performed by the Owner. The Owner will be responsible to take the actions to address discovery of damaged fuel in the spent fuel pool or dry cask storage. During the refueling process, fuel is inspected for damage prior to reinsertion in the reactor or in the Spent Fuel Pool.

Establishing instructions for receipt of SNF for final storage facilities or reprocessing plants is the responsibility of External Party.

The operation on SNF casks and on-site dry storage is the responsibility of the Owner. Off-site management of SNF is generally the responsibility of the Owner as well.

Plant’s (SSCs for SNF management) aging management and qualification and re-qualification system as well as maintenance of SSCs related to SNF are the responsibility of the Owner, which is also responsible for scheduling of condition maintenance activities and ensuring the safety of the maintenance personnel and safety of the equipment.

Technical inputs on the design basis for safe operation are provided to the plant Owner, and such documentation is a part of Quality Assurance Program which is a responsibility of the Owner.

Provision for and use of measuring instruments for calibration-checking are within the scope of the Owner.

Fulfilment of the requirements related to changes in the design and in the operating practice are responsibility of the Owner. Westinghouse proceeds to design changes taking into account a methodology that follows safety evaluations as required by 10 CFR 50.59 regulation. Westinghouse does have a Program for AP1000 Design Change Control.

Section IV (Decommissioning) entirely fall under responsibility of the Owner.

One COM-B item was identified in Section I (General) and is related to:

* **(Article 80 (2))** Data that should be included to the information necessary for the implementation of licensee’s strategy for the management of SNF, including the safe **decommissioning of the facilities**.

One CWO item was identified in Section I (General) and is related to:

* **(Article 82)** Conditions related to storage, placement and transport routes to SNF that have to be fulfilled as part of organization of the activities and their implementation done by the licensee.

*Refer to* ***Subsection 2.5*** *of the compliance assessment report [34] for more details.*

#### Chapter 5 (Radiation protection)

All relevant requirements in **Chapter 5** of the reference document [4] are met by Westinghouse “as stated”, apart from requirement addressing decommissioning which belongs to “COM-B” scope.

Emergency planning and maintenance of emergency preparedness is the responsibility of the Owner.

Owner is responsible for ensuring radiation control and protection of the operating personnel and for ensuring the statutory regulations are met. The Owner will be responsible for meeting the radiation protection standards through appropriate Radiation Protection Program for the referenced plants.

Owner will shall ensure that the radiation control system includes portable dosimeters and devices for radiation monitoring and that an operating radiation monitor is mounted on any crane or fuel handling machine when it is handling fuel.

Finally, the Owner is responsible for design to define a control network of monitoring points at the site, in the precautionary action zone and the supervised area, monitored parameters and their baseline values. The Owner has to ensure that monitoring data is systematized and stored reliably for the entire service life of the facilities.

One COM-B item was identified in Section I (General) and is related to:

* **(Article 96 (1))** Means of radiation protection during the commissioning, operation and decommissioning stages.

*Refer to* ***Subsection 2.6*** *of the compliance assessment report [34] for more details.*

#### Chapter 6 (Safety assessment)

Large majority of relevant design related requirements in **Chapter 6** of the reference document [4] are met by Westinghouse either “as stated” or via compliance with their objectives. One requirement which involves addressing of safety aspects related to the decommissioning in SAR was assigned to the “COM-B” scope.

Providing PSAR and all its mandatory content is the responsibility of the Owner, however WEC provides all the input to the Owner in terms of storage, transport and handling of SNF in normal operation, design and beyond design basis accidents.

Owner is responsible for performing periodic and systematic safety assessments of the plant to ensure safe operation, updating the safety analysis report with crucial information to meet the regulatory and safety requirements as well as taking into account the results of the safety analyses of the on-site SNF storage, handling and transport when developing the operating procedures and updating the on-site emergency plan.

Reshuffling of the assemblies in casks and packages is not analyzed because dry storage and transport of the SNF is not part of AP1000 plant standard design and is the responsibility of the Owner. Also, Independent Spent Fuel Storage Installations (ISFSI) and casks are not supplied with the AP1000 and are Owner's scope.

One COM-B item was identified in this Chapter and is related to:

* **(Article 110 (3))** Requirement for safety analysis report to reflect the actual condition of the facilities throughout their lifetime and during decommissioning and safety aspects that shall be addressed in it.

Several CWO items were identified in this Chapter and are related to:

* **(Article 111) (2)** A recommended list of initiating events for the analysis of design basis accidents. **(3)** A recommended list of initiating events for the analysis of beyond design basis accidents.
* **(Article 112)** Events and their possibilities that shall be assessed in the analysis of initiating events under Article 111 (2) and (3).
* **(Article 113)** List of conservative requirements according to which the subcriticality analysis shall be conducted at the maximum neutron multiplication factor for all considered states.
* **(Article 114)** List of conservative requirements under which the residual heat removal analysis shall be conducted.

*Refer to* ***Subsection 2.7*** *of the compliance assessment report [34] for more details.*

#### Chapter 7 (Management System)

Implementation of the integrated management system ensuring the safety of SNF management, discussed in **Chapter 7** of the reference document [4], is the requirement to the Owner.

#### Additional provisions

**Additional provisions** to the reference document [4] contain only one requirement to the External party while the rest are explanatory statements.

#### Transitional and final provisions

Westinghouse demonstrates compliance with objective of the design related requirements presented in **Transitional and final provisions** to the reference document [4].

Safety analysis report is the responsibility of the Owner. However, The Design Basis Accidents and Beyond Design Basis Accidents are included in the AP1000 plant design and provided to the Owner as the input to safety analysis report.

Rest of the requirements are not applicable to this scope of work, as they refer to existing facilities or are the responsibility of an External Party.

Several CWO items were identified in this Chapter and are related to:

* **(Annex 1 to Article 12 (2))** Recommended list of initiating events for safety analysis of SNF management facilities in design basis accidents.
* **(Annex 2 to Article 111 (3))** Recommended list of initiating events for safety analysis of SNF management facilities in beyond design basis accidents.

*Refer to* ***Subsection 2.10*** *of the compliance assessment report [34] for more details.*

### Non Compliance

No “Non Compliances” have been identified in the assessment [34].

### Identified Potential Risks To Be Addressed In Bulgaria Project

The analyses of section 2 of the compliance assessment report [34] show that the following articles and sub-articles need to be considered in the risk assessment:

* **Article 4, Points (1) to (4)** on AP1000 plant releases and associated doses to the workers and the public. Previous studies have shown that AP1000 plant will meet this regulation, nevertheless, a detailed assessment will need to be performed to confirm these preliminary results of feasibility at further steps.
* **Article 6** on providing the input, as per Bulgarian requirements, to the Owner for the Bulgarian Project in terms of SAR as appropriate and contracted, on the AP1000 plant design, construction and operations items.
* **Article 16, Point (4); Article 80, Point (2); Article 96, Point (1); Article 110, Point (3)** on decommissioning of the SNF management facilities. At later stages of the project a specific decommissioning plan might be requested and provided by Westinghouse upon the further agreements.
* **Article 23, Points (1) and (2)** on site-specific external hazards to the AP1000 power plant. Provided that the selected site is bounded by the generic site parameters considered for the standard design identified in DCD Table 2-1, the design will be expected to be acceptable without the need for site-specific reconciliation. If necessary, the analyses and, if needed, design modifications, to reconcile the design for new conditions, depending on a site-specific data and site-specific regulatory expectations about recurrence intervals, can be performed using the similar methodologies and acceptance criteria as used for the generic design. Similarly, in the event that site-specific external events are identified for a specific site that are not bounded by the existing AP1000 plant design analyses, additional analysis can be performed to demonstrate either acceptability of the design or if necessary, identification of design modifications to meet the site-specific hazard.
* **Article 30, Point (2)** on evacuation plans and evacuation routes at the NPP site. As appropriate, site-specific studies will need to be performed to review applicable Bulgarian life-safety regulations/standards and assess the specific compliance of the AP1000 NPP design, as well as define and justify necessary exemptions to those requirements pursuant to the requirements of nuclear safety and security.
* **Article 34**. No design changes are currently expected to be required to meet other obligations arising from international treaties (control, storage and physical protection on site) signed by Bulgaria, but will have to be confirmed during site-specific project development.
* **Article 113**. **Point 8** on credit for soluble boron for subcriticality analysis in the spent fuel pool. Generally, no soluble boron is required to maintain subcriticality below Keff<0.95 in the Spent Fuel Pool for most the credited accidents. In the case of a mislocated or misloaded maximum enrichment fresh fuel assembly, to maintainKeff<0.95, the analysis concludes that 800 ppm of boron is the maximum required soluble boron concentration for the most limiting condition.

Indeed, the criticality analyses performed for the AP1000 plant design and which support the certification of the standard design reflect two requirements:

* + - First, keff must be maintained less than or equal to 0.95 with the Spent Fuel Pool loaded with fuel of the maximum fuel assembly reactivity when crediting boron for normal operation as well as design basis events such as misloading or mislocated fuel.
    - Second, it must be shown that keff will remain below 1.0 in the unlikely event of a complete loss of boron.

The double contingency principle outlined in Section 4.2.2 of ANSI/ANS-8.1-1998;R2007;W2014, “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors” indicates that it should take two independent, unlikely, and concurrent events for a criticality to occur. This combination of events is considered outside of the design basis of the plant. Because soluble boron is not credited to ensure that all normal operating conditions remain subcritical (keff< 1.0 at 0 ppm soluble boron), the soluble boron present in the pool can be credited to offset the reactivity impacts associated with accident conditions. For accidents that are independent of a soluble boron dilution event, the full minimum soluble boron requirement in the Technical Specifications can be credited. For accidents which are not independent of the soluble boron accident (they share a common mode initiator), soluble boron credit can be taken up to the minimum credible concentration identified in the boron dilution analysis.

Dilution of the spent fuel pool water to below this concentration is considered highly unlikely.:

* + - The Spent Fuel Pool is initially filled with water at a nominal boron concentration of 2700 ppm.
    - Demineralized water can be added for makeup purposes (e.g., to replace evaporation losses from the demineralized water transfer and storage system). Boron concentration in the Spent Fuel Pool during operation is monitored via manual sampling occurring at a minimum of every 7 days. If the sampling shows any decrease in boron concentration outside Technical Specification limits, immediate actions are required to be taken. If needed, boron is added to the Spent Fuel Pool from the chemical and volume control system (CVS).
    - Dilution of the Spent Fuel Pool cannot be caused by a single design basis external or internal event, such as flooding. Furthermore, it is not expected that a beyond design basis (BDB) external flood event could cause a boron dilution of the SFP. Indeed the top of the Spent Fuel Pool is located at elevation 135 ft (110.7 m) compared to the design basis flooding which is defined as elevation 100 ft (100 m). This provides a very large margin even against BDB flooding.
    - Therefore, the only credible initiator of a dilution in the Spent Fuel Pool is operator error such as opening manual valves. There are different ways of detecting a dilution event. Alarms are provided in the control room for a high level in the Spent Fuel Pool, water detected in normally empty tanks, high levels in normally full tanks as the Spent Fuel Pool water spills into the adjacent tanks, and low levels in the dilution sources. There will be either high-ranked alarms or multiple alarms for any Spent Fuel Pool dilution events. In addition, these specific alarm combinations are unique for this event case, and the operators will repeatedly see this event at their simulator training. Taken together, these items make it unlikely that the operators will not interpret the alarms correctly.

It should also be noted that the water in the spent fuel pool will become intimately mixed with the water in the reactor coolant system during fuel handling and refueling operations. As such, the spent fuel pool must remain borated to ensure subcriticality in the reactor even if no credit is taken for soluble boron in the spent fuel pool criticality analyses.

However, if this accidental condition must be considered in coincidence with the severe inadvertent boron dilution in the spent fuel pool, additional measures can be taken in the design. These include elimination of the number of available spaces in Rack 2 storage; including caps on some positions in the spent fuel pool which will not allow the insertion of elements in such positions; or if so desired, but not envisaged, proceed to a redesign of Racks in Region 2 more similar to the Racks in Region 1 (since this new design will not need soluble boron).

Input from the Customer for future discussions of this topic is Included as Attachment 1 to this document.

## \*REGULATION ON SAFE MANAGEMENT OF RADIOACTIVE WASTE

Following summary reflects results of the assessment conducted against reference [5] as part of the compliance assessment report [35] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [35] indicates that compliance with the requirements presented in Bulgarian “Regulation on Safe Management of Radioactive Waste” [5] is expected to be demonstrated but will require some additional site-specific analyses and various activities to be performed. Since no non-compliances have been identified and no design changes are anticipated, Bulgarian “Regulation on Safe Management of Radioactive Waste” has been assigned to a “**Medium Risk**” category.

Risks associated with the Bulgarian “Regulation on Safe Management of Radioactive Waste” are presented in “Identified Potential Risks To Be Addressed In Bulgaria Project” subsection.

### Roadmap of Items Specifically Discussed in the Regulation Assessment

#### Chapter 4 (Radioactive Waste Classification)

**Chapter 4** of the reference document [5] contains two articles related to plant design (articles 6-7) with article 7 not being a requirement.

One “Compliant with planned update for Bulgaria Units” (COM-B) item was identified in **Chapter 4** and is related to:

* **(Article 6)** Classification of Radioactive Waste and associated categories.

*Refer to* ***Subsection 2.2*** *of the compliance assessment report [35] for more details.*

#### Chapter 5 (Radiation Protection Requirements)

**Chapter 5** of the reference document [5] contains two articles related to plant design (articles 8-9).

One COM-B item was identified in **Chapter 5** and is related to:

* **(Article 8)** The individual effective dose for the respective critical group of members of the public resulting from RAW management activities and/or following the normal operation of all nuclear facilities located on a single site

One “Compliant with objective” (CWO) item was identified in **Chapter 5** and is related to:

* **(Article 9)** Limit on the estimated individual effective dose to relevant critical groups of the populations at the site boundary in the event of a design basis accident at a RAW management facility.

*Refer to* ***Subsection 2.3*** *of the compliance assessment report [35] for more details.*

#### Chapter 6 (Radioactive Waste Generation and Treatment)

**Chapter 6** of the reference document [5] contains four articles related to plant design (articles 10-13).

Articles 10-12 fully belong to the scope of responsibility of the Owner.

One COM-B item was identified in **Chapter 6** and is related to:

* **(Article 13) (1)** Collection and segregation of waste in pretreatment; **(2)** Taking into account **t**he requirements for volume reduction and/or extraction of radionuclides, as well as modification of their characteristics to facilitate their subsequent storage and/or conditioning in the RAW treatment; **(3)** Conditioning of the radioactive waste for conversion to a specific form and required characteristics of such form; **(4)** Meeting the acceptance criteria for storage and/or disposal by the conditioned RAW; **(5)** Development and implementation of technical specifications for the RAW packaging that comply with the handling and transport requirements and the acceptance criteria for storage and/or disposal.

*Refer to* ***Subsection 2.4*** *of the compliance assessment report [35] for more details.*

#### Chapter 7 (Radioactive Waste Storage)

**Chapter 7** of the reference document [5] contains three articles related to plant design (articles 14-16).

Westinghouse fully complies with article 14. However, the provisions for RAW to be cleared from regulatory control and to be treated as conventional material are within the scope of the Owner as part of the site-specific Process Control Program.

Articles (15-16) are entirely within responsibility of the Owner.

#### Chapter 9 (Radioactive Waste Management Facilities)

**Chapter 9** of the reference document [5] contains twenty six articles related to plant design (articles 21-46) in Sections I-VII.

Section V (Operation) is entirely within responsibility of the Owner.

Section VI (Closure) and Section VII (Control after closure) were deemed not applicable.

AP1000 is a Nuclear Power Plant NPP regulations and not A RAW Facility itself, even though it contains Radwaste Storage in the Auxiliary Building and Radwaste Building, thus it will follow NPPs regulation, thus Chapter 9 is not applicable to the AP1000 Standard Design as presented in the DCD [28]. In case if it was needed to develop a site specific SRTF it could be considered as a RAW facility.

The design of the SRTF Site Radwaste Treatment Facility for Bulgaria, in case it was later decided to include one, will need to consider appropriate provisions to meet requirements presented in “Section III (Design)” in **Chapter 9**. As a result, a number of articles were assigned to “COM-B” scope.”

COM-B items that were identified in Section III (Design) are related to:

* **(Article 28)** **(1)** Taking into account the national policy and strategy by The design of the RAW management facility and associated requirements to be fulfilled; **(2)** Classification of the SSCs according to their safety significance and define the operational states and safety limits; **(3)** Applying a graded approach to the design decisions depending on the risk the RAW poses.
* **(Article 30)** Criteria and constraints to be included by the design limits.
* **(Article 34)** Requirements to be observedwhen designing a RAW processing facility.
* **(Article 35) (1)** Requirements to be fulfilled by technical solutions that should be contained in the design of a RAW storage facility; **(2)** Having inherent properties of passive safety by the design pursuant and keeping application of the active safety systems to the minimum.

One CWO item was identified in Section III (Design) and is related to:

* **(Article 32) (1)** Classification of SSCs into SSCs important to safety and SSCs not related to safety; **(2)** Parameters to be determined for SSCs important to safety; **(3)** Withstanding the conditions of postulated initiating events with sufficient margins by SSC important to safety; **(4)** Analysis and consideration of the potential for common cause failures in order to identify the cases, where application of the principles of diversity, redundancy and independence is required to achieve the necessary reliability.

*Refer to* ***Subsection 2.6*** *of the compliance assessment report [35] for more details.*

### Non Compliance

No “Non Compliances” have been identified in the assessment [35].

### Identified Potential Risks To Be Addressed In Bulgaria Project

The assessment conducted in Section 2 of the support report [35] indicates that the following articles and sub-articles need to be considered in the risk assessment:

* **(Article 6)** Waste categorization in Bulgaria requires consideration in the site-specific operational programs for radwaste management. This should also specifically apply in the case of needing to develop a SRTF. Per discussion with KNPP-NB, that happened after the assessment [35] issuance, see letter L.WEC\_KNP\_230019 [58], on the Role of SERAW Bulgarian State Enterprise for “Radioactive Waste”. It was Clarified by KNPP-NB that SERAW will be responsible for the predisposal treatment and Conditioning of Radwaste (e.g., Volume Reduction, Immobilization, Waste Packaging for Pre-Disposal conditioning) if the radwaste (e.g., Resins and Liquids) to be delivered to SERAW agreed to the Radwaste that can be accepted by SERAW. In this aspect KNPP-NB requested Westinghouse to provide data on our waste streams to confirm their acceptability by SERAW, this was addressed by transmitting the document BGP-GW-GEH-001[60] as attachment 1 to L.WEC\_KNP\_230025 [59]. Hence currently Envisaged SERAW activities will preclude the need to include treatment systems in a Site Specific Radwaste Treatment Facility (SRTF) and possibly even the need to consider the need for one (unless some additional storage is required).
* **(Article 8)** Feasibility of the radiation dose constraints with relation to the critical group of members of the public was preliminary discussed in Feasibility Study (TD-WES-13-002 PART II Revision II), however specific analyses need to be developed.

## \*GSR PART 3 RADIATION PROTECTION AND SAFETY OF RADIATION SOURCES

Following summary reflects results of the assessment conducted against reference [6] as part of the compliance assessment report [36] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [36] indicates that compliance with the requirements presented in “IAEA Safety Standards No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards” [6] is expected to be demonstrated but will require some additional analyses. Since no non-compliances have been identified and no design changes are anticipated, “IAEA Safety Standards No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards” has been assigned to a “**Medium Risk**” category.

Risks associated with the “IAEA Safety Standards No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards” are presented in “Identified Potential Risks To Be Addressed In Bulgaria Project” subsection.

These risks are related to:

* Differences in framework between IAEA and the United States (NRC). These differences are analyzed more detailed in subsection 1.4 and 1.7 of the compliance assessment report [36], and might in some specific cases require some additional analyses, but are not expected to lead to any major design changes,
* Some minor design changes might be requested for some plant and effluent monitors, however since they are considered as minor, we maintain the classification as medium risk.

The AP1000 plant is designed with administrative programs and procedures to maximize the incorporation of good engineering practices and lessons learned to accomplish ALARA objectives. This has been reviewed and verified by US-NRC Reviews of Standard Design Certification, the beginning of the Operation at Vogtle Unit 3, China Operating Plants, as well as by the Generic Design Assessment in the United Kingdom (UK).

Thus, the AP1000 design should be able to accommodate the IAEA Standard, with some specific dose analyses that need to be performed for compliance, and some minor changes in some radiation and effluent monitors.

### Roadmap of Items Specifically Discussed in the Safety Requirements Assessment

#### Section 2 (General requirements for protection and safety)

Majority of the requirements in **Section 2** of the reference document [6] are responsibility of the Owner/Licensee and/or External Parties.

“Application of principles of radiation protection” subsection elaborates on Requirement 1 which falls under responsibility the Owner/Licensee and/or external parties as well as Westinghouse. Westinghouse demonstrates compliance with the presented requirement and related paragraphs either “as stated” or via compliance with their objectives.

One “Compliant with Objective” (CWO) item was identified in this subsection and is related to:

* **(Item 2.11)** Ensuring that specified dose limits are not exceeded for planned exposure situations other than for medical exposure.

Subsequent two subsections elaborate on responsibilities of the Government and Regulatory Body.

“Responsibilities for protection and safety” and “Management requirements” subsections fall within responsibility of the Owner/Licensee and/or other external parties.

*Refer to* ***Subsection 2.2*** *of the compliance assessment report [36] for more details.*

#### Section 3 (Planned exposure situations)

Majority of requirements in **Section 3** of reference document [6] are either not applicable to the technology or are a responsibility shared between Owner/Licensee and other external parties.

Requirements applicable to the AP1000 technology are met by Westinghouse either “as stated” or via compliance with their objectives.

CWO items were identified in three subsections.

“Generic requirements” subsection contains a CWO item related to:

* **(Item 3.28)** Dose limits.

“Occupational exposures” subsection contains a CWO item related to:

* **(Requirement 21)** Responsibilities of employers, registrants and licensees for the protection of workers. Even when it is recognized as a responsibility of the Owner/Licensee or/and other external parties

“Public exposure” subsection contains a CWO item related to:

* **(Item 3.125)** Responsibilities of relevant parties specific to public exposure

*Refer to* ***Subsection 2.3*** *of the compliance assessment report [36] for more details.*

#### Section 4 (Emergency exposure situations)

Requirements in **Section 4** of reference document [6] fall under responsibility of the Owner/Licensee and/or external parties.

One CWO item was identified in “public exposure” subsection and is related to:

* **(Item 4.8)** Development of protection plan with a reference level expressed in terms of residual dose shall be set, typically an effective dose in the range of 20–100 mSv

*Refer to* ***Subsection 2.4*** *of the compliance assessment report [36] for more details.*

#### Section 5 (Existing exposure situations)

Requirements presented in **Section 5** of reference document [6] belong to the responsibility scope of the Owner / Licensee and/or Government, Regulatory Body or other external parties.

### Non Compliance

No “Non Compliances” have been identified in the assessment [36].

### Identified Potential Risks To Be Addressed In Bulgaria Project

The analyses of section 2 of the support report [36] show that the following requirement and sub-requirements need to be considered in the risk assessment:

Requirements related to different expected Occupational dose as explained in subsection 1.4 of the compliance assessment report [36]:

* **Requirement 1**: Application of the principles of radiation protection.
* **Requirement 12**: Dose limits.
* **Requirement 21**: Responsibilities of employers, registrants and licensees for protection of workers.

Compliance with dose limits and dose constraints for occupationally exposed workers and members of the public has to be demonstrated. As explained in subsection 1.4 of the compliance assessment report [36], the current AP1000 design is capable of fulfilling this requirement with appropriate radiation programs in place, even though these limits differ from the ones used in 10 CFR 20. Thus conservatively, this is considered as CWO since at some point there could be potentially the need to make additional analyses.

Other requirements based on differences between US-NRC radiation protection regulatory framework and IAEA for Radiation Protection:

* **Requirement 30**: Responsibilities of relevant parties specific to public exposure. Application of the IAEA General Safety Requirements Part 3 and its Dose Limits and Constraints.
* **Requirement 44**: Preparedness and response for an emergency. Development of a protection strategy with A reference level expressed in terms of residual dose shall be set, typically an effective dose in the range of 20–100 mSv.

Realistic assessment of the doses to members of the public and for comparison with dose constraints due to releases has to be done. A specific calculation will be needed for releases in normal operation and accidental/emergency conditions, even though some feasibility studies of compliance were performed previously.

## \*GSR PART 4 SAFETY ASSESSMENT FOR FACILITIES AND ACTIVITIES

Following summary reflects results of the assessment conducted against reference [7] as part of the compliance assessment report [37] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [37] indicates that compliance with the requirements presented in “IAEA Safety Standards No. GSR Part 4, Safety assessment for facilities and activities” [7] is expected to be demonstrated but might require an additional decommissioning related scope to be performed. Since no non-compliances have been identified and no design changes are anticipated, “IAEA Safety Standards No. GSR Part 4, Safety assessment for facilities and activities” has been assigned to a “**Medium Risk**” category.

Risks associated with the “IAEA Safety Standards No. GSR Part 4, Safety assessment for facilities and activities” are presented in “Identified Potential Risks To Be Addressed In Bulgaria Project” subsection.

### Roadmap of Items Specifically Discussed in the Safety Requirements Assessment

#### Section 2 (Basis for requiring a safety assessment)

Assessment of **Section 2** of the reference document [7] concludes that the Owner will be responsible to identify the safety assessments to be performed over the lifetime of the plant and the level of resources required. The prime responsibility for safety during the lifetime of a Nuclear Power Plant rests with the licensed operator.

As the designer, Westinghouse supports the Owner in meeting the objectives described in **Section 2** of the compliance assessment report [37], by providing input to the safety assessment.

#### Section 3 (Graded approach to safety assessment)

Assessment of **Section 3** of the reference document [7] concludes that the Owner will be responsible to identify the safety assessments to be performed over the lifetime of the plant and the level of resources required. The prime responsibility for safety during the lifetime of a Nuclear Power Plant rests with the licensed operator.

As the designer, Westinghouse will provide input to the safety assessment, such as the DCD content and project-specific Preliminary Safety Analysis Report, depending on the project-specific scope.

#### Section 4 (Safety assessment)

Assessment of **Section 4** of the reference document [7] concludes that the Owner will be responsible to identify the safety assessments to be performed over the lifetime of the plant and perform their required scope within the safety assessments.

Overall, the DCD provides documentation that the AP1000 plant meets the requirements specific to the plant designer described in **Section 4** for a safety assessment.  In some cases, these are met with CWO due to differences in terminology or country-specific requirements (for example radiation analysis limits) which may differ from the US, but have the same intent. The requirement addressing availability of the background information relating to decommissioning of the facility was marked as CWO as decommissioning is not included in the DCD, however Westinghouse has experience in providing the necessary information.

There will be project specific aspects of the safety assessment for a specific AP1000 project.  For example, review of site specific characteristics, external hazards as well as analysis that is performed to confirm that radiation risks to individuals and population groups specific to the site are bounded by the limits established for the standard plant design. Evaluation of meeting country-specific limits that may differ from the US licensing basis is performed on a project specific basis.

“Compliance with Objective” (CWO) items were identified in four subsections.

“Requirement 4: Purpose of the safety assessment” subsection contains CWO items related to:

* **(Requirement 4)** Primary purposes of the safety assessment
* **(Item 4.4)** Including an assessment of the provisions in place for radiation protection, to determine whether radiation risks are being controlled within specified limits and constraints, and whether they have been reduced to a level that is as low as reasonably achievable.
* **(Item 4.5)** Safety assessment addressing all radiation risks that arise from normal operation (that is, when the facility is operating normally or the activity is being carried out normally) and from anticipated operational occurrences and accident conditions (in which failures or internal or external events have occurred that challenge the safety of the facility or activity).
* **(Item 4.10)** Safety assessment addressing all the radiation risks to individuals and population groups that arise from operation of the facility or conduct of the activity.
* **(Item 4.11)** Safety assessment addressing radiation risks in the present and in the long term.

“Requirement 5: Preparation for the safety assessment” subsection contains a CWO item related to:

* **(Item 4.18)** Objectives to be ensured with the necessary preparations.

“Requirement 9: Assessment of the provisions for radiation protection” subsection contains CWO items related to:

* **(Requirement 9)** Determining whether adequate measures are in place to protect people and the environment from harmful effects of ionizing radiation.
* **(Item 4.25)** Determining whether adequate measures are in place to control the radiation exposure of workers and members of the public within relevant dose limits and whether protection is optimized so that the magnitude of individual doses, the number of people exposed and the likelihood of exposures being incurred have all been kept as low as reasonably achievable, economic and social factors having been taken into account.

“Requirement 20: Documentation of the safety assessment” subsection contains a CWO item related to:

* **(Item 4.62)** Documentation of the results and findings in the form of a safety report that reflects the complexity of the facility or activity and the radiation risks associated with it. Presentation of the assessments and the analyses that have been carried out for the purposes of demonstrating that the facility or activity is in compliance with the fundamental safety principles and the requirements established in this Safety Requirements publication, and with any other safety requirements as established in national laws and regulations.

“Project or site-specific scope” (POS) items were identified in ten subsections.

“Requirement 4: Purpose of the safety assessment” subsection contains POS items related to:

* **(Item 4.5)** Safety assessment addressing all radiation risks that arise from normal operation (that is, when the facility is operating normally or the activity is being carried out normally) and from anticipated operational occurrences and accident conditions (in which failures or internal or external events have occurred that challenge the safety of the facility or activity).
* **(Item 4.10)** Safety assessment addressing all the radiation risks to individuals and population groups that arise from operation of the facility or conduct of the activity.
* **(Item 4.13)** Including a safety analysis, which consists of a set of different quantitative analyses for evaluating and assessing challenges to safety by means of deterministic and also probabilistic methods, to the safety assessment.

“Specific requirements for safety assessment” subsection contains a POS item related to:

* **(Item 4.16)** The process for safety assessment and verification.

“Requirement 5: Preparation for the safety assessment” subsection contains a POS item related to:

* **(Item 4.18)** Objectives to be ensured with the necessary preparations.

“Requirement 6: Assessment of the possible radiation risks” subsection contains POS items related to:

* **(Requirement 6)** Assessment of the possible radiation risks.
* **(Item 4.19)** Subjects included into the possible radiation risks associated with the facility or activity.

“Requirement 8: Assessment of site characteristics” subsection contains POS items related to:

* **(Requirement 8)** Assessment of site characteristics
* **(Item 4.22)** Subjects to be covered within an assessment of the site characteristics relating to the safety of the facility or activity.
* **(item 4.23)** Considerations about scope and level of details of the site assessment.

“Requirement 9: Assessment of the provisions for radiation protection” subsection contains POS items related to:

* **(Requirement 9)** Determining whether adequate measures are in place to protect people and the environment from harmful effects of ionizing radiation.
* **(Item 4.24)** Determining whether adequate measures are in place to protect people and the environment from harmful effects of ionizing radiation.
* **(Item 4.25)** Determining whether adequate measures are in place to control the radiation exposure of workers and members of the public within relevant dose limits and whether protection is optimized so that the magnitude of individual doses, the number of people exposed and the likelihood of exposures being incurred have all been kept as low as reasonably achievable, economic and social factors having been taken into account.
* **(Item 4.26)** Addressing normal operation of the facility or activity, anticipated operational occurrences and accident conditions in the safety assessment of the provisions for radiation protection.

“Requirement 10: Assessment of engineering aspects” subsection contains POS items related to:

* **(Item 4.31)** Addressing external events in the safety assessment.
* **(Item 4.36A)** Effects of the external events on all facilities and activities at sites with multiple facilities or multiple activities.
* **(Item 4.36B)** Demonstration that forfacilities, on a site that would share resources in accident conditions, the required safety functions can be fulfilled at each facility in accident conditions.
* **(Item 4.37)** Specification of the provisions made for the decommissioning and dismantling of a facility or for the closure of a disposal facility for radioactive waste.

“Requirement 12: Assessment of safety over the lifetime of a facility or activity” subsection contains a POS item related to:

* **(Item 4.42)** Assessment of safety over the lifetime of a facility or activity.

“Requirement 14: Scope of the safety analysis” subsection contains POS items related to:

* **(Item 4.49)** Determining in the safety analysis whether the facility or activity is in compliance with the relevant safety requirements and regulatory requirements.
* **(Item 4.51)** Identification of anticipated operational occurrences and accident conditions that challenge safety in the safety analysis.

“Requirement 20: Documentation of the safety assessment” subsection contains a POS item related to:

* **(Item 4.62)** Documentation of the results and findings in the form of a safety report that reflects the complexity of the facility or activity and the radiation risks associated with it. Presentation of the assessments and the analyses that have been carried out for the purposes of demonstrating that the facility or activity is in compliance with the fundamental safety principles and the requirements established in this Safety Requirements publication, and with any other safety requirements as established in national laws and regulations.

*Refer to* ***Subsection 2.5*** *of the compliance assessment report [37] for more details.*

#### Section 5 (Management, use and maintenance of the safety assessment)

Assessment of **Section 5** of the reference document [7] concludes that the Owner will be responsible to identify and complete the safety assessments to be performed over the lifetime of the plant and identify the process by which the safety assessment in planned, organized, applied, audited, and reviewed.

Plant specific programs for maintenance, surveillance and inspection, operational activities, etc. are in the domain of the operating organization. However, the AP1000 plant DCD [28] safety assessment provides the plant design related bases for the operating organization to establish the plant specific programs and activities as necessary to maintain the plant design basis. The plant specific safety assessment is to be maintained through plant life under the responsibility of the Operator / Licensee.

Plant specific procedures and controls as well as specification of plant staff competences are in the domain of the operating organization. The Owner is also responsible for the management system for the operation of the plant.

Westinghouse has conducted the AP1000 plant design development and safety assessment under its recognized Quality Management System and has taken prime responsibility for safety during the design development. During AP1000 plant project implementation, responsibilities for plant specific leadership and management of safety are transferred systematically to the plant operator.

The AP1000 plant DCD [28] safety assessment provides the plant design related bases for the operating organization to manage facilities and activities safely and to manage regulatory compliance. Additionally, it provides the plant design related bases for establishing and maintaining the plant specific safety assessment through plant life.

POS items were identified in “Requirement 24: Maintenance of the safety assessment” subsection and are related to:

* **(Item 5.8)** Using the results of the safety assessment to make decisions in an integrated, risk informed approach, by means of which the results and insights from the deterministic and probabilistic assessments and any other requirements are combined in making decisions on safety matters in relation to the facility or activity.
* **(Item 5.10)** Periodic review and updating the safety assessment at predefined intervals in accordance with regulatory requirements.

*Refer to* ***Subsection 2.6*** *of the compliance assessment report [37] for more details.*

### Non Compliance

No “Non Compliances” have been identified in the assessment [37].

### Identified Potential Risks To Be Addressed In Bulgaria Project

The analyses of **Section 2** of the compliance assessment report [37] show that the following requirement and sub-requirements related to decommissioning need to be considered in the risk assessment:

* **(Item 1.8)** Stages in the lifetime of a facility or activity for which a safety assessment is carried out, updated and used by the designers, the operating organization and the regulatory body.

**Requirement 5: Preparation for the safety assessment**

* **(Item 4.18 (b))** Requirement to make necessary preparations to ensure that background information relating to the location, design, construction, commissioning, operation, decommissioning and dismantling (or closure) of the facility or activity, as relevant, is available, together with any other evidence that is required to support the safety assessment.

**Requirement 10: Assessment of engineering aspects**

* **(Item 4.37)** Specifying the provisions made for the decommissioning and dismantling of a facility or for the closure of a disposal facility for radioactive waste.

**Requirement 12: Assessment of safety over the lifetime of a facility or activity**

* **(Item 4.42)** A safety assessment is carried out at the design stage for a new facility or activity. Requirement for the safety assessment to cover all the stages in the lifetime of a facility or activity in which there are possible radiation risks. The assessment includes activities that are carried out over a long period of time, such as the decommissioning and dismantling of a facility, the long term storage of radioactive waste, and activities in the post-closure phase of a disposal facility for radioactive waste in significant quantities, and the time at which such activities are conducted (that is, whether they are conducted early or deferred to a later time when radiation levels are lower).

Project specific scope would be performed to support the Owner in a decommissioning plan and safety assessment for a specific site.

## \*SSR-2/1 SAFETY OF NUCLEAR POWER PLANTS: DESIGN

Following summary reflects results of the assessment conducted against reference [88] as part of the compliance assessment report [3738] that is included in Attachment 1.

### High Level Risk Assessment

Conducted compliance assessment [3738] indicates that compliance with the requirements presented in “IAEA Safety Standard No. SSR-2/1 (Rev. 1) – Safety of Nuclear Power Plants: Design” [88] is expected to be demonstrated either via compliance with the requirements “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “IAEA Safety Standard No. SSR-2/1 (Rev. 1) – Safety of Nuclear Power Plants: Design” has been assigned to a “**Low Risk**” category.

The AP1000 plant passive design represents a significant improvement over conventional PWRs and is developed around the fundamental design principles of safety, simplification and standardization. The development of the AP1000 plant safety concept based on passive systems allows full realization of the benefits of these fundamental design principles. The adoption of passive systems as the primary means to deliver safety functions, combined with reliable Defense in Depth (DiD) active systems, achieves both an un-paralleled level of safety and optimized support for investment protection. The active DiD systems are effective in minimizing the demand on the passive systems for more frequent postulated faults, thus ensuring stable and continued production of electricity.

### Roadmap of Items Specifically Discussed in the Safety Requirements Assessment

#### Section 2 (Applying the safety principles and concepts)

#### Requirements discussed in Section 2 of the reference document8 [88] are fully met by Westinghouse.

Several “Project or site-specific scope” (POS) items were identified in two subsections.

“Safety in design” subsection contains a POS item related to:

#### (Item 2.11) Plant event sequences that could result in high radiation doses or large radioactive releases, plant event sequences with a significant frequency of occurrence and off-site protective measures.

“The concept of defence in depth” subsection a POS item related to:

* **(Item 2.13)** Levels of Defense in depth.

*Refer to* ***Section 6*** *of the compliance assessment report [38] for more details.*

#### Section 3 (Management of safety in design)

Requirements discussed in **Section 3** of the reference document8 [88] are fully met by Westinghouse.

Ensuring safety of the plant design throughout the lifetime of the plant is responsibility of the Owner/Operator. The plant designer serves as a resource to the Owner/Operator over the plant life.

One POS item was identified under Requirement 1 and is related to:

* **(Requirement 1)** Responsibilities in the management of safety in plant design. An applicant for a license to construct and/or operate a nuclear power plant shall be responsible for ensuring that the design submitted to the regulatory body meets all applicable safety requirements.

*Refer to* ***Section 6*** *of the compliance assessment report [38] for more details.*

#### Section 4 (Principal technical requirements)

#### All the requirements presented in Section 4 of the reference document8 [88] are fully met by Westinghouse.

#### Section 5 (General plant design)

Requirements discussed in **Section 5** of the reference document8 [88] are fully met by Westinghouse.

Several POS items were identified in five subsections.

“Requirement 17: internal and external hazards” subsection contains POS items related to:

* **(Requirement 17)** Internal and external hazards
* **(Item 5.15B)** Impact of specific hazards on multiple unit plant sites. Even when AP1000 is built as a standalone unit, interaction needs to be studied.
* **(Item 5.17)** Consideration of natural and human induced external events in the design and dependence on off-site services

#### (Item 5.21) Adequate margin to protect items important to safety against levels of hazards considered for design, derived from the hazard evaluation for the site and avoiding cliff edge effects.

“Requirement 28: Operational limits and conditions for safe operation” subsection contains a POS item related to:

* **(Item 5.44)** Subjects to be included in the requirements and operational limits and conditions established in the design for the nuclear power plant. Since Westinghouse provides standard Technical Specifications that shall be used by the plant operator to develop its own plant specific Technical Specifications.

“Requirement 32: Design for optimal operator performance” subsection contains a POS item related to:

* **(Item 5.53)** Number of operating personnel required to perform all the simultaneous operations necessary to bring the plant into a safe state.

**“**Requirement 35: Nuclear power plants used for cogeneration of heat and power, heat generation or desalination.**”** subsection contains a POS item related to:

* **(Requirement 35)** Nuclear power plants used for cogeneration of heat and power, heat generation or desalination. This is not considered within the standard AP1000 plant design. However, this is a well understood requirement for such eventual site specific adaptations.

**“**Requirement 38: Control of access to the plant” subsection contains a POS item related to:

* **(Item 5.68)** Provision to be made in the design of the buildings and the layout of the site for the control of access to the nuclear power plant by operating personnel and/or for equipment.

*Refer to* ***Section 6*** *of the compliance assessment report [38] for more details.*

#### Section 6 (Design of specific plant systems)

Requirements discussed in **Section 6** of the reference document8 [88] are met by Westinghouse either “as stated” or via compliance with their objectives.

“Compliant with objective” (CWO) items were identified two subsections.

“Requirement 68: Design for withstanding the loss of off-site power” subsection contains a CWO item related to:

* **(Item 6.43)** Requirements for emergency power supply and for the alternate power source at the nuclear power plant. Offsite power requirements can be relaxed due to the passive design of the AP1000 plant. AC power does not have a safety functions. IDS battery powered DC emergency power supply system for safety functions after postulated initiating events has sufficient capacity to achieve and maintain safe shutdown of the plant for 72 hours following a complete loss of all AC power sources

“Requirement 71: Process sampling systems and post-accident sampling systems” subsection contains a CWO item related to:

* **(Requirement 71)** Provision of Process sampling systems and post-accident sampling systems.

*Refer to* ***Section 6*** *of the compliance assessment report [38] for more details.*

### Non Compliance

No “Non Compliances” have been identified in the assessment [38].

### Identified Potential Risks To Be Addressed In Bulgaria Project

Potential risks are associated with

**Requirement 71** the CWO for the primary sampling system (PSS), regarding its sampling capabilities, might request additional discussion. For the time being we are assuming it will be validated as is thus we maintain “**Low Risk**” for this regulation

Also, **Requirement 35** on “Nuclear power plants used for cogeneration of heat and power, heat generation or desalination”, since it is not considered in the standard design. If it were requested to provide Heat/District Heating by KNPP-NB additional scope will need to be added to the project, this is scope is considered feasible and well understood, but notwithstanding the previous according to document classification on section 1.3 this will become a High Risk since this will incur a modification on design, however this shall be less impacting since Turbine Block Design is still not defined. Since this is not yet a project requirement, thus we keep this No. SSR-2/1 Specific Safety Requirements as “**Low Risk**”.

## SSG-30 SAFETY CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS IN NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [9] as part of the compliance assessment report [39] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [39] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-30: Safety Classification of Structures Systems and Components in Nuclear Power Plant” [9] is expected to be demonstrated either via compliance with the guidelines “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “IAEA Specific Safety Guide No. SSG-30: Safety Classification of Structures Systems and Components in Nuclear Power Plant” has been assigned to a “**Low Risk**” category.

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### Section 2 (General approach)

The general approach to provide a structure and method for identifying and classifying SSCs important to safety recommended by IAEA in **Section 2** of the reference document [9] is consistent with the approach defined in the AP1000 plant documentation.

Several “Compliant with objective” (CWO) items were identified in two subsections.

“Basic requirements” subsection contains a CWO item related to:

* **(Item 2.2/Requirement 27)** Classification of support service systems that ensure the operability of equipment forming part of a system important to safety.

“Outline of the safety classification process” subsection contains several CWO items related to:

* **(Item 2.12)** Factors taken into account for safety functions categorization.
* **(Item 2.13)** Possibility for the SSCs implemented as design provisions to be assigned directly to a safety class without the need for a further analysis of safety function categories.
* **(Item 2.17)** Possibility of using other approaches utilizing a larger or smaller number of categories and classes.

*Refer to* ***Subsection 2.2.1*** *of the compliance assessment report [39] for more details.*

#### Section 3 (Safety classification process)

The safety classification process identified in the AP1000 documentation differs from the process defined by the IAEA in **Section 3** of the reference document [9]. However, the result of the classification process defined for the AP1000 plant is equivalent to the classification output according to IAEA recommendations.

CWO items were identified in four subsections.

“Identification of functions to be categorized” subsection contains CWO items related to:

* **(Item 3.3)** Functions to be categorized.
* **(Item 3.4)** Identification of functions that need to be categorized with respect to each plant state separately.
* **(Item 3.7)** Including of functions credited in the safety analysis with either preventing some sequences resulting from additional independent failures from escalating to a severe accident, or mitigating the consequences of a severe accident in functions associated with design extension conditions.

“Identification of design provisions” subsection contains a CWO item related to:

* **(Item 3.9)** Subjects that have to be included in design provisions.

“Categorization of functions” subsection contains CWO items related to:

* **(Item 3.10)** Categorization of the functions required for fulfilling the main safety functions in all plant states, including modes of normal operation, the basis of their safety significance.
* **(Item 3.11)** Definitions of three levels of severity.
* **(Item 3.12)** Definition of factor 2 (frequency of occurrence).
* **(Item 3.16)** Categorization of a function where it could be considered to be in more than one category (e.g. because the function is needed for more than one postulated initiating event).

“Classification of structures, systems and components” subsection contains CWO items related to:

* **(Item 3.17)** Assigning SSC to the safety class basing on the functions they perform and safety categories of those functions.
* **(Item 3.18)** Identification and classification of all SSCs required to perform a function that is safety categorized.
* **(Item 3.19)** Initial assigning SSCs (including supporting SSCs) that are designed to carry out identified functions to the safety class corresponding to the safety category of the function to which they belong by applying factors “a” (The safety function(s) to be performed by the item) and “c” (The frequency with which the item will be called upon to perform a safety function).
* **(Item 3.20)** Amendment of the initial classification, as necessary, to take into account factors “b” (The consequences of failure to perform a safety function) and “d” (The time following a postulated initiating event at which, or the period for which, the item will be called upon to perform a safety function).
* **(Item 3.21)** Classification of an SSC that contributes to the performance of several functions of different categories.
* **(Item 3.22)** Selection of final safety class of SSC by applying relevant considerations.
* **(Item 3.24)** Classification of any SSC that does not contribute to any categorized function, but whose failure could adversely affect a categorized function (if this cannot be precluded by design).

*Refer to* ***Subsection 2.2.2*** *of the compliance assessment report [39] for more details.*

#### Section 4 (Selection of applicable engineering design rules for structures, systems and components)

The selection of applicable engineering design rules for SSCs recommended by IAEA in **Section 4** of the reference document [9] is consistent with the approach considered in the AP1000 design.

All the relevant guidelines are fulfilled “as stated”.

Provision and justification of the correspondence between the safety class and the associated engineering design and manufacturing rules, including the codes and/or standards that apply to each SSC is responsibility of the Owner.

### Non Compliance

No “Non Compliances” have been identified in the assessment [39].

## SSG-34 DESIGN OF ELECTRICAL POWER SYSTEMS FOR NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [10] as part of the compliance assessment report [40] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [40] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-34: Design of Electrical Power Systems for Nuclear Power Plants” [10] is expected to be demonstrated but will require additional evaluations to be performed. Since no non-compliances have been identified and no design changes are anticipated, “IAEA Specific Safety Guide No. SSG-34: Design of Electrical Power Systems for Nuclear Power Plants” has been conservatively assigned to a “**Medium Risk**” category.

“Medium risk” has been assigned as a result of an additional scope needed to fully comply with **(Item 5.141/187)**, **(Item 5.142/188)** and **(Item 5.144/190)** which are described in **Subsection 2.10.2.4** of this report. The required scope includes evaluation of the extent of internal surge protection and extent of the internal cable routing.

***Note****: first number represents number of a guideline in the reference document [10] and second number represents a number of the same guideline in the compliance assessment report [40]. I.e. Item 5.141 in SSG-30 [10] is the 187 assessed item in APP-GW-G0R-006 [40].*

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### Section 2 (Electrical power systems at nuclear power plants)

Guidelines in **Section 2** of the reference document [10] that are applicable to AP1000 design are met by Westinghouse.

One “Compliant with objective” (CWO) item was identified in “Design considerations imposed by requirements for nuclear safety” subsection and is related to:

* **(Item 2.19/6)** Requirements to robust systems.

*Refer to* ***Subsection 2.2.2*** *of the compliance assessment report [40] for more details.*

#### Section 3 (Classification of electrical power systems)

All the guidelines in **Section 3** of the reference document [10] are met by Westinghouse “as stated”.

#### Section 4 (Design bases for electrical power systems)

All the guidelines in **Section 4** of the reference document [10] are met by Westinghouse “as stated”.

Owner is responsible for confirmation of the design bases when major replacements and major modifications of the electrical power system (on-site or off-site) as well as changes in loading are made. Owner is also responsible for aperiodic cumulative evaluation, for example as part of periodic safety reviews.

#### Section 5 (General design guidelines for electrical power systems)

Guidelines discussed in **Section 5** of the reference document [10] are met by Westinghouse either “as stated” or via compliance with their objectives.

The electrical fault clearing time should be defined by the grid operator and falls under responsibility of the Owner and/or Licensee.

Owner or/and Licensee is responsible for including activities to identify any trend towards degradation (ageing) that could cause equipment to become incapable of performing its safety function in maintenance programs, surveillance programs and ageing management programs.

In addition, requirements addressing test program fall under the responsibility of the Owner and/or Licensee.

Finally, the Owner or/and Licensee is responsible that provisions for removing electrical equipment from service ensure the equipment is properly isolated in order to protect personnel and to avoid spurious operation.

CWO items were identified in four subsections.

“Grounding practices” subsection contains CWO items related to:

* **(Item 5.132/178)** High impedance grounding of medium voltage AC power systems.
* **(Item 5.136/182)** Necessity for detection of low impedance to ground to only alarm and still allow the equipment to perform its function.

“Lightning and surge protection” subsection contains CWO items related to:

* **(Item 5.141/187)** Including shielding and surge arresters to internal lightning protection to protect against both the induced high voltage caused by the lightning current and high transferred voltage.
* **(Item 5.142/188)** Location considerations for safety classified raceways and cables.
* **(Item 5.144/190)** Connection of the internal protection grounding to rest of the lightning grounding.
* **(Item 5.148/194)** Ensuring that electrical noise and voltage perturbations generated by equipment in ancillary buildings do not adversely affect the plant power systems (If plant buses are used to supply power to ancillary buildings).

“Design to cope with ageing” subsection contains a CWO item related to:

* **(Item 5.216/261)** Determining the qualified life of component as a result of ongoing component qualification.

“Safety related standby AC power sources” subsection contains a CWO item related to:

* **(Item 5.293/338)** Possession of safety related standby AC power sources to provide reliable power for defence in depth functions that supplement the safety systems and reduce the challenges to them.

Several “Project or site-specific scope” (POS) items were identified in three subsections.

“Design for reliability” subsection contains a POS item related to:

**(Item 5.41/87)** Requirements to the lower class circuit (associated circuit) for cases when it is impractical to provide adequate separation and isolation from electrical faults between a safety circuit and a circuit of a lower class function.

“Lightning and surge protection” subsection contains POS items related to:

* **(Item 5.141/187)** Including shielding and surge arresters to internal lightning protection to protect against both the induced high voltage caused by the lightning current and high transferred voltage. In AP1000 Design Internal surge protection is provided for some power circuits. The extent of protection needs further evaluation
* **(Item 5.142/188)** Location considerations for safety classified raceways and cables. In AP1000 design The Class 1 safety power system equipment is located internally in the auxiliary building. A portion of the cabling is routed internally. The extent of internal routing needs further evaluation.
* **(Item 5.144/190)** Connection of the internal protection grounding to rest of the lightning grounding. The extent of protection needs further evaluation

“Equipment qualification” subsection contains POS items related to:

* **(Item 5.187/233)** Design and installation of equipment and systems important to safety to withstand the electromagnetic conditions in the environments in which they are located.
* **(Item 5.195/240)** Types of electromagnetic interference that should be considered in the design of electrical power systems and components.

*Refer to* ***Subsection 2.2.5*** *of the compliance assessment report [40] for more details.*

#### Section 6 (Design guidelines for preferred power supplies)

Guidelines discussed in **Section 6** of the reference document [10] are met by Westinghouse either “as stated” or via compliance with their objectives.

Switchyard design is responsibility of the switchyard owner.

Guidelines introduced in “Grid stability and reliability”, “Interface and interaction between transmission system operator and nuclear power plant operating organization” and “Assessment of the reliability of grid connections” subsections fall under responsibility of the Owner and/or Licensee.

CWO items were identified in three subsections.

“Reliability of protective devices and high voltage equipment” subsection contains a CWO item related to:

* **(Item 6.8/348)** Events to be considered in the design of the grid connection and the relay protection.

‘Off-site power supplies” subsection contains CWO items related to:

* **(Item 6.16/356)** Higher forced outage rate for nuclear power plants with a single transmission line.
* **(Item 6.19/359)** Connection of each unit at multi-unit plants to two off-site power supplies such that the technical safety objectives are fulfilled simultaneously for all units. In AP1000 Off-site power supply is not required for safe shutdown, therefore one off-site power source could be sufficient. This is also related to POS Item **(Item 6.20/360).**

“Switchyard” subsection contains a CWO item related to:

* **(Item 6.40/380)** Requirements to the switchyard control power.

*Refer to* ***Subsection 2.2.6*** *of the compliance assessment report [40] for more details.*

#### Section 7 (Design guidelines for electrical safety power systems)

Guidelines discussed in **Section 7** of the reference document [10] are met by Westinghouse either “as stated” or via compliance with their objectives.

CWO items were identified in three subsections.

“General” subsection contains CWO items related to:

* **(Item 7.5/408)** Restricting the magnitude of variations in voltage and frequency from affecting equipment that is starting, already sequenced or operating.
* **(Item 7.16/419)** Recommendations that apply to bus voltage and frequency monitoring and protection schemes for protection against degradation in voltage, degradation in frequency or loss of voltage. Note that The AP1000 electrical system design does not require safety-classified AC power to achieve safe shutdown.

“Design for reliability” subsection contains a CWO item related to:

* **(Item 7.25/428)** Combination of conditions under which, each safety group shall perform all actions required.

**The AP1000 passive plant design does not require the use of safety standby AC power sources.** Therefore, entire “Safety standby AC power sources” subsection has been assigned to CWO category.

“DC power systems” subsection contains CWO items related to:

* **(Item 7.84/486)** Preference of having two battery chargers and two parallel batteries in order to have more flexibility for maintenance.
* **(Item 7.90/492)** Requirements for cases if forced ventilation is necessary. Forced Ventilation is not credited for Batteries for Design Basis Events, cooling is provided by passive means.
* **(Item 7.102/504)** Ability of battery chargers to supply the loads without any battery connected.
* **(Item 7.124/526)** Preference for the inverter not to not have an overvoltage protection on the DC side.

*Refer to* ***Subsection 2.2.7*** *of the compliance assessment report [40] for more details.*

#### Section 8 (Alternate ac power supplies)

**Section 8** of the reference document [10] at large does not apply. Standby AC power generation is included in the AP1000 design for additional defense and investment protection, but provides no safety function.

#### Section 9 (Confirmation and documentation of the design)

Guidelines discussed in **Section 9** of the reference document [10] are met by Westinghouse either “as stated” or via compliance with their objectives.

CWO items were identified in “Verification” subsection and are related to:

* **(Item 9.3/550)** Demonstrations to be performed as part of the design and design verification.
* **(Item 9.5/552)** Performing the demonstration of the reliability and availability of the off-site circuits together with the transmission system operator. In AP1000 No off-site circuits are required to provide power to safety systems.
* **(Item 9.9/556)** Consideration of test facilities (that are part of the safety system) in determining the availability of systems.

*Refer to* ***Subsection 2.2.9*** *of the compliance assessment report [40] for more details.*

### Non Compliance

No “Non Compliances” have been identified in the assessment [40].

## SSG-39 DESIGN OF INSTRUMENTATION AND CONTROL SYSTEMS FOR NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [11] as part of the compliance assessment report [41] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [41] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-39: Design of Instrumentation and Control Systems for Nuclear Power Plants” [11] is expected to be demonstrated but might require additional evaluations/analyses to be performed. Since identified non-compliances and other items are not expected to result in design changes, “IAEA Specific Safety Guide No. SSG-39: Design of Instrumentation and Control Systems for Nuclear Power Plants” has been assigned to a “**Medium Risk**” category.

“**Medium Risk**” category was assigned due to the guidelines addressing “Common cause failures” **(Items 4.31, 4.32, 4.34, 6.14, 6.68, 7.6, 7.7, 7.95-7.99)** and “Decommissioning” **(Item 2.19)**.

Those guidelines, identified risks associated with them and additional topics for further consideration are covered in **Subsection 3.2.1.30** of this report.

### Roadmap of Items Specifically Discussed in the Guide Assessment

Due to the big number of identified “Compliant with objective” (CWO) items, they are presented in a form of tables. First row of each table contains title of the subsection from the reference document [11] where such items were identified, while subsequent rows contain numbers of individual items or ranges of items among which CWO category has been assigned.

#### Section 2 (The management system for instrumentation and control design)

Overall, these topical items reflect the Westinghouse’s organizational culture and policies and procedures per its high-level Quality Management System (QMS). However, some exceptions were taken regarding lifecycle models and development phases because they were not applicable to Westinghouse’s approach for several of the specified guidelines.

“Compliant with objective” (CWO) items were identified among the following guidelines in **Section 2** of the reference document [11]:

|  |  |  |  |
| --- | --- | --- | --- |
| **“General”** | **“Use of life cycle models”** | **“Activities common to all life cycle phases”** | **“Life cycle activities”** |
| (Item 2.2) | (Item 2.13 – 2.19) | (Item 2.39 – 2.45) | (Item 2.92 – 2.110) |
| (Item 2.3) | (Item 2.24 – 2.37) | (Item 2.47 – 2.70) | (Item 2.118 – 2.156) |
| (Item 2.4) |  | (Item 2.75 – 2.91) | (Item 2.158) |

One “project or site-specific” item was identified in “General” subsection and is connected to **(Item 2.9)**.

**(Items 2.159-167)** in “Lifecycle activities” subsection were identified to belong to “project or site-specific” (POS) category as well.

*Refer to* ***Subsection 2.2.1*** *of the compliance assessment report [41] for more details.*

#### Section 3 (Design basis for instrumentation and control systems)

The general design principles for design basis for safety functions and functions important to safety were met for the most part, however, some topical design items were general statements or recommendations for which the AP1000 plant design does not include specific design provisions.

CWO items were identified among the following guidelines in **Section 3** of the reference document [11]:

|  |  |
| --- | --- |
| **“Identification of instrumentation and control functions”** | **“Content of design basis for instrumentation and control systems”** |
| (Item 3.4) | (Items 3.13 – 3.14) |
|  | (Item 3.16) |

*Refer to* ***Subsection 2.2.2*** *of the compliance assessment report [41] for more details.*

#### Section 4 (Instrumentation and control architecture)

The AP1000 plant design for I&C systems and platforms met the guidelines and recommendations of this SSG-39. For example, the concept of defense-in-depth permeates through the AP1000 I&C design principles and specific references and discussions were provided. Likewise, for other principles such as redundancy and functional independence and separation of systems, and others.

However, some technical details pertaining to how the AP1000 implements some of these principles in practice, were not directly applicable to the stipulations on the SSG-39.

CWO items were identified among the following guidelines in **Section 4** of the reference document [11]:

|  |  |  |  |  |
| --- | --- | --- | --- | --- |
| **“Architectural design”** | **“Content of the overall instrumentation and control architecture”** | **“Content of individual instrumentation and control system architectures”** | **“Independence”** | **“Consideration of common cause failure”** |
| (Item 4.4) | (Items 4.11 – 4.12) | (Item 4.13) | (Items 4.20 – 4.24) | (Items 4.27 – 4.30) |
| (Items 4.7 – 4.10) |  |  |  | (Items 4.32 – 4.33) |
|  |  |  |  | (Item 4.38) |
|  |  |  |  | (Item 4.40) |

*Refer to* ***Subsection 2.2.3*** *of the compliance assessment report [41] for more details.*

#### Section 5 (Safety classification of instrumentation and control functions, systems and equipment)

The classification of systems, functions and components were evaluated per the design principles and approaches stipulated in the SSG-39 against the equivalent design provisions of the AP1000 plant design. And the intent of such principles was met by the latter. Discussions on the safety significance of SSCs pertaining to I&C systems and functions were included, as justification for how the AP1000 design met such design provisions—for safety related and important to safety functions.

**Section 5** contains only three CWO items, names **Item 5.3**, **Item 5.5** and **Item 5.10**.

*Refer to* ***Subsection 2.2.4*** *of the compliance assessment report [41] for more details.*

#### Section 6 (General recommendations for all instrumentation and control systems important to safety)

Topical designs regarding complexity of I&C systems, their reliability and testability and maintainability and Single Failure criterion and Redundancy and Independence and, again, physical separation between safety systems and those important to safety, were evaluated and for the most part the AP1000 plant met such criteria and principles. However, the underlying mechanisms used in the development of I&C systems in AP1000 differ to the stated recommendations and guidelines of the SSG-39.

CWO items were identified among the following guidelines in **Section 6** of the reference document [11]:

|  |  |  |  |  |
| --- | --- | --- | --- | --- |
| **“General”** | **“Design for reliability”** | **“Equipment qualification”** | **“Design to cope with ageing and obsolescence”** | **“Control of access to systems important to safety”** |
| (Items 6.4 – 6.5) | (Item 6.6) | (Items 6.78 – 6.134) | (Items 6.135 – 6.152) | (Items 6.154 – 6.158) |
|  | (Item 6.9) |  |  |  |
|  | (Items 6.12 – 6.14) |  |  |  |
|  | (Item 6.16) |  |  |  |
|  | (Items 6.18 – 6.19) |  |  |  |
|  | (Items 6.25 – 6.34) |  |  |  |
|  | (Item 6.43) |  |  |  |
|  | (Items 6.46 – 6.52) |  |  |  |
|  | (Items 6.54 – 6.55) |  |  |  |
|  | (Item 6.66) |  |  |  |
|  | (Items 6.69 – 6.76) |  |  |  |

CWO items were identified among the following guidelines in **Section 6** of the reference document [11]:

|  |  |  |  |
| --- | --- | --- | --- |
| **“Testing and testability during operation”** | **“Provisions for removal from service for testing or maintenance”** | **“Set points”** | **“Marking and identification of items important to safety”** |
| (Items 6.166 – 6.172) | (Items 6.198 – 6.200) | (Items 6.206 – 6.212) | (Items 6.213 – 6.219) |
| (Item 6.180) |  |  |  |
| (Item 6.183 – 6.188) |  |  |  |

*Refer to* ***Subsection 2.2.5*** *of the compliance assessment report [41] for more details.*

#### Section 7 (Design guidelines for specific instrumentation and control systems, and equipment)

The design principles from previous sections were used as building design blocks for further discuss and estipulate design guidelines for control systems. A key distinction was made by this SSG-39 regarding the interactions between the automatic safety actions and manual safety actions and between safety systems and non-safety systems, in general. The concept of Spurious initiation was also discussed and WEC determine how the I&C platforms ensures that such spurious actions are mitigated and addressed in the AP1000 design.

CWO items were identified among the following guidelines in **Section 7** of the reference document [11]:

|  |  |  |
| --- | --- | --- |
| **“Sensing devices”** | **“Control systems”** | **“Protection system”** |
| (Items 7.1-7.9) | (Item 7.14) | (Items 7.18 – 7.59) |

CWO items were identified among the following guidelines in **Section 7** of the reference document [11]:

|  |  |  |
| --- | --- | --- |
| **“Power supplies”** | **“Digital systems”** | **“Software tools”** |
| (Items 7.60 – 7.65) | (Items 7.66 – 7.69) | (Items 150 – 164) |
|  | (Items 7.71 – 7.130) |  |

**(Items 7.131 – 147)** in “Digital Systems” subsection were identified to belong to “project or site-specific” (POS) category.

*Refer to* ***Subsection 2.2.6*** *of the compliance assessment report [41] for more details.*

#### Section 8 (Considerations relating to the human–machine interface)

In depth discussions were provided from the perspective of the plant personnel, the operators and technical staff, regarding how HFE principles are incorporated into the AP1000 design for I&C system, from the ground up. This bottom-up approach of the PMS and PLS and DAS and HSI (DDS, OCS), are reflected in the capabilities of the MCR and Remote Shutdown Room (supplementary control room), to enable the operators to monitor the plant during ALL modes of operations and to take actions, when needed, to ensure the safety of the plant and mitigate operational risks. Of course, key features of the AP1000 design regarding the passive systems and automatic actuation afford the operators with great flexibility and simplicity on how to monitor and control the plant daily and safely.

Discussions of displays in the MCR and RSR and Alarms and accident monitoring systems were evaluated against the SSG-39 and the work environment and the plant digital system and communication systems that make reliable and consistent MCR operations possible. For this section, exceptions were also made where the AP1000 design met the intent and spirit of the SSG-39, but the approach to develop and implement such design provisions and control measures were different than the latter.

CWO items were identified among the following guidelines in **Section 8** of the reference document [11]:

|  |  |  |
| --- | --- | --- |
| **“Control rooms** | **“Accident monitoring”** | **“General principles relating to human factors engineering for instrumentation and control systems”** |
| (Items 8.7 – 8.8) | (Items 8.23 – 8.27) | (Items 8.57 – 8.69) |
| (Items 8.14 – 8.18) | (Items 8.32 – 8.35) |  |

*Refer to* ***Subsection 2.2.7*** *of the compliance assessment report [41] for more details.*

#### Section 9 (Software)

This section included many of the design principles and recommendations and guidelines of the other sections of the SSG-39. The AP1000 design for I&C systems incorporate many of such recommendations and principles. However, because the unique characteristics of software tools and features deployed in the I&C platforms (PMS, PLS, DAS) and DDS/OCS to support software development in the respective platforms, several of the SSG-39 recommendations are not in alignment, as written, and therefore, exceptions for such discrepancies between the AP1000 design and the former were recorded in the compliance assessment report [41].

CWO items were identified among the following guidelines in **Section 9** of the reference document [11]:

|  |  |  |  |
| --- | --- | --- | --- |
| **“General”** | **“Software requirements”** | **“Software design”** | **“Software implementation”** |
| (Items 9.2 – 9.3) | (Items 9.6 – 9.15) | (Items 9.16 – 9.43) | (Items 9.44 – 9.63) |

CWO items were identified among the following guidelines in **Section 9** of the reference document [11]:

|  |  |  |  |
| --- | --- | --- | --- |
| **“Software verification and analysis”** | **“Predeveloped software”** | **“Software tools”** | **“Third party assessment”** |
| (Items 9.64 – 9.95) | (Items 9.96 – 9.98) | (Item 9.99) | (Items 9.100 – 9.103) |

*Refer to* ***Subsection 2.2.8*** *of the compliance assessment report [41] for more details.*

### Non Compliance

Two “Non-Compliance” items were identified in **Section 5** (Safety classification of instrumentation and control functions, systems and equipment) and are related to:

* **(Item 5.11)** Guideline stating that the I&C systems and components performing each function assigned in a safety category should be identified and classified and that they should be primarily classified according to the category assigned to the function that they perform.
* **(Item 5.13)** Guideline stating that In SSG-30 [9], three safety categories for functions and three safety classes for structures, systems and components are recommended, based on the experience of Member States. However, a larger or smaller number of categories and classes may be used, provided that they are aligned with the guidance provided in paras 2.12 and 2.15 of SSG-30 [9].

Those two items were assigned to the “Non-compliance” category basing on the data from “Revision A” of the SSG-30 compliance assessment report.

In the “Revision B” of the SSG-30 compliance assessment report [39], compliance with the IAEA guidelines for safety classification has been reassessed. Basing on the updated inputs from the “Revision B” of the SSG-30 compliance assessment report [39], Item 5.11 and 5.13 will be reassigned to CWO category as they comply with objectives of the IAEA guidelines.

This change of the compliance category for Item 5.11 and 5.13 will be reflected in the “Revision B” of SSG-39 compliance assessment report [41].

One “Non-Compliance” item was identified in “Design for reliability” subsection of **Section 6** (General recommendations for all instrumentation and control systems important to safety) and is related to:

* **(Item 6.68)** Guideline stating that the non-systematic failure modes of I&C components and systems should be known and documented.

This requirement has been considered for AP1000 safety-related system (PMS), failure modes and effects analysis was performed, see DCD Subsection 7.2.3.

Therefore, there might potentially be a need for improved analysis for other important to safety systems.

## SSG-52 DESIGN OF THE REACTOR CORE FOR NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [12] as part of the compliance assessment report [42] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [42] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-52: Design of the Reactor Core for Nuclear Power Plants” [12] is expected to be demonstrated either via compliance with the guidelines “as stated” or via compliance with their objectives. Since no non-compliances have been identified and no design changes or additional design analyses are anticipated, “IAEA Specific Safety Guide No. SSG-52: Design of the Reactor Core for Nuclear Power Plants” has been assigned to a “**Low Risk**” category.

The AP1000PWR approach to design of the Reactor Core is aligned with the guidance provided in the IAEA guidelines for factors such as design limits, internal and external events influence on design bases, reliability, possible load combinations, manufacturing, installation, testing, maintenance, and other aspects listed in the specific safety guide itself. A small amount of SSRs are not applicable to the AP1000PWR design since they describe requirements for design of NPPs using BWR and PHWR technology. Because of the nature of this particular IAEA SSG which focuses purely on design activities, fulfillment of all of the requirements is the responsibility of Westinghouse as a design provider.

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### Section 2 (General safety considerations in the design of the reactor core)

Westinghouse demonstrates compliance with the guidelines discussed in **Section 2** of the reference document [12].

Several “Compliant with objective” (CWO) items were identified in two subsections.

“Management system” subsection contains a CWO item related to:

* **(Item 2.1)** IAEA recommendations that the design of reactor core has to take into account.

“Design basis for structures, systems and components of the reactor core” subsection contains CWO items related to:

* **(Item 2.15)** Safety classification of fuel rods and fuel assemblies.
* **(Item 2.16)** Safety classification of control rods.

*Refer to* ***Subsection 2.3*** *of the compliance assessment report [42] for more details.*

#### Section 3 (Specific safety considerations in the design of the reactor core)

All the guidelines applicable to the AP1000 Standard Design in **Section 3** of the reference document [12] are met by Westinghouse.

Rest of the guidelines is responsibility of the Owner/Licensee or is not applicable AP1000 Standard Design.

#### Section 4 (Qualification and testing)

All the guidelines introduced in **Section 4** of the reference document [12] are fully met by Westinghouse.

### Non Compliance

No “Non Compliances” have been identified in the assessment [42].

## SSG-53 DESIGN OF THE REACTOR CONTAINMENT AND ASSOCIATED SYSTEMS FOR NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [13] as part of the compliance assessment report [43] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [43] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-53: Design of the Reactor Containment and Associated Systems for Nuclear Power Plants” [13] is expected to be demonstrated either via compliance with the guidelines “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “IAEA Specific Safety Guide No. SSG-53: Design of the Reactor Containment and Associated Systems for Nuclear Power Plants” has been assigned to a “**Low Risk**” category.

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### Section 2 (Containment safety functions and the design approach)

#### The AP1000 design meets guidelines addressed in Section 2 of the reference document [13] are met by Westinghouse “as stated”.

#### Section 3 (Design basis of the containment structure and its components and associated systems)

#### The AP1000 plant design meets most of the guidelines addressed in Section 3 of the reference document [13] “as stated”.

“Compliant with objective” (CWO) items were identified in five subsections.

“Internal hazards” subsection contains a CWO item related to:

* **(Item 3.12)** Adequate margins in the design methods and construction codes used to avoid cliff edge effects in the event of a slight increase in the severity of the internal hazards. Cliff edge effects has been also assessed in different assessments as COM/CWO, and low risk see as an example assessment in article 87 of [33].

“Accident conditions” subsection contains CWO items related to:

* **(Item 3.33)** Identification of relevant design extension conditions and using them to establish the design bases of the containment and its associated systems necessary to meet the objectives established for that category of accident.
* **(Item 3.36)** Three types of failure to be considered for design extension conditions without significant fuel degradation.

“Design limits” subsection contains a CWO item related to:

* **(Item 3.48)** Establishing a set of primary design limits for the containment and its associated systems as a means of ensuring the overall safety functions of the containment in all operational states and in accident conditions. Terms in which primary design limits are expressed.

“Safety classification” subsection contains CWO items related to:

* **(Item 3.73)** Establishing the safety classification in a consistent manner such that all systems necessary for the accomplishment of one safety function, including the associated support service systems, are assigned to the same safety class.
* **(Item 3.75)** Considerations regarding safety classes.

“Codes and standards” subsection contains a CWO item related to:

* **(Item 3.86)** Required attributes of the selected codes and standards.

For the guidelines related to cliff-edge-effects, responses are set as compliance with objective “CWO” since the AP1000 plant design uses a different but equivalent terminology for internal hazards.

For the guidelines related to design extension conditions without core melt, responses are set as compliance with objective “CWO” since the AP1000 plant design uses a different but equivalent terminology for design extension conditions.

For the guidelines related to safety classification, the response is set as compliance with objective “CWO” since the AP1000 plant design uses a different but equivalent approach for safety classification.

For the guidelines related to newest codes and standards, the response is set as compliance with objective “CWO” since the AP1000 plant design meets with the codes when it is acquired the NRC design certification. This is a more conservative approach that doesn’t compromises any safety since the newest codes include more best estimate guidance.

One “Project or site-specific scope” (POS) item was identified in “External hazards” subsection and is related to:

* **(Item 3.14)** Adaptation or supplementation of external hazards list to include site specific hazards.

*Refer to* ***Subsection 2.2.2*** *of the compliance assessment report [43] for more details.*

#### Section 4 (Design of the containment and its associated systems)

The AP1000 design meets most of the guidelines addressed in **Section 4** of the reference document [13] “as stated”.

CWO items were identified in four subsections.

“Structural design of containment” subsection contains CWO items related to:

#### (Item 4.38) Levels to be considered for the design of leaktightness.

#### (Item 4.40) Using an adequate statistical combination of the loads resulting from an SL-2 earthquake and design basis accidents in order to provide margins.

“Mechanical features of the containment” subsection contains CWO items related to:

#### (Item 4.157) Necessity for small dead-ended instrumentation lines that penetrate the containment to have at least one isolation valve outside the containment.

#### (Item 4.158) Non-necessity of containment isolation valves for instrumentation lines that are closed, provided that the lines are designed to withstand the accident conditions for which confinement is necessary. Equipping the rooms where this lines emerge with a filtration–ventilation system to maintain subatmospheric pressure. Ability of such rooms and equipment within them to withstand increased levels of temperature and humidity.

#### (Item 4.171) Basis for the means for ensuring the leaktightness of penetrations through the containment for electrical power cables and instrument cables.

“Materials” subsection contains CWO items related to:

#### (Item 4.196) Objectives to be achieved by selected materials used to insulate pipes and tanks inside the containment.

#### (Item 4.198) A cleaning system for the filters that should be installed, taking into account the large uncertainties about the types and amount of debris that could clog the filters.

“Instrumentation” subsection contains CWO items related to:

#### (Item 4.208) Monitoring of deformation or movement of the containment structures or the containment walls throughout the lifetime of the containment.

#### (Item 4.228) Consideration of the measurement and analysis of audio signals from the containment for the detection of abnormalities.

#### (Item 4.230) Incorporation of the appropriate instrumentation for conducting periodic leak tests inside the containment. Combination of measurements of temperature, pressure, and humidity, as well as flow rates for the periodic calculation of the mass of the containment atmosphere and for the estimation of the leak rate. Measuring of steel temperature for steel containments.

#### (Item 4.232) Means for verification of the availability of the systems.

For the guidelines related to leak tightness design levels, the response is set as compliance with objective “CWO” since the AP1000 plant design uses a different but equivalent leak tightness design approach.

For the guidelines related to instrumentation lines that penetrate the containment, responses are set as compliance with objective “CWO” since the AP1000 plant design uses a different but equivalent approach for instrumentation lines that penetrate the containment.

For the guidelines related to pipes and tanks insulation, the response is set as compliance with objective “CWO” for conservatism since the AP1000 plant design uses metallic removable insulations.

For some guidelines related to containment monitoring, the respond is set as compliance with objective “CWO” since the AP1000 plant design uses a different approach that doesn’t compromises containment safety.

*Refer to* ***Subsection 2.2.3*** *of the compliance assessment report [43] for more details.*

#### Section 5 (Tests and inspections)

The AP1000 design meets guidelines addressed in **Section 5** of the reference document [13] “as stated”.

### Non Compliance

No “Non Compliances” have been identified in the assessment [43].

## SSG-54 ACCIDENT MANAGEMENT PROGRAMMES FOR NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [14] as part of the compliance assessment report [44] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [44] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-54: Accident management programmes for nuclear power plants” [14] is expected to be demonstrated either via compliance with the guidelines “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “IAEA Specific Safety Guide No. SSG-54: Accident management programmes for nuclear power plants” has been assigned to a “**Low Risk**” category.

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### Section 2 (General guidance for an accident management programme)

The AP1000 follows an approach that is aligned with the IAEA general guidance to define safety standards for an accident management program presented in **Section 2** of the reference document [14].

Following subsections almost entirely fall under the responsibility of the Owner/Licensee:

* “Verification and validation of the accident management programme”
* “Roles and responsibilities”
* “Staffing, qualification, training and working conditions for accident management”

“Project or site-specific scope” (POS) items were identified in five subsections.

“Forms of accident management guidance” subsection contains POS items related to:

* **(Item 2.50)** Taking into account the capability of communicating within the plant emergency command and control structure and with off-site organizations in accident management guidance.
* **(Item 2.52)** Ensuring by the management system of the operating organization that accident management guidance is not adversely impacted by plant changes.
* **(Item 2.54)** Making potential changes to the EOPs or the SAMGs to the relevant background documentation.

“Verification and validation of the accident management programme” subsection contains POS items related to:

* **(Item 2.56)** Assessment of technical accuracy and adequacy of the accident management guidance to the extent possible, as well as the ability of personnel to follow and implement this guidance.
* **(Item 2.57)** The staff involved in the validation of accident management guidance.
* **(Item 2.58)** The findings and insights from the verification and validation processes.

“Accident management and external hazards” subsection contains a POS item related to:

* **(Item 2.60)** Level of severity of external hazards that has to be considered in the accident management program.

“Accident management for multiple unit sites” subsection contains POS items related to:

* **(Item 2.72)** Sharing of information with the operating organizations of neighboring units at which a severe accident has occurred.
* **(Item 2.73)** Addressing the possibility of multiple units being affected by simultaneous accidents in the accident management guidance.

“Equipment upgrades” subsection contains a POS item related to:

* **(Item 2.79)** Equipment qualification and establishment of the related design requirements when the addition or upgrade of existing equipment or instrumentation is considered for accident management.

*Refer to* ***Subsection 2.2.1*** *of the compliance assessment report [44] for more details.*

#### Section 3 (Development and implementation of a severe accident management programme)

The safety standards about development and implementation of a severe accident management program specified in **Section 3** of the reference document [14] are consistent with the approach followed for the AP1000 accident management program development and implementation.

Following subsections almost entirely fall under the responsibility of the Owner/Licensee:

* “Establishment of a verification and validation process for the severe accident management programme”
* “Integration of the severe accident management programme into the management system and the emergency preparedness and response arrangements”
* “Hardware provisions for severe accident management”
* ‘’Training, exercises and drills for accident management’’
* “Updating the severe accident management programme”

One “Compliant with objective” (CWO) item was identified in “Instrumentation and control for severe accident management” subsection and is related to:

* **(Item 3.96)** Assessment of the uncertainty of readings of instruments essential for severe accident management.

POS items were identified in ten subsections.

“Technical bases” subsection contains POS items related to:

* **(Item 3.2)** Six main steps that should be executed to set up and develop a severe accident management program.
* **(Item 3.3)** Considerations for activities for developing severe accident management guidance.

“Identification of plant vulnerabilities” subsection contains a POS item related to:

* **(Item 3.12)** Identification of the vulnerabilities to postulated hazardous conditions

“Identification of plant capabilities” subsection contains a POS item related to:

* **(Item 3.17)** Preparation of relevant instructions to take actions safely and effectively, defining a set of steps that have been appropriately reviewed and identifying the prerequisites necessary.

“Development of severe accident management guidance” subsection contains POS items related to:

* **(Item 3.25)** Preparation of relevant instructions to take actions safely and effectively, defining a set of steps that have been appropriately reviewed and identifying the prerequisites necessary.
* **(Item 3.59)** Involvement of organizational units responsible for the evaluation, decision making and implementation of accident management actions in the course of a severe accident at early stage of development of a severe accident management programme.

“Establishment of a verification and validation process for the severe accident management programme” subsection contains POS items related to:

* **(Item 3.64)** Possible methods for the validation of the SAMGs and background documents.
* **(Item 3.65)** Encompassing of the uncertainties in the magnitude and timing of phenomena via validation in case of using a full scope simulator.
* **(Item 3.66)** Requirements to validation.

“Integration of the severe accident management programme into the management system and the emergency preparedness and response arrangements” subsection contains a POS item related to:

* **(Item 3.72)** Scope of the on-site emergency plan and review a of the emergency plan and the accident management programme and their testing in exercises.

“Hardware provisions for severe accident management” subsection contains POS items related to:

* **(Item 3.85)** Conditions when changes in the design should be evaluated for existing plants.
* **(Item 3.86)** (For new plants) Independence of additional equipment provided to mitigate the consequences of severe accidents from equipment and systems used to cope with design basis accidents.

“Instrumentation and control for severe accident management” subsection contains a POS item related to:

* **(Item 3.91)** Identification of the instrumentation essential for monitoring the conditions of the core, the containment and the spent fuel during a severe accident. Maintaining monitoring functions throughout an extended loss of AC power. Preparation of a plant specific assessment to identify the equipment, materials, and actions necessary to restore power to the minimum essential components in the event that installed DC batteries are depleted.

“Analyses for development of a severe accident management programme” subsection contains POS items related to:

* **(Item 3.108)** Assessment of applicability of severe accident management guidance based on the generic plant analysis to the specific plant.
* **(Item 3.109)** Preferable usage of plant specific data for analyses.
* **(Item 3.110)** Sufficient input for the development of severe accident management guidance.
* **(Item 3.111)** Provision of sufficient information regarding environmental conditions for the assessment of the operability of the plant equipment and the habitability of working places for personnel involved in the execution of the severe accident management actions.

“Updating the severe accident management programme” subsection contains POS items related to:

* **(Item 3.119)** Evaluation of The effect of changes to the plant design, the available non-permanent equipment and the operating organization for any impact on the severe accident management programme.
* **(Item 3.120)** Development of action plan when modification of the severe accident management programme is deemed appropriate.
* **(Item 3.121)** Evaluation of the capability of installed equipment and the severe accident management procedures and guidelines in order to determine if fundamental safety functions could be compromised when new information is received that challenges current design assumptions relating to external events. Identification of measures for updating the severe accident management programme.
* **(Item 3.123)** Including , as appropriate, a revision of background documentation into any update of the severe accident management programme.

*Refer to* ***Subsection 2.2.2*** *of the compliance assessment report [44] for more details.*

#### Section 4 (Execution of the accident management programme)

Most of the guidelines specified by IAEA in **Section 4** of the reference document [14] are relevant to the Owner/Licensee. However, for the parts that should be considered in the AP1000 accident management program are in compliance with the IAEA guidance and safety standards.

### Non Compliance

No “Non Compliances” have been identified in the assessment [44].

## SSG-56 DESIGN OF THE REACTOR COOLANT SYSTEM AND ASSOCIATED SYSTEMS FOR NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [15] as part of the compliance assessment report [45] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [45] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-56: Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants” [15] is expected to be demonstrated either via compliance with the guidelines “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “IAEA Specific Safety Guide No. SSG-56: Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants” has been assigned to a “**Low Risk**” category.

The AP1000 approach to design of the RCS and associated systems is aligned with the guidance provided in the IAEA guidelines for factors such as design limits, internal and external events influence on design bases, reliability, possible load combinations, manufacturing, installation, testing, maintenance and other aspects listed in the specific safety guide itself.

A significant amount of SSRs are not applicable to the AP1000 design since they describe requirements for design of NPPs using BWR and PHWR technology. Because of the nature of this particular IAEA SSG which focuses purely on design activities, fulfillment of all of the requirements is the responsibility of Westinghouse as a design provider. Several requirements have been labeled as fulfilled due to AP1000 design compliance with the objective of the requirement, but not the requirement itself. Most of these requirements are related to the actuation of safety related systems, residual heat removal and ultimate heat sink design. Compliance with objectives of these requirements have been justified using design bases of innovative, passive safety systems introduced in AP1000 NPP.

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### Section 2 (Extent of the reactor coolant system and associated systems)

The character of **Section 2** of the reference document [15] is explanatory.

#### Section 3 (Design basis of the reactor coolant system and associated systems)

Guidelines presented in **Section 3** of the reference document [15] are met by Westinghouse either “as stated” or via compliance with their objectives.

“Compliant with objective” (CWO) items were identified in five subsections.

“General” subsection contains a CWO item related to:

* **(Item 3.3)** Requirements to be met by design process.

“Internal hazards” subsection contains a CWO item related to:

* **(Item 3.13)** Recommendations that should be followed identify internal hazards to be considered in the design of the reactor coolant system and associated systems.

“External hazards” subsection contains CWO items related to:

* **(Item 3.18)** The recommendations provided in the following IAEA safety standards that should also be considered to understand the general concepts, ensure identification of the relevant external hazards and protect the reactor coolant system and associated systems against the effects of these hazards
* **(Item 3.25)** Accomplishment of short term actions necessary to preserve the integrity of the reactor coolant pressure boundary and to prevent conditions from escalating to design extension conditions with core melting

“Reliability” subsection contains a CWO item related to:

* **(Item 3.56)** Applying recommendations in respect of design extension conditions, taking into account that meeting the single failure criterion is not necessary and that the additional safety features for design extension conditions are supplied by the alternate AC power source and batteries.

“Overpressure protection” subsection contains a CWO item related to:

* **(Item 3.120)** Using the same code for the design, manufacturing and overpressure analysis of a given component.

*Refer to* ***Subsection 2.4*** *of the compliance assessment report [45] for more details.*

#### Section 4 (Ultimate heat sink and residual heat transfer systems)

Guidelines presented in **Section 4** of the reference document [15] are met by Westinghouse either “as stated” or via compliance with their objectives.

CWO items were identified in two subsections.

“Ultimate heat sink” subsection contains CWO items related to:

* **(Item 4.16)** Precise identification of various heat sources and their time dependent behaviour in determining the capacities demanded of the ultimate heat sink and its directly associated heat transfer systems.
* **(Item 4.18)** Evaluation of the total heat load and the rejection rate of heat from spent fuel.

“Residual heat transfer systems” subsection contains a CWO item related to:

* **(Item 4.39)** Protection of cooling system pumps that are directly connected to the ultimate heat sink against debris and biofouling. AP1000 does not count with such active safety related pumps.

*Refer to* ***Subsection 2.5*** *of the compliance assessment report [45] for more details.*

#### Section 5 (Specific considerations in design of the reactor coolant system)

Guidelines presented in **Section 5** of the reference document [15] are met by Westinghouse either “as stated” or via compliance with their objectives.

CWO items were identified in two subsections.

“Pressure control and overpressure protection” subsection contains CWO items related to:

* **(Item 5.36)** Provision of safety valves, safety relief valves and relief valves with a position indicator that is independent of the control equipment.
* **(Item 5.40)** Prevention of the spurious opening of a safety valve (or, for BWRs, a safety relief valve) should be prevented, and the frequency limits of such spurious opening

“Specific design aspects” subsection contains CWO items related to:

* **(Item 5.102)** Controlling of seal leakage in reactor coolant pumps
* **(Item 5.121)** Pipe arrangement to limit the possibility of the accumulation of non-condensable gases.

*Refer to* ***Subsection 2.6*** *of the compliance assessment report [45] for more details.*

#### Section 6 (Specific considerations in design of the associated systems for PWR technology)

Guidelines presented in **Section 6** of the reference document [15] are met by Westinghouse either “as stated” or via compliance with their objectives.

CWO items were identified in two subsections.

“The system for the long term removal of residual heat” subsection contains CWO items related to:

* **(Item 6.63)** Location and qualification of the emergency core cooling system equipment. This requirement is CWO since it is a fundamental AP1000 design choice focusing on containing all potentially contaminated equipment in the containment. Thus, they will be physically protected by containment and shield building.
* **(Item 6.64)** limiting the risk of causing overpressurization of the reactor coolant system via the operation of the emergency core cooling system
* **(Item 6.68)** Ensuring the minimum, net positive suction head for the normal operation of the emergency core cooling system pumps considering limiting phenomena.
* **(Item 6.77)** Consideration of the diversity of the emergency feedwater system pumps
* **(Item 6.78)** Qualification of the steam dump valves to atmosphere
* **(Item 6.81)** Performing of isolation of main steam relief valves from the affected steam generator in the event of a steam generator tube rupture
* **(Item 6.83)** The system for the long term removal of residual heat
* **(Item 6.90)** Considerations of additional design provisions in order to cope with multiple failures resulting in the loss of the systems and safety systems designed to remove residual heat during reactor coolant system conditions that are not compatible with the residual heat removal operation
* **(Item 6.91)** Including connection lines to the emergency feedwater system to facilitate the management of conditions beyond design basis accidents.
* **(Item 6.93)** Inclusion of a fast depressurization of the primary circuit that should be used at the onset of a core melting accident into the design for the practical elimination of the phenomena associated with high pressure melt ejection in severe accidents.

“Systems for control of core reactivity in accident conditions” subsection contains a CWO item related to:

* **(Item 6.101)** Design of the system for control of core reactivity in accident conditions in accordance with the engineering design rules for safety systems.

*Refer to* ***Subsection 2.7*** *of the compliance assessment report [45] for more details.*

#### Section 7 (Specific considerations in design of the associated systems for BWR technology)

All of the guidelines in **Section 7** of the reference document [15] do not apply to AP1000 design.

#### Section 8 (Specific considerations in design of the associated systems for PHWR technology)

All of the guidelines in **Section 8** of the reference document [15] do not apply to AP1000 design.

### Non Compliance

No “Non Compliances” have been identified in the assessment [45].

## SSG-61 FORMAT AND CONTENT OF THE SAFETY ANALYSIS REPORT FOR NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [16] as part of the compliance assessment report [46] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [46] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-61: Format and content of the safety analysis report for nuclear power plants” [16] is expected to be demonstrated but will require development of a SAR input file for the chapter 21 on “Decommissioning”. Since no non-compliances have been identified and no design changes are anticipated, “IAEA Specific Safety Guide No. SSG-61: Format and content of the safety analysis report for nuclear power plants” has been conservatively assigned to a “**Medium Risk**” category.

The format and contents of the SAR as presented in IAEA SSG-61 are very similar to the content of the DCD [28] and UFSAR [29], as demonstrated by the assessment in Section and summarized in **Table 2-1.**

The content of the SAR for future AP1000 plant projects will follow closely the content specified in the DCD [28] and the UFSAR for the Reference Plant. The use of this standardized content and format further ensures minimal regulatory risk as it has been demonstrated through NRC reviews and approvals for both the certified AP1000 plant design documented in the DCD [28] and the design and licensing updates implemented and approved by the NRC for the Reference Plant design. Specifically, Westinghouse proposes to develop the SAR content based on the guidance provided in U.S. NRC Regulatory Guide 1.70 with the appropriate adjustments for national nuclear regulations for a specific AP1000 plant project. The use of this basis provides a direct tie to the extensive NRC reviews and approvals of the standard AP1000 plant design and the Final Safety Analysis Report for the Vogtle 3 and 4 units.

The following is specific content identified in IAEA SSG-61 that is not included as specific chapters in the DCD/UFSAR:

* IAEA SSG-61 Chapter 13: Emergency preparedness and response
* IAEA SSG-61 Chapter 20: Environmental aspects
* IAEA SSG-61 Chapter 21: Decommissioning and end of life aspects

**Table 2-1: IAEA SSG-61 Content and Format Comparison to DCD/UFSAR Content and Format Comparison**

| **IAEA SSG-61 Chapter and Title** | | **DCD/UFSAR Chapter and Title** | | **Comparison Notes** |
| --- | --- | --- | --- | --- |
| 1 | Introduction and general considerations | 1 | Introduction and General Description of the Plant | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content. |
| 2 | Site characteristics | 2 | Sites Characteristics and Site Parameters | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content. |
| 3 | Safety objectives and design rules for structures, systems and components | 3 | Design of Structures, Components, Equipment, and Systems | The IAEA SSG-61 Chapter 3 content for the design rules of structures, systems, and components is equivalent to DCD/UFSAR Chapter 3.  The safety objectives identified for the IAEA SSG-61 Chapter 3 are based on IAEA SSR-2/1. DCD Section 3.1 is the equivalent section; it identifies conformance with the US NRC General Design Criteria. |
| 4 | Reactor | 4 | Reactor | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content. |
| 5 | Reactor coolant system and associated systems | 5 | Reactor Coolant System and Connected Systems | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content. |
| 6 | Engineered safety features | 6 | Engineered Safety Features | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content. |
| 7 | Instrumentation and control | 7 | Instrumentation and Controls | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content. |
| 8 | Electrical power | 8 | Electrical power | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content. |
| 9 | Auxiliary systems and civil structures | 9 | Auxiliary Systems | There are differences between the IAEA SSG-61 and DCD/UFSAR Chapter 9 as described for IAEA SSG-61 Chapters 9A and 9B |
| 9A | Auxiliary systems | - | - | DCD/UFSAR Chapter 9 content for Auxiliary systems is equivalent to the IAEA SSG-61 content. |
| 9B | Civil engineering works and structures | - | - | UFSAR/DCD Chapter 9 does not include a section on Civil engineering works and structures.  Civil engineering works and structures are described in DCD/UFSAR Chapter 1 and Chapter 3. |
| 10 | Steam and power conversion systems | 10 | Steam and power conversion systems | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content. |
| 11 | Management of radioactive waste | 11 | Radioactive Waste management | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content. |
| 12 | Radiation protection | 12 | Radiation protection | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content. |
| 13 | Conduct of operations | 13 | Conduct of operations | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content.  Conduct of Operations content is mainly the responsibility of the Owner to develop. |
| 14 | Plant construction and commissioning | 14 | Initial Test Program | The DCD/UFSAR chapter presents the Initial Test Program to be performed during construction and commissioning.  It does not include content related to the construction program for the AP1000 plant. |
| 15 | Safety analysis | 15 | Accident Analyses | IAEA SSG-61 Chapter 15 includes deterministic safety analysis, probabilistic risk assessment, and design extensions with and without core melt.  IAEA SSG-61 Chapter 15 is equivalent to the content in DCD/UFSAR Chapter 15 combined with DCD/UFSAR Chapter 19. |
| 16 | Operational limits and conditions for safe operation | 16 | Technical Specifications | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content. |
| 17 | Management for safety | 17 | Quality Assurance | The AP1000 plant DCD Chapter content focuses on the design Quality Management System utilized by Westinghouse for the AP1000 plant design.  For IAEA SSG-61, this chapter also includes the overall safety management program for the operating plant. This content development would be the responsibility of the Owner. |
| 18 | Human factors engineering | 18 | Human Factors Engineering | IAEA SSG-61 and DCD/UFSAR chapter have equivalent content. |
| 19 | Emergency preparedness and response | - | - | Emergency Preparedness and Response is not a standard DCD/UFSAR chapter.  This content is mainly the responsibility of the Owner to develop. It could be added as a new SAR chapter or developed as a separate report. |
| - | - | 19 | Probabilistic Risk Assessment | For the DCD/UFSAR, Chapter 19 describes PRA and Severe Accidents. This would be maintained as Chapter 19 for consistency with the US Reference Plant licensing basis documentation. |
| 20 | Environmental aspects |  | None | This is not a standard chapter in the AP1000 plant DCD/USAR.  It is generally the responsibility of the Owner to develop the Environmental Impact Assessment, and that document would provide input to this SAR chapter. |
| 21 | Decommissioning and end of life aspects |  | None | This is not a standard chapter in the AP1000 plant DCD/USAR.  This content would need to be developed as a new chapter or separate document. |

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### Section 2 (General Considerations)

Most of the guidelines in **Section 2** of the reference document [16] which fall under responsibility of Westinghouse are met either “as stated” or via compliance with their objectives. Compliance with the rest of relevant guidelines is expected as result of COM-P scope.

The rest of the guidelines in **Section 2** belong to the Owner’s scope of responsibilities.

Westinghouse provides a SAR Input Document to the Owner which is based on the DCD rev. 19 [1] with applicable changes from the Reference Plant UFSAR [2], and will include applicable updates on a project specific basis. This SAR Input Document will support the stages of SAR development. The DCD and UFSAR provide representative content for the stage from preliminary SAR to operational SAR.

SAR Input Document provided by Westinghouse includes 19 Chapters as they are described in the Reference Plant and is in line with the licensing process in the United States. Any deviations like adding new Chapters required per licensing process in other countries can be made on the request from the Owner. Decision to fully follow the approach proposed by the SSG-61 is the choice of the Owner of the plant. Westinghouse can comply with this guidance by providing input to proper Chapters per Owner’s request.

Some of the requirements are marked as COM-P as content for decommissioning is not included in the DCD/ Reference Plant UFSAR and will need to be developed on a project specific basis.

Owner is responsible for maintaining the SAR document in all stages of the lifetime of the plant.

Westinghouse as a mature organization with a robust experience in nuclear licensing can support the Owner in preparing the SAR document, both by providing input or by supporting the Owner in preparing and maintaining the SAR document.

SAR Document for the AP1000 plant contains sensitive information or is referencing sensitive information. Sensitive and proprietary information owned by Westinghouse is withheld from the public but has been shared with Regulators in different countries as part of many extensive licensing processes across the world. Westinghouse will identify to the Owner the information that can be public (non-proprietary) in the SAR and the information that needs to be maintained as Westinghouse proprietary.

“Compliant with objective” (CWO) items were identified in three subsections.

“Structure of the safety analysis report” subsection contains a CWO item related to:

* **(Item 2.12)** Structure of the safety analysis report.

“Unified description of the design of plant structures, systems and components” subsection contains CWO items related to:

* **(Item 2.15)** Requirement for all plant SSCs that have the potential to affect safety to be described in the safety analysis report. The type of information about each SSC to be included in the safety analysis report depends on the particular type and design of the reactor selected for construction; Requirement for this information should be sufficient to review these SSCs in terms of their compliance with national laws and regulations.
* **(Item 2.16)** Requirement for descriptions of all the SSCs important to safety to be provided, together with a demonstration of the conformance of these SSCs with the relevant design requirements; Requirement for the level of detail in each description to be commensurate with the importance of the structure, system or component to safety.

“Formal aspects regarding the documentation of the safety analysis report” subsection contains CWO items related to:

* **(Item 2.22)** The operating organization having the prime responsibility for safety; Requirement for the safety analysis report, if developed by a third party (e.g. by the nuclear power plant vendor), to contain sufficiently detailed information, either in the report itself or in referenced documents, to allow for an independent verification; Requirement for this verification to be conducted either by the operating organization or by another qualified organization on its behalf; Irrespective of the process followed for the development and verification of the safety analysis report, the operating organization remains responsible for the content, comprehensiveness and quality of the safety analysis report.

“Compliant with planned update for new European AP1000 plant projects” (COM-P) items were identified in four subsections.

“Structure of the safety analysis report for various stages of the lifetime of a nuclear power plants” subsection contains COM-P items related to:

* **(Item 2.8)** Requirement for the preliminary safety analysis report to contain sufficiently detailed information, specifications and supporting calculations to assess and demonstrate that the plant can be constructed, commissioned, operated and decommissioned in a manner that is acceptably safe throughout its lifetime; Requirement for the preliminary safety analysis report to demonstrate that the requirements specified in the initial safety analysis report are met; Requirement for the safety features incorporated into the design to be described, with due regard to any site specific aspects.
* **(Item 2.10)** Requirement for the final safety analysis report to be initially prepared as an update of the pre‑operational safety analysis report and for additional information obtained during the operational stage of the nuclear power plant to be incorporated periodically into the final safety analysis report; Requirement for this information to include any plant modifications with their justification and particular attention given to documenting information that is relevant to the decommissioning of the nuclear power plant.
* **(Item 2.11)** Consideration of periodic updates of the approach and associated conditions regarding the future decommissioning of the nuclear power plant by this Safety Guide. However, the scope of the safety analysis report for an advanced decommissioning phase, when the nuclear fuel has been removed from the plant after a suitable cooling period is not specifically addressed.

“Structure of the safety analysis report” subsection contains COM-P items related to:

* **(Item 2.12)** Structure of the safety analysis report.
* **(Item 2.14)** Requirement stating that the proposed structure of the safety analysis report incorporating several chapters that have often been covered by separate documents. Examples of such chapters are those on operational limits and conditions (OLCs) for safe operation, management for safety, emergency preparedness and response, environmental aspects, and decommissioning and end of life aspects; Requirement stating that while in general it is acceptable to have separate documents to complement the safety analysis report, at least for new nuclear power plants all such additional documents should be either summarized or referenced in the safety analysis report to ensure completeness, the appropriate use of confidential information and consistency with other licensing documents. The specific approach may differ for different stages of the safety analysis report. For example, including environmental aspects is relevant for the initial safety analysis report and uses information usually available from the report on the environmental impact assessment, while in subsequent safety analysis reports the radiological impact on people and the environment should be comprehensively covered by the safety analysis included in chapter 15 of the safety analysis report.

“Formal aspects regarding the documentation of the safety analysis report” subsection contains a COM-P item related to:

* **(Item 2.21)** Requirement for the safety analysis report to document the safety of the nuclear power plant with a scope and level of detail sufficient to support the conclusions reached and to provide an adequate input to the review undertaken by the regulatory body; Requirement for the depth of description provided in the safety analysis report to reflect the requirement for the report to be a key reference document and for it to be sufficiently detailed to be understandable by itself.

“Relationship of the safety analysis report to other licensing documents” subsection contains a COM-P item related to:

* **(Item 2.26)** Requirement stating that in addition to the safety analysis report, other documents are used in the licensing process. Typical examples include reports on the environmental impact assessment, probabilistic safety assessment studies, emergency plans and decommissioning plans. In some States, information from these reports is part of the safety analysis report.

*Refer to* ***Subsection 2.3*** *of the compliance assessment report [46] for more details.*

#### Section 3 (Content and structure of individual chapters of the safety analysis report)

**Section 3** of the reference document [16] presents guidelines that address content and structure of individual chapters of the safety analysis report.

Guidelines in **Section 3** fall under scope of responsibility of the Owner, Westinghouse or both.

Guidelines and parts of the guidelines that are responsibility of Westinghouse are mostly met “as stated” or via compliance with their objectives.

In some occasions compliance requires a project specific scope to be done. Such guidelines were assigned to a COM-P category.

CWO items were identified in twelve subsections.

“Chapter 1: Introduction and general considerations” subsection contains CWO items related to:

* **(Item 3.1.1)** Items to be included into safety analysis report introduction.
* **(Item 3.1.2)** Requirement for the information provided in the project implementation section to include a description of the existing authorization status of the plant, with an indication of future project milestones, as appropriate.
* **(Item 3.1.11)** Requirement for all operating modes of the nuclear power plant to be described: startup, power operation, shutting down, shutdown (including long term shutdown), maintenance, testing, refueling, and any other allowable modes of normal operation, including load following operation. The permissible periods of operation at different power levels in the event of a deviation from normal operating conditions should be specified.
* **(Item 3.1.13)** Requirement for “Additional supporting or complementary documents to the safety analysis report” section to provide a list and summary of the topical reports that are incorporated, by reference, as part of the safety analysis report. Typically, the results of tests and analyses (e.g. results of manufacturers’ material tests and qualification data) may be submitted as separate reports.
* **(Item 3.1.14)** Requirement for “Conformance with applicable regulations, codes and standards” section to provide an overview of the relevant regulations, codes and standards that collectively represent the safety rules used in the design, including information on the use of relevant IAEA safety standards; Requirement to provide a justification of appropriateness of regulations, codes and standards If they have not been prescribed by the regulatory body; Requirement for any deviations from existing regulations, codes and standards to be described in this section, together with a demonstration that the deviations will not be detrimental to safety.

“Chapter 3: Safety objectives and design rules for structures, systems and components” subsection contains CWO items related to:

* **(Item 3.3.2)** Requirement forthe overall safety philosophy and general approaches for ensuring safety to be presented in “General safety design basis” section; Requirement for these approaches, in addition to any national requirements and associated regulatory guidance, to be based on the requirements for the design of nuclear power plants established in SSR‑2/1.
* **(Item 3.3.3)** Requirement for “Safety objectives” section to summarize the overall safety philosophy, safety objectives and high level principles used in the project. Requirement for these to be based on the relevant safety principles set out in IAEA Safety Standards Series No. SF 1, Fundamental Safety Principles.
* **(Item 3.3.4)** Requirement for “Safety functions” section to identify the plant specific safety functions that are necessary to fulfil the main safety functions and how their fulfilment is ensured by the plant’s inherent features, in accordance with Requirement 4 of SSR‑2/1 and depending on the nature of the facility or activity; Requirement for the corresponding SSCs necessary to fulfil those safety functions should be introduced.
* **(Item 3.3.5)** Requirement for the main safety functions, if they are subdivided into more detailed specific safety functions and functional criteria, with the objective of facilitating their use, to be listed in “Safety functions” section, for example heat removal, which is considered a safety function necessary not only for the safety of the reactor core but also for the safety of any other part of the plant containing radioactive material that needs to be cooled, such as spent fuel pools and storage areas.
* **(Item 3.3.9)** Requirement for “General design basis and plant states considered in the design” section to describe the capability of the plant to cope with a specified range of operational states and accident conditions; Requirement for the modes of normal operation of the plant to be specified; Requirement for the plant states considered in the design to be listed and grouped into categories; In addition to normal operation, requirement for these categories to include anticipated operational occurrences, design basis accidents, design extension conditions without significant fuel degradation and design extension conditions with core melting.
* **(Item 3.3.10)** Requirement for the basis for the categorization of plant states (typically, frequencies or other associated characteristics) to be explained; Requirement for postulated initiating events (whether of internal origin or caused by internal and external hazards, if relevant) to be listed; Requirement for this categorization should be commensurate with the content of chapter 15 of the safety analysis report.
* **(Item 3.3.11)** Requirement for “Prevention and mitigation of accidents” section to describe the measures taken to prevent and to mitigate the consequences of accidents and to ensure that the likelihood that an accident will have harmful consequences is extremely low.
* **(Item 3.3.12)** Requirement for “Defence in depth” section to describe the approach adopted to incorporate the defence in depth concept into the design of the plant; Requirement to demonstrate that the defence in depth concept has been applied at all stages of the lifetime of the nuclear power plant, for all plant states and for all safety related activities, in accordance with paras 2.12–2.18 of SSR 2/1 (Rev. 1). Requirement to demonstrate that measures have been taken for adequate robustness and independence of levels with particular emphasis placed on describing how the independence of safety systems and safety features for design extension conditions with core melting is approached.
* **(Item 3.3.13)** Requirement to demonstrate that there are physical barriers to the release of radioactive material and systems to protect the integrity of the barriers and that measures are taken to ensure the robustness of these provisions at each level of defence in depth.
* **(Item 3.3.16)** Requirement for “Application of general design requirements and technical acceptance criteria” section to include a high level description of the deterministic design principles. Requirement for the use of such design approaches to be elaborated in this section of the safety analysis report, with reference made to the specific applicable codes and standards, for cases where aspects of the design are based on conservative deterministic principles (such as those embodied in international standards, internationally recognized industrial codes and standards, and regulatory guides).
* **(Item 3.3.17)** Requirement forthe scope of implementation of the single failure criterion and how compliance with this criterion is achieved in the design to be described in “Application of general design requirements and technical acceptance criteria” section of the safety analysis report; Requirement for this section to also include results from the consideration of the possibility of a single failure occurring while a redundant train of a system is undergoing maintenance or is impaired by internal or external hazards.

“Chapter 5: Reactor coolant system and associated systems” subsection contains CWO items related to:

* **(Item 3.5.6)** Requirement for a schematic flow diagram of the reactor coolant system and associated systems denoting all major components, principal pressures, temperatures, flow rates and coolant volume under normal steady state, full power operating conditions to be provided; Requirement for an elevation drawing of the piping and instrumentation of the reactor coolant system and associated systems showing the principal dimensions of the reactor coolant system in relation to the supporting or surrounding concrete structures to also be provided.

“Chapter 6: Engineered safety features” subsection contains CWO items related to:

* **(Item 3.6.6)** Requirement for the use of non‑permanent equipment as part of accident management to be described in “Engineered safety features” chapter of the safety analysis report; Requirement for the information provided to demonstrate that there are adequately robust design features to enable the reliable connection of non‑permanent equipment, including connection during conditions induced by external hazards exceeding those of the design basis (see paras 6.28B, 6.45A and 6.68 of SSR‑2/1 (Rev. 1)).
* **(Item 3.6.12)** Requirement for the “Safety features for stabilization of the molten core” section to provide relevant information on safety features to stabilize the molten core as a necessary means of molten core solidification — either inside the reactor pressure vessel or in a dedicated molten core localization system — as a necessary precondition for protecting the containment basemat and ensuring containment integrity in the long term.
* **(Item 3.6.17)** Requirement for the habitability of control locations under design extension conditions with core melting to be addressed in this section of the safety analysis report. Requirement for the description, for the remote sites, to include a demonstration of the habitability of these locations in the case of external hazards exceeding the design basis events combined with internal events.
* **(Item 3.6.18)** Requirement for **“**Systems for the removal and control of fission products” section to provide relevant information on the systems for the removal and control of fission products (if not already described as a part of the containment systems); Specific information that should be presented to demonstrate the performance capability of these systems.

“Chapter 7: Instrumentation and control” subsection contains CWO items related to:

* **(Item 3.7.22)** Requirement for the “Instrumentation and control in the main control room” section to provide a description of the main control room layout, with emphasis on the presentation of information from the instrumentation and control in the main control room and the human–machine interface.
* **(Item 3.7.23)** Requirement for the “Instrumentation and control in the main control room” section to describe how the human–machine interface aspects of the design of the main control room conform to the human factors engineering program described in chapter 18 of the safety analysis report.

“Chapter 8: Electrical power” subsection contains CWO items related to:

* **(Item 3.8.13)** Items to be described in the “On‑site AC power systems” section.
* **(Item 3.8.18)** Four classes of cable to be identified in the “Electrical equipment, cables and raceways” section.
* **(Item 3.8.19)** Requirement for the “Electrical equipment, cables and raceways” section to describe the environmental qualification of cables and electrical penetrations that have to withstand conditions inside the containment during and after a loss of coolant accident, a main steam line break or other adverse environmental conditions, including severe accidents.

“Chapter 9B: Civil engineering works and structures” subsection contains CWO items related to:

* **(Item 3.9.19)** Requirement for Part B of chapter 9 of the safety analysis report to describe how the general design requirements specified in chapter 3 of the safety analysis report have been complied with in the design of specific structures in the nuclear power plant; Requirement for the three groups of civil structures to be considered: the foundations, the reactor building and other civil structures; Requirement to follow , to the extent possible, a standardized format for the information provided (specified in Appendix II) in describing the structures.
* **(Item 3.9.20)** Information specific to civil engineering works and structures that should be provided.
* **(Item 3.9.21)** Requirement for“Foundations and buried structures” section to provide, information on the foundations, including diagrams containing plan and section views of the foundations, to define the primary structural aspects and elements relied on to perform the foundation function; Requirement for the description to include the soil–structure interaction; Requirement to present the type of foundation, its structural characteristics and the general arrangement of each foundation and to describe foundations of steel or concrete containment, as well as all seismically classified structures.
* **(Item 3.9.22)** Requirement for “Reactor building” section to describe the design features of the reactor building provided to comply with Requirements 54–58 of SSR 2/1 (Rev. 1); Requirement for the specific design features of the primary containment, such as its leaktightness, mechanical resistance, pressure-retaining capability and protection against hazards, to be covered and for the concrete and steel internal structures of the containment to be described; Requirement to describe a secondary containment in this section of the safety analysis report if the design incorporates it; Requirement for the information described in this section of the safety analysis report to be consistent with and complementary to the information provided in chapter 6 of the safety analysis report.
* **(Item 3.9.23)** Requirement for “Reactor building” section to also provide sufficient information to demonstrate the performance of the containment in all plant states and combinations of loads, in accordance with established acceptance criteria (see SSG‑53).
* **(Item 3.9.24)** Requirement for other civil structures of the plant that are relevant to nuclear safety to be described in “Other structures” section; this includes the control building, the auxiliary building, the ultimate heat sink structures and the emergency response facilities.

“Chapter 10: Steam and power conversion systems” subsection contains CWO items related to:

* **(Item 3.10.5)** Requirement for the descriptions to include sufficient detail to demonstrate the reliable fulfilment of safety functions, including fast and reliable isolation and steam relief; Requirement for a demonstration that the separation of steam lines prevents leakage from one affecting another, and provides protection against an aircraft crash, to also be included.
* **(Item 3.10.17)** Requirement for the “Implementation of break preclusion for the main steam and feedwater lines” section to describe the scope of the implementation of break preclusion in the main steam and feedwater lines; Requirement for the aspects that impact plant safety (either direct effects on the fulfilment of the fundamental safety functions or indirect effects such as secondary damage to the plant systems, for example by pipe whip or extraordinary pressure loading) to be emphasized; Requirement for the description to also include how the ‘leak before break’ concept has been implemented (if relevant).

“Chapter 11: Management of radioactive waste” subsection contains CWO items related to:

* **(Item 3.11.6)** Treating the assessment of gaseous and liquid releases resulting from accident conditions in chapter 15 of the safety analysis report, although the results of such assessments may also be described in “Sources of waste” section and used as input.

“Chapter 12: Radiation protection” subsection contains CWO items related to:

* **(Item 3.12.6)** Requirement for the information provided in “Radiation protection” chapter of the safety analysis report to demonstrate compliance with IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards; with paras 2.6 and 2.7 and with Requirement 81 of SSR‑2/1 (Rev. 1); and with Requirement 20 of SSR‑2/2 (Rev. 1).

“Chapter 15: Safety analysis” subsection contains CWO items related to:

* **(Item 3.15.3)** Requirement for the scope of information provided in chapter 15 of the safety analysis report should reflect the requirements on safety analysis relevant for nuclear power plant design, in particular Requirements 16, 17, 19, 20 and 42 of SSR 2/1 (Rev. 1) and Requirements 14–21 of GSR Part 4 (Rev. 1) Recommendations and guidance on deterministic safety analysis are provided in IAEA Safety Standards Series No. SSG 2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants; recommendations on probabilistic safety assessment are provided in IAEA Safety Standards Series No. SSG 3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, and IAEA Safety Standards Series No. SSG 4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants.

The AP1000 plant DCD Chapter 19 and design PRA [30] describe the process for identifying and assessing a comprehensive set of initiating events based on evaluations that included review of PWR operating experience, past PRAs, and consideration of the AP1000 plant specific features.

* **(Item 3.15.4)** Requirement for the information provided in this chapter of the safety analysis report to be sufficient to justify and confirm the design basis for items important to safety and to ensure that the overall plant design is capable of meeting the established acceptance criteria, in particular the dose limits and the authorized limits for radioactive releases associated with each plant state, and that the consequences of accidents are as low as reasonably achievable.
* **(Item 3.15.10)** Requirement for the approach used to identify Postulated Initiating Events (PIE) and accident scenarios for both deterministic and probabilistic analyses to be described in ‘’Identification, categorization and grouping of postulated initiating events and accident scenarios’’ section. This may include the use of analytical methods such as screening of defence in depth, master logic diagrams, hazard and operability analysis, and failure mode and effects analysis (see SSG‑2 (Rev. 1).
* **(Item 3.15.11)** Requirement for a confirmation in ‘’Identification, categorization and grouping of postulated initiating events and accident scenarios’’ section that the identification of postulated initiating events and accident scenarios to be analysed has been performed in a systematic way and has led to the development of a comprehensive list of events.
* **(Item 3.15.12)** Requirement for subdividing the events into categories in accordance with their anticipated frequencies and grouping according their type (i.e. taking into account their effect on the plant); Purpose of this categorization.
* **(Item 3.15.13)** Requirement for the basis for the categorization and grouping of postulated initiating events to be described and justified; In addition to normal operation, requirement for the list of scenarios to be addressed in the safety analysis report to cover anticipated operational occurrences, design basis accidents, design extension conditions without significant fuel degradation and design extension conditions with core melting; Requirement for the postulated initiating events taking place in all modes of normal operation (from shutdown to low power to full power operation) to be covered, including potential events that could occur during commissioning and testing of the nuclear power plant; Requirement for multiple failures that are considered to be plausible to be presented in ‘’Identification, categorization and grouping of postulated initiating events and accident scenarios’’ section.
* **(Item 3.15.16)** Requirement forfailures that are considered as initiated in plant systems other than the reactor coolant system, such as the containers or stores for fresh or irradiated fuel and storage tanks for radioactive gaseous or liquid wastes, to also be in ‘’Identification, categorization and grouping of postulated initiating events and accident scenarios’’ section.
* **(Item 3.15.21)** Requirement for the radiological acceptance criteria relating to radiological consequences and the technical acceptance criteria relating to the integrity of barriers to be specified in ‘’Identification, categorization and grouping of postulated initiating events and accident scenarios’’ section for different categories of events and types of analysis; Requirement for the information on acceptance criteria given in ‘’Identification, categorization and grouping of postulated initiating events and accident scenarios’’ section to be consistent with the more general information provided in chapter 3 of the safety analysis report.
* **(Item 3.15.22)** Requirement for the used specific values to be provided in ‘’Identification, categorization and grouping of postulated initiating events and accident scenarios’’ section, if probabilistic values such as core damage frequency or large release frequency are established as acceptance criteria or safety objectives.
* **(Item 3.15.23)** Requirement forthe selection of the acceptance criteria for individual postulated initiating events and for accident scenarios to be described in this section and for the scope and conditions of applicability of each specific criterion to be clearly specified.
* **(Item 3.15.68)** Requirement for ‘’Summary of results of the safety analyses’’ section to confirm that the requirements for safety analysis relevant to nuclear power plant design (i.e. mainly those established in SSR‑2/1 (Rev. 1) and GSR Part 4 (Rev. 1) have been met in every respect, providing justification if those requirements have been revised, or have been applied with changes as a result of further considerations. In the latter cases, requirement to specify any compensatory measures taken to meet the revised safety requirements.

“Chapter 18: Human factors engineering” subsection contains CWO items related to:

* **(Item 3.18.9)** Requirement for “Task analysis” section to describe the approach to task analysis for groups of operating personnel relevant to the task being analysed (e.g. operators of the reactor, operators of the turbines, shift supervisors, field operators, safety engineers, operation and maintenance staff); Requirement for the tasks described to cover all plant states.

COM-P items were identified in three subsections.

“Chapter 14: Plant construction and commissioning” subsection contains COM-P items related to:

* **(Item 3.14.4)** As a part of the commissioning program, the requirement for chapter 14 of the safety analysis report to also demonstrate that operating procedures are verified and validated in accordance with para. 6.9 of SSR‑2/2 (Rev. 1) [4] and that this verification and validation will be conducted with the participation of future operating personnel.
* **(Item 3.14.7)** Items that should be included to the specific information provided in the safety analysis report prior to plant construction.
* **(Item 3.14.8)** (Updated) information that should be included to the specific information provided in the safety analysis report prior to plant commissioning.

“Chapter 15: Safety analysis” subsection contains COM-P items related to:

* **(Item 3.15.14)** Requirement for the resulting list of plant specific events and accident scenarios of all types (both internal and external to the plant) to be presented in this section for all modes of normal operation (including operation at power or during shutdown and refueling) and for other relevant plant conditions that will be analyzed (e.g. manual or automatic plant control).
* **(Item 3.15.16)** Requirement for Failures that are considered as initiated in plant systems other than the reactor coolant system, such as the containers or stores for fresh or irradiated fuel and storage tanks for radioactive gaseous or liquid wastes, should also be described here.
* **(Item 3.15.17)** Requirement for the interactions between the reactor core and the spent fuel pool, as well as their mutual impact, to be identified, where appropriate (for consideration as sources of initiating events).
* **(Item 3.15.19)** Requirement for “Identification, categorization and grouping of postulated initiating events and accident scenarios” section to, with reference to specific analyses presented in this safety analysis report, also list the conditions that could lead to an early radioactive release or a large radioactive release and thus need to be ‘practically eliminated’, as required by para. 5.31 of SSR‑2/1 (Rev. 1).
* **(Item 3.15.21)** Requirement for the radiological acceptance criteria relating to radiological consequences and the technical acceptance criteria relating to the integrity of barriers to be specified in “Safety objectives and acceptance criteria” section for different categories of events and types of analysis; Requirement for the information on acceptance criteria given in this section to be consistent with the more general information provided in chapter 3 of the safety analysis report.
* **(Item 3.15.41 (g))** Requirement for a separate subsection to be included, for each individual group of postulated initiating events analysed, providing the following information on the results of the assessment of the radiological consequences of a given event, if applicable; Requirement for the key results to be compared with the radiological acceptance criteria. The analysis of radiological consequences can be presented together with other results in a common section for each relevant postulated initiating event analysed, or it can be placed in a separate section together with all the design basis accident analyses that show radiological consequences, with an appropriate selection of bounding cases for different categories of events.
* **(Item 3.15.51)** Requirement for “Analysis of postulated initiating events and accident scenarios associated with the spent fuel pool” section to present the safety analysis performed for postulated initiating events caused by the release of radioactive material from a subsystem or component (typically from systems for the treatment or storage of radioactive waste), from minor leakage from a radioactive waste system to the overheating of, or damage to, used fuel in transit or storage, or a large break in a gaseous or liquid waste treatment system.

The entire “Chapter 21: Decommissioning and end of life aspects” subsection was classified as NOC and COM-P.

*Refer to* ***Subsection 2.4*** *of the compliance assessment report [46] for more details.*

### Non Compliance

The following is specific content identified in IAEA SSG-61 that is not included as specific chapters in the DCD/UFSAR:

* IAEA SSG-61 Chapter 13: Emergency preparedness and response
* IAEA SSG-61 Chapter 20: Environmental aspects
* IAEA SSG-61 Chapter 21: Decommissioning and end of life aspects

NOC items were identified in four subsections.

“Chapter 14: Plant construction and commissioning” subsection contains a NOC item related to:

* **(Item 3.14.7)** Items that should be included to the specific information provided in the safety analysis report prior to plant construction.

DCD Chapter 14 provides the description of the initial test program. It does not provide a detailed description of the construction process itself. This type of content would need to be developed on a project-specific basis.

“Chapter 15: Safety analysis” subsection contains COM-P items related to:

* **(3.15.1)** Requirement for the Chapter 15 of the safety analysis report to provide a description of the safety analyses performed to assess the safety of the plant in normal operation and in response to postulated initiating events and accident scenarios on the basis of established acceptance criteria. These analyses include deterministic safety analyses of normal operation, anticipated operational occurrences, design basis accidents and design extension conditions, including considerations relating to the event sequences to be ‘practically eliminated’, as well as the probabilistic safety assessment. Analyses to justify specific operator actions can also be included in this chapter of the safety analysis report. The results of these analyses are typically used as a basis for the development of the plant operating procedures and guidelines.

“NOC” is included in the assessment only because Practical Elimination is not directly addressed in the DCD, however, the AP1000 plant design has a very low core damage frequency as described in DCD Chapter 19. Westinghouse has addressed this with respect to WENRA requirements in EPS-GW-GL-701. Westinghouse developed a preliminary position paper on Practical Elimination (APP-GW-GL-900) which will be basis for further documents in this area. More detailed information will be developed to support PSAR requirements if required by the country’s nuclear regulator. In addition, a summary of Severe Accident Mitigation Design Alternatives (SAMDA) is presented in section 1B-5 of the DCD.

**Entire “Chapter 21: Decommissioning and end of life aspects” was assigned to NOC category.**

This topic is not covered in the DCD or in the Reference Plant UFSAR document as it was not included in the NUREG-0800 and RG 1.70 standard SAR content.

Decision to fully follow the approach proposed by the SSG-61 and include the Environmental Aspects Chapter is on the Licensee based on national regulations.

Information required are site-specific and should be prepared by the Licensee.

**Entire “Chapter 21: Decommissioning and end of life aspects” subsection was classified as NOC and COM-P.**

This topic is not covered in the DCD or in the Reference Plant UFSAR document as it was not included in the NUREG-0800 and RG 1.70 standard SAR content.

Decision to fully follow the approach proposed by the SSG-61 and include the Decommissioning Chapter is on the Licensee based on national regulations.

Westinghouse will develop the SAR Input file for this Chapter, which will summarize studies already performed for the AP1000 plant decommissioning.

AP1000 plant is designed to be amenable to decommissioning, with features that both facilitate decommissioning and minimize scope and cost.

## SSG-62 DESIGN OF AUXILIARY SYSTEMS AND SUPPORTING SYSTEMS FOR NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [17] as part of the compliance assessment report [47] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [47] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-62: Design of auxiliary systems and supporting systems for nuclear power plants” [17] is expected to be demonstrated either via compliance with the guidelines “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “IAEA Specific Safety Guide No. SSG-62: Design of auxiliary systems and supporting systems for nuclear power plants” has been assigned to a “**Low Risk**” category.

“IAEA Specific Safety Guide No. SSG-62: Design of auxiliary systems and supporting systems for nuclear power plants” [17] is applicable to the AP1000 plant design, analysis, and licensing aspects for auxiliary and supporting systems. In particular, the following Sections contain guidelines related to plant design:

* Section 3 (General Considerations in Design)
* Section 4 (Specific Considerations in Design)

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### Section 3 (General Considerations in Design)

Large majority of the guidelines presented in **Section 3** of the reference document [17] are met by the Westinghouse “as stated”.

The rest of the guidelines are “Compliant with objective” (CWO) or are a subject of a “Project of site-specific scope” (POS).

The Plant Licensee will address site-specific information related to the security, contingency, and guards training plans. The Plant Licensee will develop the Physical Security Plan, the Training and Qualification Plan, and the Safeguards Contingency Plan. The Plant Licensee will develop and implement a Cyber Security Program.

One CWO item was identified in the “Design basis” subsection and is related to:

* **(Item 3.68)** Preferable consideration of the latest editions of codes and standards for design and construction.

One POS Item was identified in the “Design basis” subsection and is related to:

* **(Item 3.25)** Requirement for the autonomy time of systems supporting safety functions to be longer than the time at which off‑site services are credited. Possibility of crediting the measures taken at the plant and at the site determining this time, provided that the potential for specific hazards to give rise to impacts on several or even all units on the site simultaneously has been considered. Consideration to the adverse conditions and damage caused by the external hazards.

*Refer to* ***Subsection 2.2*** *of the compliance assessment report [47] for more details.*

#### Section 4 (Specific Considerations in Design)

Majority of the design related guidelines presented in **Section 4** of the reference document [17] are met by the Westinghouse “as stated”.

The rest of the design related guidelines are met via CWO or are a subject of a POS.

The emergency offsite communication system, including the crisis management radio system, will be addressed by the Combined License applicant.

Operational requirements fall under the Owner’s responsibility.

Operating requirements for Personal Protection Equipment (PPE) will be developed by the Owner. Individual monitoring of operating personnel is an Owner requirement. However, design considerations to monitor the environment and protect plant personnel are included in the standard plant design.

The processing of radioactive liquid samples is an Owner’s responsibility. Gaseous waste management system and liquid wase management system are not safety-related hence operational test are not required in the DCD as these tests are the Owner’s requirement.

In addition, the Combined License applicant will provide a site-specific cost-benefit analysis to address the requirements of 10 CFR 50 Appendix I, regarding population doses due to liquid effluents. The Combined License applicant will also comply with individual dose limits to members of the public in 10 CFR 50 Appendix I and 10 CFR 20.1301.

Safe load paths should be defined for the movement of heavy loads to minimize the potential for a load drop on irradiated fuel in the reactor vessel or the spent fuel pool or on equipment necessary for achieving or maintaining a safe shutdown of the reactor. This is an Owner’s requirement. Westinghouse provides procedures and layout drawings for the safe movement and handling of cranes.

Plant decommissioning is not addressed in the AP1000 plant DCD as this falls under the Owner’s scope. The basic AP1000 plant design principles minimize the creation of radwaste during operations and decommissioning.

Finally, the Owner will be responsible for ensuring that the design of the system for treatment of solid waste addresses monitoring and removal of surface contamination from the external surfaces of waste packages, measurements to determine the inventory of waste packages (e.g. in terms of radioactivity and mass), and package marking.

CWO items were identified in ten subsections.

“Communication systems” subsection contains a CWO item related to:

* **(Item 4.16)** Requirement for the sound level of the audible alarm system (e.g. sirens) to be higher than the station background noise and should be compatible with the use of personal protective equipment and for an illuminated alarm signal to be used in noisy areas in addition to the audible alarm system.

‘’Heat transport systems’’ subsection contains a CWO item related to:

* **(Item 4.35)** Requirement for the parts of the chilled water system that support a system that fulfils a safety function (safety category 1 or 2) in the event of a design basis accident to have an appropriate safety classification, to meet the corresponding design requirements (e.g. in terms of redundancy, emergency power, protection against internal and external hazards, periodic inspection and testing, maintenance, and quality assurance) and to be designed and fabricated in accordance with acceptable design codes.

‘’Process and post‑accident sampling system’’ subsection contains CWO items related to:

* **(Item 4.47)** Capability of the process and post‑accident sampling system to provide the liquid and gaseous samples necessary in normal operation for analysing the chemical and radiochemical characteristics of the reactor coolant and associated auxiliary systems and supporting systems (e.g. emergency core cooling system; residual heat removal system; chemical and volume control system; for boiling water reactors, the reactor water cleanup system), as well as the containment atmosphere and the secondary system.
* **(Item 4.48)** Requirement for the process and post‑accident sampling system to sample all normal process systems and principal components, including the auxiliary systems and supporting systems necessary for monitoring sample compliance with operational limits and conditions (e.g. sampling of the boron concentration in the accumulators in pressurized water reactors)
* **(Item 4.49)** Capability the process and post‑accident sampling system to take samples during normal operation that provide information to enable the identification of conditions that could jeopardize the integrity of the reactor coolant pressure boundary.
* **(Item 4.51)** Capability the process and post‑accident sampling system, for the spent fuel pool, to detect conditions that could result in excessive radiation levels. Requirement for the system to also provide information for the control of water chemistry necessary for the integrity of the fuel assembly cladding, the internal structures of the spent fuel pool and the cooling systems of the spent fuel pool.
* **(Item 4.52)** Capability the process and post‑accident sampling system to monitor the concentration of soluble neutron absorbers in operational states and in accident conditions.
* **(Item 4.53)** Requirement for the process and post‑accident sampling system to be designed to provide the samples necessary in normal operation to ensure the fulfilment of design requirements and operational needs. Requirement for the design to provide monitoring to demonstrate that the correct water and gas characteristics in the reactor coolant and associated auxiliary systems and supporting systems (e.g. the moderator and its auxiliaries for pressurized heavy water reactors), and in the containment atmosphere and secondary system, are being maintained.
* **(Item 4.54)** Requirement for the process and post‑accident sampling system to be designed to function in design basis accidents and in design extension conditions for which related sampling or monitoring are necessary (e.g. sampling of gas and water within the reactor containment during severe accidents).
* **(Item 4.58)** Requirement for the process and post‑accident sampling system to monitor variables and systems (in normal operation) that ensure safety, including variables and systems that can affect the fission process and the integrity of the reactor core and of the reactor coolant pressure boundary.
* **(Item 4.59)** Functions to be performed the process and post‑accident sampling system.
* **(Item 4.65)** Requirement for sampling lines connected to systems located inside the containment to be provided with appropriate features for automatic isolation of the containment; Requirement for sampling lines from the reactor coolant system to have at least two isolation valves; Requirement for these containment isolation features to be safety classified (based on their safety function of safety category 1) and to meet the corresponding design requirements (e.g. in terms of redundancy, emergency power, protection against internal and external hazards, periodic inspection and testing, maintenance, and quality assurance) and to be designed and fabricated in accordance with acceptable design codes; Possibility of reopening the primary coolant sampling lines to check the boron concentration, to measure the radioactivity and to determine the composition of fission products, once the radiological conditions at the sampling locations allow (if necessary, subject to the implementation of specific precautions).
* **(Item 4.69)** Requirement for the design of the process and post‑accident sampling system to allow the collection and analysis of highly radioactive samples after an accident. This includes samples from the reactor coolant, the containment sump and the containment atmosphere, for example to provide information on the pH of recirculating water and the concentration of hydrogen and fission products within the containment atmosphere.
* **(Item 4.71)** Re-injecting highly radioactive samples into the containment if there is a risk of exceeding the capability of the plant to manage the samples as radioactive waste, for cases when an analysis is performed outside the containment.

‘’Process radiation monitoring system’’ subsection contains CWO items related to:

* **(Item 4.81)** Requirement for the radiation monitoring system to provide monitoring to enable the assessment of the radiological release into the containment atmosphere for some post‑accident conditions, such as after a loss of coolant accident or a severe accident.
* **(Item 4.82)** Continuous monitoring ofthe atmosphere of the containment and other buildings where radioactive releases could occur to allow actions to be taken and to trigger an alarm for the evacuation of personnel, in particular, in the event of a fuel handling accident. Monitoring surface contamination in all areas containing large amounts of radioactive liquids and solid radioactive waste.
* **(Item 4.91)** Providing continuous measurements of dose rate for each sump that could collect highly contaminated water. Requirement for the automatic isolation of the discharge of sumps to the radioactive waste processing system to occur if the dose rate exceeds a pre‑set threshold.

‘’Compressed air system’’ subsection contains CWO items related to:

* **(Item 4.98)** Requirement forthe part of the compressed air system that supplies air to items important to safety to be designed to ensure that it functions during adverse environmental phenomena, in anticipated operational occurrences (including loss of off‑site power) and in accident conditions (in particular a loss of coolant accident or a main steam line break); Requirement to take into account any increased internal pressures caused by high temperatures inside the containment during design basis accidents in the design of these tanks, for cases where reserve air supply tanks are installed inside the containment.
* **(Item 4.102)** Requirement for the compressed air system to be designed to avoid a containment bypass or pressurization of the containment and for systems located inside the containment that are needed in the long term after an accident not to depend on compressed air systems to fulfil their safety functions; Giving considerations to the installation of a dedicated post‑accident compressed air system to supply instruments inside the containment with air exhausted from the containment to avoid gradual pressurization of the containment due to the leakage of compressed air systems.
* **(Item 4.106)** Requirement for the routing of the pipework of the compressed air system to provide for the draining of condensable gases and vapours; Avoiding the possibility of liquid plugs by providing an adequate slope in the routing of the pipework.

‘’Heating, ventilation and air‑conditioning systems’’ subsection contains CWO items related to:

* **(Item 4.118)** Requirement for he design of the heating, ventilation and air‑conditioning systems that contribute to the limitation of radioactive releases to filter the exhausted air using pre‑filters, HEPA filters and, if necessary, iodine filters before the air is discharged through the stack; Requirement for the efficiency of the HEPA and iodine filters to be commensurate with the authorized limits on discharges in normal operation and anticipated operational occurrences and with the acceptable limits on discharges during accident conditions.
* **(Item 4.129)** Requirement forthe emergency core cooling system rooms, the residual heat removal system rooms and the containment spray system rooms to be considered areas where the risk of internal exposure from radioiodine is significant during accident conditions.
* **(Item 4.131)** Capability ofThe parts of the engineered safety feature ventilation system that are not important to safety to automatically isolate in the event of accident conditions.
* **(Item 4.140)** Requirement for the functions of the ventilation system for the effluent treatment building to be to maintain suitable ambient conditions for personnel access and for the correct operation of equipment during normal operation.
* **(Item 4.141)** Requirement for the ventilation system for the effluent treatment building to ensure the confinement of radioactive material in the effluent treatment building in accident conditions, including conditions caused by an SL‑2 design basis earthquake. Depending on the results of the safety analysis, the confinement of radioactive material could be based on static confinement or dynamic confinement.
* **(Item 4.143)** Requirement forthe ventilation system for the effluent treatment building to be designed such that in normal operation the level of radioactivity in gaseous releases to the environment is below the authorized limits and is as low as reasonably achievable; Requirement for the components of the ventilation system for the effluent treatment building that ensure the control of radioactive releases to have an appropriate safety classification (based on their safety function of at least safety category 3).
* **(Item 4.144)** Requirement for the ventilation system for the effluent treatment building to be designed such that the airflow is from areas of the effluent treatment building that are not designated as controlled areas towards the controlled area.
* **(Item 4.145)** Requirement for design provisions (e.g. means of isolation or intake and exhaust ducts that are resistant to earthquakes) to be made if the confinement of radioactive material within the controlled area of the effluent treatment building in the event of an SL‑2 design basis earthquake is ensured by static confinement.
* **(Item 4.148)** Requirement for the containment sweeping ventilation system to ensure the confinement of radioactive material in the event of a fuel handling accident within the containment.
* **(Item 4.151)** Requirement for the containment sweeping ventilation system should limit radioactive releases to the environment to meet the safety objectives in the event of a fuel handling accident within the containment; Including an outage with an open containment to the scenarios to be considered with respect to the design of this system.
* **(Item 4.152)** Requirement for the design of the containment sweeping ventilation system to take into account that, during the transfer of spent fuel in the fuel storage pool, damaged fuel cladding could cause releases of radioactive gases and aerosols in some areas of the containment; Objectives to be achieved by the containment sweeping ventilation system.
* **(Item 4.156)** Requirement for the operation of heating, ventilation and air‑conditioning systems in essential areas of the electrical building in the event of a station blackout to be ensured.

‘’Lighting and emergency lighting systems’’ subsection contains a CWO item related to:

* **(Item 4.175)** Requirement for emergency lighting to be provided in areas where items important to safety are located, as well as in the access and rescue routes to these areas; Areas included.

‘’Overhead lifting equipment’’ subsection contains a CWO item related to:

* **(Item 4.191)** Requirement for all overhead lifting equipment to be equipped with a load weighing device that has a display that is always visible to the operator of the equipment in order to prevent the lifting of excessive loads. Requirement for this weighing device to include an overload protection system.

‘’Supporting systems for the emergency power supply and the alternate power source’’ subsection contains a CWO item related to:

* **(Item 4.235)** Design basis items to be included for for any diesel engine or other prime mover that provides an emergency power supply to items important to safety.

‘’ Other systems’’ subsection contains CWO items related to:

* **(Item 4.272)** Requirement for the equipment and floor drainage system to have the capability (in accident conditions) to reinject highly contaminated liquids from the auxiliary buildings or secondary containment into the containment if the level of radioactivity in the effluent is too high to be treated in the short term (i.e. if storage would be needed before the treatment) or if the volume of fluids exceeds the waste treatment capacity.
* **(Item 4.273)** Requirement for the equipment and floor drainage system to help reduce the retention of activity in the nuclear island buildings and limit discharges to the environment by monitoring levels of radioactivity during normal operation. The equipment and floor drainage system could contribute directly to the safety functions that are fulfilled by protecting against the effects of internal flooding and explosion (e.g. the prevention of hydrogen explosion from hydrogenated effluents).
* **(Item 4.275)** Requirement for the equipment and floor drainage system to have sufficient capability to collect, treat and dispose of radioactive and non‑radioactive liquid effluents in all plant states; Requirement for the radioactive and non‑radioactive liquids to be collected separately.
* **(Item 4.276)** Requirement for components of the equipment and floor drainage system to be classified on the basis of their functions and their role as barriers and to meet the corresponding design requirements (e.g. in terms of redundancy, emergency power, protection against internal and external hazards, periodic inspection and testing, maintenance, and quality assurance); List of equipment that is usually classified.
* **(Item 4.278)** Requirement for the components of the equipment and floor drainage system that carry radioactive material and whose failure would lead to off‑site radiological consequences to be considered items important to safety and to have a corresponding safety classification; Requirement for the parts of the system that are considered items important to safety to be capable of being isolated from the parts of the system that are not important to safety.
* **(Item 4.279)** Requirement for the drainage capacity of the equipment and floor drainage system to be sufficient to ensure that safety functions continue to be fulfilled in the event of flooding from pipe breaks, tank leak and other potential sources (e.g. an earthquake causing a leak from a tank of non‑seismic design).
* **(Item 4.285)** Requirement for the equipment and floor drainage system to, as far as practicable, be independent of similar equipment in other fire compartments in order to maintain the operability of the equipment and floor drainage system in the event of fire in adjacent fire compartments.

POS items were identified in three subsections.

“Communication systems” subsection contains a POS item related to:

* **(Item 4.20)** Requirement to provideA wireless system with the capability to ensure normal and emergency communications with on‑site personnel and off‑site personnel; Requirement for the wireless system to be independent of the main telephone system and the secondary telephone system and to be tested to determine whether there are locations on the site where a signal cannot be received (‘dead zones’); Requirement for areas of the plant in which wireless radio transmission could cause serious electromagnetic interference and have consequences for the plant, for example plant trips, to be clearly marked in the plant as radio exclusion areas.

‘’Heating, ventilation and air‑conditioning systems’’ subsection contains a POS item related to:

* **(Item 4.118)** Requirement for he design of the heating, ventilation and air‑conditioning systems that contribute to the limitation of radioactive releases to filter the exhausted air using pre‑filters, HEPA filters and, if necessary, iodine filters before the air is discharged through the stack; Requirement for the efficiency of the HEPA and iodine filters to be commensurate with the authorized limits on discharges in normal operation and anticipated operational occurrences and with the acceptable limits on discharges during accident conditions.

‘’Overhead lifting equipment’’ subsection contains POS items related to:

* **(Item 4.191)** Requirement for all overhead lifting equipment to be equipped with a load weighing device that has a display that is always visible to the operator of the equipment in order to prevent the lifting of excessive loads. Requirement for this weighing device to include an overload protection system.
* **(Item 4.196)** Requirement for handling equipment to be tested to at least its maximum expected load before commissioning; Requirement for periodic inspections and tests during normal operation to be carried out to ensure and verify the operation of safety devices, including the upper limit switch, overspeed interlock, overload interlock and restricted areas interlock.
* **(Item 4.198)** Requirement for the overhead heavy load handling systems that are credited in the preliminary decommissioning plan to have a design life and specific design provisions commensurate with the expected decommissioning activities.

*Refer to* ***Subsection 2.3*** *of the compliance assessment report [47] for more details.*

### Non Compliance

No “Non Compliances” have been identified in the assessment [47].

## SSG-63 DESIGN OF FUEL HANDLING AND STORAGE SYSTEMS FOR NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [18] as part of the compliance assessment report [48] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [48] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-63: Design of fuel handling and storage systems for nuclear power plants” [18] is expected to be demonstrated either via compliance with the guidelines “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “IAEA Specific Safety Guide No. SSG-63: Design of fuel handling and storage systems for nuclear power plants” has been assigned to a “**Low Risk**” category.

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### SSG 63 - Section 2 (Design objectives and design approach):

Westinghouse shows full compliance to the guidelines addressing design objectives and design approach in **Section 2** of the reference document [18].

One “Project of site-specific scope” (POS) item identified in this section and is related to:

* **(Item 2.16)** The design of fuel handling and storage systems and facilitation of the application and maintenance of IAEA safeguards, and the State system of accounting for, and control of, nuclear material.

*Refer to* ***Subsection 2.2*** *of the compliance assessment report [48] for more details.*

#### SSG 63 - Section 3 (Design basis for structures, systems and components of fuel storage):

Subsections of **Section 3** of the reference document [18] that are addressing “general design basis for SSCs of fuel storage”, “Defense in depth”, “Safety functions”, “Postulated initiating events”, “Internal hazards”, “Design limits”, “Reliability”, “Prevention of criticality”, “Monitoring”, “Design of water purification system for spent fuel pool” are fully met by Westinghouse.

Subsections of the guidelines that are addressing “External hazards”, “Plant conditions to be taken into account in design”, “Structural integrity”, “Environmental qualification”, “Radiation Protection” are fully met by Westinghouse, but also are partially responsibility of the Owner.

The operation on SNF casks and on-site dry storage , ensuring safety during operation , maintenance of structures and components throughout lifetime of the plant , execution of an inspection program during the plant lifetime , addressing any possibility of site-specific external hazards and maintaining systems operability during and after the plant lifetime, etc. are responsibility of the Owner.

“Compliant with objective” (CWO) items were identified in the three subsections.

“Safety classification” subsection contains CWO related to:

* **(Item 3.93)** SSCs safety classification.

“Environmental qualification” subsection contains CWO related to:

* **(Item 3.97)** Factors considered in the environmental qualification.

“Illumination equipment” subsection contains CWOs related to:

* **(Item 3.139)** Resistance to impact and thermal shocks.
* **(Item 3.140)** Selection of lighting technologies with a high temperature spectrum

POS items was identified in “Radiation protection” subsection and is related to:

* **(Item 3.112)** Additional shielding to limit the exposure of operating personnel from the handling and storage of fresh fuel containing fissionable material recovered by reprocessing.

*Refer to* ***Subsection 2.3*** *of the compliance assessment report [48] for more details.*

#### SSG 63 - Section 4 (Design basis for equipment and components of fuel handling systems):

Compliance with **Section 4** of the reference document [18] is fully demonstrated by Westinghouse.

Some of the guidelines were identified as non-applicable as they refer to reactor and fuel types different from the ones of AP1000 Standard Design.

#### SSG 63 - Section 5 (Design basis for equipment used for inspection and repair of irradiated fuel, handling of damaged fuel, and handling and storage of irradiated core components):

Design related guidelines presented in the “reusable reactor items”, and part of the “irradiated core components” subsection of **Section 5** of the reference document [18] are fully met by Westinghouse. The rest is responsibility of the Owner.

#### SSG 63 - Section 6 (Handling of fuel casks):

Guidelines presented **Section 6** of the reference document [18] are met by Westinghouse.

Guidelines addressing items such as providing spent fuel shipping cask, vehicle for cask movement offsite, cask testing equipment, provision of appropriately designed spent fuel casks, administrative means of ensuring sufficient fuel cooling time before loading into the cask, appropriate procedures for the inspection, testing and maintenance of the crane during the plant lifetime, design of vehicles for cask movement, the equipment required for monitoring the radiation of the casks, ensuring the external surface contamination of the cask is met with the requirements are responsibility of the Owner.

### Non Compliance

No “Non Compliances” have been identified in the assessment [48].

## SSG-64 PROTECTION AGAINST INTERNAL HAZARDS IN THE DESIGN OF NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [19] as part of the compliance assessment report [49] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [49] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-64: Protection Against Internal Hazards in the Design of Nuclear Power Plants” [19] is expected to be demonstrated but will require project-specific egress studies to be performed to reflect the final design of the AP1000 units for a specific site. Since no non-compliances have been identified and no design changes are anticipated, “IAEA Specific Safety Guide No. SSG-64: Protection Against Internal Hazards in the Design of Nuclear Power Plants” has been assigned to a “**Medium Risk**” category.

“Medium Risk” category has been assigned due to guidelines reflected in **(Item II.22)** related to escape routes for personnel and general conditions to be met. Project-specific egress studies will need to be performed to reflect the final design of the AP1000 units for a specific site and taking into consideration local laws.

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### SSG 64 - Section 2 (General considerations):

All the guidelines presented in **Section 2** of the reference document [19] are met by Westinghouse.

One “Project of site-specific scope” (POS) item was identified and is related to:

* **(Item 2.13)** General considerations on defense in depth and site hazards.

Identification of site-specific hazards, proper surveillance and inspections of the SSCs are the responsibility of the Owner.

#### SSG 64 - Section 3 (General design recommendations for protection against internal hazards):

All the guidelines presented in **Section 3** of the reference document [19] are met by Westinghouse.

Several POS items were identified and are related to:

* **(Item 3.6)** Identification of site-specific hazards.
* **(Item 3.7)** Possible combinations of internal–internal and internal–external hazards.
* **(Item 3.32)** Safety consequences of internal hazard in one unit for a neighboring operating unit or other installations on the site.

Identification of site-specific hazards, implementation of administrative measures to reduce the frequency and potential magnitude of the hazards and their effects on SSCs important to safety, ensuring that the assessment (Demonstration that the internal hazards relevant to the design of the nuclear power plant have been considered, and that provisions for prevention and mitigation have been designed with sufficient safety margins to address the uncertainties in the identification and characterization of internal hazards and their effects, as well as for the avoidance of cliff edge effects) will be updated before initial loading of the reactor with nuclear fuel, and during plant operation are responsibilities of the Owner.

*Refer to* ***Subsection 2.3*** *of the compliance assessment report [49]* *for more details.*

#### SSG 64 - Section 4 (Recommendations for specific internal hazards):

Most of the guidelines in **Section 4** of the reference document [19] were met “as stated”. The rest were classified as “Compliant with objective” or belong to a “Project or site-specific scope”.

“Compliant with objective” (CWO) items were identified in the three subsections.

“Internal fires” subsection contains CWOs related to:

* **(Item 4.14)** Providing of spray guards for systems containing pressurized combustible liquids.
* **(Item 4.23)** Independency of fire detection systems, fire extinguishing systems and support systems.
* **(Item 4.24)** Combination of fire protection systems to control the fire as well as elements to be considered in ensuring of an adequate level of protection for fire compartments.
* **(Item 4.39)** Ensuring ventilation systems do not compromise building compartmentation nor compromise the availability of redundant divisions of safety systems.
* **(Item 4.43)** Precautions related to usage of combustible filters.
* **(Item 4.56)** Fire protection guidelines regarding flammable materials in turbine building.
* **(Item 4.57)** Fire protection of safety features for design extension conditions that are needed to function in the long term under such conditions.

“Pipe breaks (Pipe whip and jet effect and flooding)” subsection contains CWOs related to:

* **(Item 4.112)** Locations of postulated failure.
* **(Item 4.113)** Locations of postulated failure for small diameter piping systems, which are sensitive to vibration induced failure and to rupture due to external forces.

“Internal flooding” subsection contains CWOs related to:

* **(Item 4.151)** Doors failure under flooding conditions.

POS items were identified in the three subsections.

“Internal fires” subsection contains POS items related to:

* **(Item 4.15)** Potential ignition sources arising from plant systems and equipment.

“Heavy load drop” subsection contains POS items related to:

* **(Item 4.181)** Measures to prevent dropped loads.

“Release of hazardous substances inside the plant” subsection contains POS items related to:

* **(Item 4.203)** Revision of inventory of hazardous materials .
* **(Item 4.204)** Establishing of a list of the hazardous substances that could potentially be released, using hazard identification process.
* **(Item 4.205)** Requirement to the contents of a list of the hazardous substances.

*Refer to* ***Subsection 2.4*** *of the compliance assessment report [49]* *for more details.*

#### SSG 64 - Appendix I (Hazard combinations):

All the guidelines in **Appendix I** to the reference document [19] are fully met by Westinghouse.

Several POS items were identified in this section and are related to:

* **(Item I.2)** Combined hazards.
* **(Item I.6)** Screening criteria for a hazard combination list.

*Refer to* ***Subsection 2.5*** *of the compliance assessment report [49]* *for more details.*

#### SSG 64 - Appendix II (Detailed guidance on internal fires):

Guidelines in Appendix II to the reference document [19] that fall under the scope of responsibility of Westinghouse are met. Meeting the rest of the guidelines is responsibility of the Owner.

Several CWO items were identified and are related to:

* **(Item II.49)** Design considerations for any fixed extinguishing system that is solely manually actuated.
* **(Item II.60)** Required provisions for the fire hydrant system for the reactor.
* **(Item II.65)** Considerations for the main loop of the water supply system for the fire extinguishing equipment and water distribution water to the fire extinguishing equipment.

POS items were identified in this section and are related to:

* **(Item II.22)** Escape routes for personnel and general conditions to be met.
* **(Item II.28)** Electrical cable fire testing.
* **(Item II.62)** Hydrant hose and standpipe riser connections.
* **(Item II.63)** Fire protection accessories and their compatibility with those of external fire services.

*Refer to* ***Subsection 2.6*** *of the compliance assessment report [49]* *for more details.*

### Non Compliance

No “Non Compliances” have been identified in the assessment [49].

## SSG-67 SEISMIC DESIGN FOR NUCLEAR INSTALLATIONS

Following summary reflects results of the assessment conducted against reference [20] as part of the compliance assessment report [50] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [50] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-67: Seismic design for nuclear installations” [20] is expected to be demonstrated either via compliance with the guidelines “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “IAEA Specific Safety Guide No. SSG-67: Seismic design for nuclear installations” has been assigned to a “**Low Risk**” category.

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### Section 2 (Requirements for seismic design and general seismic design aspects)

Westinghouse complies with relevant guidelines presented in **Section 2** of the reference document [20], while rest of the guidelines presented in this Section belong to the Owner’s or site-specific scope.

The AP1000 plant has been reviewed for conformance with SSR-2/1 Safety Standard meeting all the requirements set for seismic design of nuclear power plants. Seismic design of facilities other than nuclear power plants is not a subject of this document.

The AP1000 plant is designed for an earthquake, geotechnical, seismic and tectonic characteristics specified in DCD Section 2.5. The AP1000 design earthquake is referred to as the AP1000 Certified Seismic Design Response Spectra (CSDRS) and was developed using the U.S. Regulatory Guide 1.60 response spectra. The items important to safety are designed so that hazards resulting from such earthquake won’t impact the plant’s safety.

The AP1000 plant Owner will address the site-specific information related to the vibratory grounds motion aspects of the site and region, identify the potential seismic hazards and assess if the site-specific vibratory ground motion aspects fall within the general seismic criteria of the AP1000, so that no additional design evaluation, per U.S. licensing basis, is needed (unless otherwise specified by other national codes and regulatory bodies). Detailed design, including seismic design is considered in accordance with site-specific project input data by the plant Owner.

“Project of site-specific scope” (POS) items were identified in four subsections.

“Requirements for seismic design and general seismic design aspects” subsection contains a POS item related to:

* **(Item 2.1)** Evaluation of seismic hazards associated with a site for a nuclear installation which serves as an input to the seismic design of the installation.

“External hazards” subsection contains a POS item related to:

* **(Item 2.3)** Statements from SSR‑2/1 (Rev. 1) with regard to considering external hazards in the design of nuclear power plants.

“Engineering design rules” subsection contains a POS item related to:

* **(Item 2.4)** Statement from SSR‑2/1 (Rev. 1) addressing engineering design rules.

“Other seismic design aspects” subsection contains POS items related to:

* **(Item 2.10)** Basing seismic design of items important to safety on the seismic hazards determined during the site evaluation process for the nuclear installation. Assessing the adequacy of the design basis earthquake for the nuclear installation using the site specific vibratory ground motions assessment which uses deterministic and/or probabilistic approaches.
* **(Item 2.12)** Specific aspects to be considered in the seismic design of nuclear installations
* **(Item 2.15)** Requirements on how The design process for a nuclear installation should be conducted. Integration of The seismic design process into the management system.

*Refer to* ***Subsection 2.3*** *of the compliance assessment report [50] for more details.*

#### Section 3 (Input for seismic design)

Westinghouse complies with the guidelines presented in **Section 3** of the reference document [20] either “as stated” or via compliance with their objectives.

The AP1000 plant Owner will address site-specific information related to basic geological, seismological, and geotechnical engineering of the site and the region, as discussed in DCD Section 2.5. Additionally, a reconciliation of the seismic analyses described in DCD Subsection 3.7.2 for detail design changes will be performed as provided in DCD Subsection 3.7.5.4, based on the site-specific ground motion input.

DCD Subsection 2.5.4 addresses stability and uniformity of subsurface of materials and foundations and identifies site characteristics, and investigation programs to be identified, analyzed, and established by the Owner.

The standard AP1000 plant is conservatively designed to protect the plant against a design basis SSE with a PGA of 0.3g and the Certified Seismic Design Response Spectra (CSDRS) specified in DCD section 3.7.1.1, DCD Figures 3.7.1-1 and 3.7.1.-2. The seismic ground design response spectra are based on Regulatory Guide 1.60 and have been enhanced to include the high frequency characteristics of the eastern US. The seismic design of the AP1000 plant SSCs and applicable codes and standards are provided in detail in DCD Section 3.7.

The operating basis earthquake (OBE), has been eliminated as a design requirement for the AP1000 plant. Low-level seismic effects are included in the design and qualification of certain equipment potentially sensitive to a number of such events based on a percentage of the responses calculated for the SSE.

Per the U.S. licensing basis, the standard AP1000 plant has been designed for 6 generic soil cases (site types) ranging from hard rock to soft soil sites. For specific information on the governing parameters of the six sites considered refer to DCD Subsection 3.7.1.4.

Several “Compliant with objective” (CWO) items were identified in “Design basis earthquake” subsection and are related to:

* **(Item 3.14)** Site categorization used in performing of seismic site response analyses
* **(Item 3.15)** Performing seismic site response analysis for Type 2 and Type 3 sites.
* **(Item 3.20)** SL-1 level earthquakes (The operating basis earthquake (OBE)).
* **(Item 3.25)** Definition of SL-1 level.

POS items were identified in five subsections.

“General concepts of seismic design” subsection contains POS items related to:

* **(Item 3.4)** Deeming site unsuitable If the characteristics of some geological and geotechnical hazards are such that satisfactory engineering solutions to protect against them have not been identified.
* **(Item 3.5)** Steps to be considered in the seismic design process.

“Design basis earthquake” subsection contains POS items related to:

* **(Item 3.6)** If a generic seismic design basis is used, it should be shown to envelop the site specific seismic ground motion. Otherwise, the design will need to be reassessed with a design basis earthquake enveloping the site specific earthquake.
* **(Item 3.7)** Availability of the seismic hazard assessment from the specific site characterization.
* **(Item 3.10)** Availability of the site specific static and dynamic properties of the soil parameters at the site area.
* **(Item 3.11)** Carrying out a detailed programme of geophysical and geotechnical investigations and preparation of a detailed subsurface exploration and testing programme.
* **(Item 3.12)** Data that should be available as a result of the geological, geophysical and geotechnical investigations conducted at the site area and at the location of the buildings and structures of the nuclear installation.
* **(Item 3.13)** Preliminary and final site response analyses. Assessment of final vibratory ground motions at the control point specified by the designer of the evaluation and based on the seismic hazard assessment performed at the bedrock level.
* **(Item 3.16)** Two approaches to properly considering the geological and geotechnical specific soil conditions at a site as part of the estimation of the seismic vibratory ground motion.
* **(Item 3.17)** Considerations regarding the approaches for the estimation of the seismic vibratory ground motion.
* **(Item 3.18)** Determining the design basis earthquake.
* **(Item 3.21)** Definition of SL-2 earthquake level on the basis of the results and parameters obtained from the seismic hazard assessment (see para. 3.7), in accordance with specific criteria established by the regulatory body to achieve a certain target level for the annual frequency of exceedance for SL‑2.
* **(Item 3.22)** Considerations on SL-2 level calculations using probabilistic approach.
* **(Item 3.23)** Making an estimate of the associated return period of the calculated earthquake level when using deterministic approach.
* **(Item 3.24)** Design conservatism regarding the design basis earthquake level.
* **(Item 3.25)** Definition of SL-1 level.
* **(Item 3.26)** Requirement to nuclear installation design to be able to withstand a minimum earthquake level. Definition of the “minimum earthquake level”.

“Beyond design basis earthquake” subsection contains a POS item related to:

* **(Item 3.29)** Methods to determine the beyond design basis earthquake and the associated loads.

“Seismic categorization for structures, systems and components” subsection contains POS items related to:

* **(Item 3.34)** Requirement to the physical barriers designed to protect the installation against the effects of internal or external hazards other than seismic hazards (e.g. fires, floods) to remain functional and maintain their integrity after an SL‑2 earthquake.
* **(Item 3.39)** Correlation between safety and seismic categories. Aspects that should be considered and determined as input for the seismic design to establish the limiting acceptable conditions.

“Selection of seismic design and qualification standards” subsection contains POS items related to:

* **(Item 3.41)** Specification of corresponding engineering design rules once the seismic categories of the items in the nuclear installation have been established.
* **(Item 3.42)** Establishing of consistent acceptance criteria and using good engineering practices to provide consistency in the application of selected codes and standards in seismic design.
* **(Item 3.43)** Performing of an analysis and evaluation of the codes, norms and standards to be applied in the design, fabrication and construction of the nuclear installation and documenting it as part of the management system.

*Refer to* ***Subsection 2.4*** *of the compliance assessment report [50] for more details.*

#### Section 4 (Seismic design of structures, systems and components)

Large majority of the guidelinespresented in **Section 4** of the reference document [20] are met “as stated” by Westinghouse. Several guidelines were met via compliance with their objectives and some require site-specific scope to be performed.

The standard AP1000 plant design was evaluated for the most suitable layout of the installation in terms of seismic design. The standard plant seismic design includes evaluation of interaction with applicable adjacent structures. Site-specific changes from the standard design are evaluated on a case-by-case basis using the same methodology as described in the DCD.

SC-I structures are designed to maintain both functionality and integrity under seismic loading within the design basis. SC-II structures are designed so that an SSE does not cause unacceptable structural failure of, or interaction with, SC-I items that could degrade the functioning of a safety significant SSC to an unacceptable level, or could result in incapacitating injury to occupants of the Main Control Room (MCR). Non-nuclear seismic structures will not cause unacceptable interaction with higher class structures as well.

The AP1000 plant includes all Seismic Category I structures, systems and components in the scope of the design certification. See DCD Section 3.2 for a list of seismic classification of the AP1000 plant structures. Refer to DCD Section 2.5 on the requirements for the foundations and earth structures under the AP1000 structures. Site-specific information regarding the underlying site conditions and geologic features will be addressed. This information will include site topographical features, as well as the locations of Seismic Category I structures.

The AP1000 plant Owner will address site-specific information about the static and dynamic stability of soil and rock slopes, and site-specific information about the static and dynamic stability of embankments and dams, the failure of which could adversely affect the nuclear island.

The AP1000 plant design does not include seismic isolation of the SSCs in the Nuclear Island. Seismic isolation is only included in the foundation design of the condenser, which does not affect the safety of AP1000 nuclear power plant.

SC-I and SC-II equipment including its anchorage is qualified for seismic loads. The anchorage of non-seismic systems and components shall be designed using the seismic analysis methods given in the Uniform Building Code (Reference [7] of the compliance assessment report [50]). The AP1000 Seismic Design Criteria allows the same criteria to be applied for design of anchorage for non-seismic systems and components located in Seismic Category I or II building. For classification of AP1000 plant mechanical equipment and applicable codes and standards refer to DCD Sections 3.2, 3.9, 3.7 and 3.10.

The AP1000 does not utilize safety-related buried piping. Any site-specific information on buried piping and its design will be provided by the Owner to ensure no additional hazards to the plant are identified.

For compliance of AP1000 plant SSCs to the specific design requirements refer to specific compliance justifications in Section 2.5 of the compliance assessment report [50].

Several CWO items were identified two subsections.

“Engineered earth structures and buried structures” subsection contains a CWO item related to:

* **(Item 4.13)** Consistency of the seismic design of engineered earth structures and buried structures with SSG-30.

“Buried pipes” subsection contains CWO items related to:

* **(Item 4.31)** Recommendations to be followed by the design.
* **(Item 4.32)** Considerations for the design of buried pipes.

Project or site-specific (POS) items were identified in six subsections.

“Seismic design of structures, systems and components” subsection contains a POS item related to:

* **(Item 4.1)** Basing all procedures for seismic design on a good understanding of the consequences of past destructive earthquakes, adopting and realistically applying this knowledge.

“Layout of the installation” subsection contains a POS item related to:

* **(Item 4.2)** Establishing the layout of the installation early in the design stage and aiming to achieve the most suitable solution for the seismic design.

“Buildings and civil structures” subsection contains a POS item related to:

* **(Item 4.12)** Massive mat foundations.

“Engineered earth structures and buried structures” subsection contains POS items related to:

* **(Item 4.13)** Consistency of the seismic design of engineered earth structures and buried structures with SSG-30.
* **(Item 4.14)** Engineered earth structures important to safety that may be encountered at nuclear installation sites.
* **(Item 4.15)** Seismic related effects to be taken into account by the seismic design of earth structures and buried structures.

“Buried pipes” subsection contains POS items related to:

* **(Item 4.31)** Recommendations to be followed by the design.
* **(Item 4.32)** Considerations for the design of buried pipes.

“Seismic capacity” subsection contains POS items related to:

* **(Item 4.46)** Stringency of the acceptance criteria for seismic category 3 SSCs.
* **(Item 4.49)** Determining the seismic capacities associated with failures of the soil.

*Refer to* ***Subsection 2.5*** *of the compliance assessment report [50] for more details.*

#### Section 5 (Seismic analysis)

Large majority of the guidelinespresented in **Section 5** of the reference document [20] are met “as stated” by Westinghouse. One guideline was met via compliance with its objective and several guidelines require site-specific scope to be performed.

Seismic analyses of the AP1000 plant – using response spectra analysis, the equivalent static acceleration method, the mode superposition time-history method, and the complex frequency response analysis method – are performed for the safe shutdown earthquake (SSE) to determine the seismic force distribution for use in the design of the nuclear island structures, and to develop in-structure seismic responses (accelerations, displacements, and floor response spectra) for use in the analysis and design of seismic subsystems. For more information on seismic analyses refer to DCD Section 3.7 and Appendix 3G on Nuclear Island Seismic Analyses. Refer to Table 3G.1-1 for the summary of models and analysis methods.

Seismic systems are defined, according to SRP 3.7.2, Section II.3.a, as the Seismic Category I structures that are considered in conjunction with their foundation and supporting media to form a soil-structure interaction model. The SSI analyses generate a set of in-structure responses (design member forces, nodal accelerations, nodal displacements, and floor response spectra), which are used in the design and analysis of Seismic Category I structures, components, and seismic subsystems.

In addition to the design basis seismic analysis, the AP1000 plant nuclear island robustness was evaluated for a seismic margin assessment which extends to 67% above the SSE design basis PGA of 0.3g. This larger seismic event is referred to in the U.S. as the Review Level Earthquake (RLE), which has a PGA level for the AP1000 plant of 0.5g. A more detailed discussion of the seismic margin assessment is provided in Section 19.55 of the DCD. Many of the components presented in DCD Section 19.55 have HCLPF in excess of 0.5g.

Analyses including adjacent buildings showed that the effect of the adjacent buildings on the Nuclear Island response was small. Based on this, the 3D SASSI analysis of the Nuclear Island can be performed without adjacent buildings. The Nuclear Island does affect the response of the adjacent buildings and the results of the 2D SASSI analyses are used for design of the adjacent

buildings for the AP1000 plant. For the details on modelling and analysis see DCD Subsection 3.7.2.8.4 and Appendix 3G.

One CWO item was identified in “Structural response” subsection and is related to:

* **(Item 5.8)** Stiffness values for seismically isolated structures and variation in the stiffness of the isolators during the design life of the structure.

POS items were identified in three subsections.

“Site response analysis” subsection contains a POS item related to:

* **(Item 5.2)** Performing ground (free field) response analysis for soil and soft rock sites.

“Structural response” subsection contains POS items related to:

* **(Item 5.4)** Recommendations applied to calculations of structural response.
* **(Item 5.5)** Structural response calculation on the basis of the simultaneous application of the two horizontal components and one vertical component of seismic input, provided that the components of the seismic input are demonstrated to be statistically independent.
* **(Item 5.15)** Obtaining in‑structure response spectra. Number of necessary analyses for each soil–structure configuration.

“Dynamic soil–structure interaction” subsection contains POS items related to:

* **(Item 5.17)** Aspects to be assessed to identifyacceptable models and analysis procedures when consideration of soil–structure interactioneffects is necessary.
* **(Item 5.25)** Required level of soil discretization.

*Refer to* ***Subsection 2.6*** *of the compliance assessment report [50] for more details.*

#### Section 6 (Seismic qualification)

All the guidelines presented in **Section 6** of the reference document [20] are met “as stated” by Westinghouse.

Inspection and maintenance program of the equipment is to be specified by the Owner basing on the input on aging degradation provided by Westinghouse.

#### Section 7 (Seismic margin to be achieved by the design)

All the guidelinespresented in **Section 7** of the reference document [20], except for one, are met “as stated” by Westinghouse. One guideline is met via compliance with its objective and several guidelines require site-specific inputs by Owner.

In addition to the design basis seismic analysis, the AP1000 plant nuclear island robustness was evaluated for a seismic margin assessment which extends to 67% above the SSE design basis PGA of 0.3g. This larger seismic event is referred to in the U.S. as the Review Level Earthquake (RLE), which has a PGA level for the AP1000 plant of 0.5g. A more detailed discussion of the seismic margin assessment is provided in Section 19.55 of the DCD. Many of the components presented in DCD Section 19.55 have HCLPF in excess of 0.5g which proves robustness in avoiding the cliff edge effects and significantly limits early radioactive releases following an earthquake.

The obtained seismic margin (HCLPF capacity) and fragilities for all sequences leading to core damage or large early release for the AP1000 standard plant is sufficient enough to bring the plant to a safe, stable condition following a Review Level Earthquake. Site-specific Seismic Margin Assessment and meeting the requirement of a specific regulatory body is the Owner’s scope.

One CWO item was identified in “Procedures to assess the seismic margin” subsection and is related to:

* **(Item 7.7)** Procedures for quantification of seismic margins for existing nuclear installations.

POS items were identified in two subsections.

“Concept of seismic margin” subsection contains a POS item related to:

* **(Item 7.1)** Provision of the seismic margin in the design by a conservative definition of SL‑2 and by acceptance criteria specified in applicable nuclear design codes.

“Procedures to assess the seismic margin” subsection contains POS items related to:

* **(Item 7.7)** Procedures for quantification of seismic margins for existing nuclear installations.
* **(Item 7.11)** Comparing the seismic margin (HCLPF capacity) for the installation with the adequate seismic margin or with values established by the regulatory body.

*Refer to* ***Subsection 2.8*** *of the compliance assessment report [50] for more details.*

#### Section 8 (Seismic instrumentation and post‑earthquake actions)

Most of the guidelinespresented in **Section 8** of the reference document [20], are responsibility of the Owner and are site-specific.

In the AP1000 standard plant, as described in DCD Subsection 3.7.4, the seismic instrumentation is designed to provide the following:

• Collection of seismic data in digital format,

• Analysis of seismic data after a seismic event,

• Operator notification that a seismic event exceeding a preset value has occurred,

• Operator notification (after analysis of data) that a predetermined cumulative absolute velocity value has been exceeded.

The seismic instrumentation serves no safety-related function. Site-specific seismic instrumentation, if required, is the responsibility of the Owner.

The AP1000 plant Owner will determine the location for the free-field acceleration sensor as described in subsection 3.7.4.2.1. The AP1000 plant seismic monitoring system will provide for signal input from the free field sensor. Seismic sensor located on the Nuclear Island basemat is considered in the AP1000 plant design. There are also two sensors located on the shield building structure and containment internal structure near the operating floor at elevation 138’

The Owner will prepare site-specific procedures for activities following an earthquake. These procedures will be used to accurately determine both the response spectrum and the cumulative absolute velocity of the recorded earthquake ground motion from the seismic instrumentation system. The procedures and the data from the seismic instrumentation system will provide sufficient information to guide the operator on a timely basis to determine if the level of earthquake ground motion requiring shutdown has been exceeded. An activity of the procedures will be to address measurement of the post-seismic event gaps between the new fuel rack and the walls of the new fuel storage pit, and between the individual spent fuel racks and from the spent fuel racks to the spent fuel pool walls, and to take appropriate corrective action if needed (such as repositioning the racks or analysis of the as-found condition). For more details, refer to DCD Subsection 3.7.5.

POS items were identified in two subsections.

“Seismic instrumentation” subsection contains POS items related to:

* **(Item 8.1)** Reasons why seismic instrumentationshould be installed at nuclear installations.
* **(Item 8.2)** The seismic categorization and safety classification of seismic instrumentation.
* **(Item 8.4)** Definition, specification, procurement, installation, calibration, maintenance and upgrading of the seismic instrumentation installed at the nuclear installation in accordance with the specific needs of the nuclear installation and the significance of the seismic risk to the safety of the installation.
* **(Item 8.5)** Including processing, interpretation and use of the data obtained from seismic instrumentation in the operating procedures (including emergency operating procedures) for the installation and its management in accordance with the management system.
* **(Item 8.6)** Installation of a suggested minimum amount of seismic instrumentation.
* **(Item 8.7)** Ability of the seismic instrumentation to provide damage parameters based on the integration of the acceleration record, as an important tool for assessing the installation response in the event of an earthquake.
* **(Item 8.8)** Comparison of damage indicators with values of the same quantities derived from the free field design basis earthquake and with data from earthquake experience.
* **(Item 8.9)** The seismic instrumentation allowing an easy comparison of the response spectra of the actual seismic event with the design basis response spectra.

“Post‑earthquake actions” subsection contains POS items related to:

* **(Item 8.10)** Planning Post‑earthquake actions.
* **(Item 8.11)** Inputs to the post‑earthquake action programme.
* **(Item 8.12)** Comprehensibility of the post‑earthquake action programme should be comprehensive enough to minimize the likelihood of a prolonged installation shutdown following seismic vibratory ground motion that does not damage SSCs important to safety.
* **(Item 8.13)** “Felt earthquake” and earthquake detection via seismic instrumentation.
* **(Item 8.14)** “Significant earthquake” and subsequent initiation of actions as part of the post‑earthquake action programme.
* **(Item 8.15)** Definition of “Significant earthquake”.
* **(Item 8.16)** The objective of the post‑earthquake action programme.
* **(Item 8.17)** Basic stages of the post‑earthquake action programme.

*Refer to* ***Subsection 2.9*** *of the compliance assessment report [50] for more details.*

#### Section 9 (Seismic design for nuclear installations other than nuclear power plants)

**Section 9** of the reference document [20], does not apply to nuclear power plant applications and therefore is not assessed for the AP1000 plant.

#### Section 10 (Application of the management system)

Majority of the guidelines presented in **Section 10** of the reference document [20] are met “as stated” by Westinghouse, while several require site-specific project input data from the Owner.

Westinghouse has conducted the AP1000 plant design development under its recognized Quality Management System (QMS) and has taken prime responsibility for safety during the design development. This QMS and those of other participating organizations are applied in AP1000 plant project implementations to ensure design specifications are met. The Westinghouse program for quality assurance during the plant design, construction, and operation phases is described in DCD Chapter 17 - Quality Assurance.

It is the Owner’s responsibility to establish the management system and have in place the appropriate Quality Assurance (QA) program, as well as to make sure that the QA arrangements are consistent with regulatory requirements.

POS items were identified in two subsections.

“Application of the management system” subsection contains POS items related to:

* **(Item 10.1)** Requirement to the management system to ensure the quality and the control of processes and activities performed as part of the seismic design.
* **(Item 10.2)** Establishing and application of the design processes for the development of the concept, detailed plans, supporting calculations and specifications for a nuclear installation and its SSCs.
* **(Item 10.3)** Establishing seismic design inputs, requirements, outputs, changes, control and records in the design processes.

*Refer to* ***Subsection 2.11*** *of the compliance assessment report [50] for more details.*

### Non Compliance

No “Non Compliances” have been identified in the assessment [50].

## SSG-68 DESIGN OF NUCLEAR INSTALLATIONS AGAINST EXTERNAL EVENTS EXCLUDING EARTHQUAKES

Following summary reflects results of the assessment conducted against reference [21] as part of the compliance assessment report [51] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [51] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-68: Design of nuclear installations against external events excluding earthquakes” [21] is expected to be demonstrated either via compliance with the guidelines “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “IAEA Specific Safety Guide No. SSG-68: Design of nuclear installations against external events excluding earthquakes” has been assigned to a “**Low Risk**” category.

The AP1000 plant design complies with all the design related guidelines presented in the reference document [21]. Site-specific data or project-specific concerns resulting from the site evaluation process are to be provided by the Owner, so that Westinghouse can ensure that the site falls into the standard AP1000 plant site envelope. Risks category might change depending on the site-specific data provided by the Owner.

With respect to external hazards, the standard AP1000 plant design uses conservative, bounding site characteristics (temperatures, wind velocities, etc.) to achieve a high level of safety. As a result, the AP1000 plant design can be applied to different geographical regions around the world with varying regulatory standards and utility expectations. The design of the AP1000 plant includes allowances for natural hazards such as earthquakes, floods, storms, and tornadoes at the site. The nuclear island structures are designed to withstand these phenomena without the loss of capability to perform safety functions. Specific sites are evaluated with respect to the AP1000 plant site envelope to assure site-specific safety capabilities.

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### SSG 68 - Section 2 (General concepts and application of safety criteria to the design of nuclear installations against external events)

Guidelines presented in **Section 2** of the reference document [21] are met by Westinghouse either “as stated” or via compliance with their objectives. Multiple guidelines require a site-specific scope to be performed.

A standard set of expected external hazards selected based on operating experience have been identified and incorporated into the design of the AP1000 plant, as described in DCD Chapters 2 and 3. This standard design basis is applied to the applicable SSCs at all AP1000 plant sites.

Additional work might be required to ensure that the AP1000 plant design is compatible with the site conditions provided by the Owner and/or Licensee. Additionally, emergency response plans will need to be evaluated on a site-specific basis.

If a general design concept of the AP1000 plant does not cover the site-specific external event, additional work may be required to address design and protective measures against such phenomenon.

Some of the requirements were identified as CWO with the AP1000 plant because the AP1000’s plant design meets this criteria in different fashion but the main safety objective goal is achieved. This may also result from a different approach to the design or terminology based on the NRC Regulations.

Several “Compliant with objective” (CWO) items were identified in three subsections.

“Other aspects of design against external events” subsection contains CWO item related to:

* **(Item 2.18)** The best estimate value in probabilistic approach.

“Structures, systems and components to be protected against external events” subsection contains three CWO items related to:

* **(Item 2.23)** External event category 1.
* **(Item 2.24)** External event category 2.
* **(Item 2.25)** External event category 3.

“Design safety features for design basis external events and beyond design basis external events” subsection contains a CWO item related to:

* **(Item 2.49)** Seismic isolation of a SSC and its effect on the response to other external hazards

In addition, multiple “Project of site-specific scope” (POS) items were identified in four subsections.

“Design requirements for nuclear installations” contains POS items that are related to:

* **(Item 2.3)** Identification of all internal and external hazards.
* **(Item 2.4)** Engineering design rules.

“Other aspects of design against external events” subsection contains POS items related to:

* **(Item 2.8)** Hazard evaluation.
* **(Item 2.9)** Design basis and Beyond Design Basis external events.
* **(Item 2.10)** Loading conditions for Design basis and Beyond Design Basis external events.

“Design and evaluation for design basis external events and beyond design basis external events” subsection contains POS items related to:

* **(Item 2.26)** The design of a nuclear installation for an external event and all credible consequential effects of that event
* **(Item 2.27)** Evaluation of the effects of selected external events on the installation, including all credible secondary effects.
* **(Item 2.34)** Written procedures clearly defining the actions to be taken once an advanced warning is received
* **(Item 2.35)** Off-site infrastructure
* **(Item 2.36)** Mitigatory actions that involve the support of off‑site facilities
* **(Item 2.42)** Approaches to a provision of the protection of a nuclear installation against external events
* **(Item 2.44)** Representation of a balance of system layout, safety aspects (system and nuclear installation) and operational aspects, taking into account external events, by the design of a new nuclear installation
* **(Item 2.45)** Limitation of design modifications to an existing nuclear installation to specifically address changes in the assessment of the site specific hazard, design options such as relocating redundant systems or elements of systems
* **(Item 2.46)** Considerations in the design of a nuclear installation against design basis external events

“Administrative measures” subsection contains a POS item related to:

* **(Item 2.50)** Development of administrative measures, in conjunction with other measures, as part of the protection scheme for each external event, as appropriate.

*Refer to* ***Subsection 2.3*** *of the compliance assessment report [51] for more details.*

#### SSG 68 - Section 3 (Design basis for external events)

Guidelines presented in **Section 3** of the reference document [21] are met by Westinghouse “as stated”. However, the site-specific hazards identified by the Owner and/or Licensee will need to be evaluated on a site-specific basis to determine if they are applicable based on the screening process for which site parameters are described in DCD Section 2.2.

A standard set of expected external hazards selected based on operating experience have been identified and incorporated into the design of the AP1000 plant, as described in DCD Chapters 2 and 3. This standard design basis is applied to the applicable SSCs at all AP1000 plant sites.

The possibility of creating cliff edge effects have been assessed, resulting in a very low core damage frequency and a very low large release frequency with the inclusion of conservative assumptions made in specifying the success criteria of the passive systems.

Multiple POS items were identified in five subsections.

“Derivation of the design basis from the site hazard evaluation” subsection contains POS items related to:

* **(Item 3.1)** Adequate communication between parties responsible for site hazard evaluation and designing nuclear installations
* **(Item 3.2)** Provision of information to the site hazard evaluation regarding the derivation of design basis external events and beyond design basis external events, including the appropriate level of annual frequency of exceedance to be considered.
* **(Item 3.3)** Screening as a part of the hazard analysis in site evaluation
* **(Item 3.5)** Establishing the design loading conditions through a combination of deterministic and probabilistic methods and proceeding with the design in a deterministic manner
* **(Item 3.6)** Possibility of exclusion of a specific loading condition as a potential design basis external event from further analysis.
* **(Item 3.7)** Analysis of site hazards and their representation in a set of hazard curves
* **(Item 3.8)** Considerations in the specification of design basis external events and beyond design basis external events serving to satisfy the requirements for limits to the mean annual large early release frequency and/or the mean annual large release frequency.

“Overall design approach” subsection contains a POS item related to:

* **(Item 3.12)** Inclusion of the effects of causal and concomitant events to the initial conditions of the installation for the design basis external event and beyond design basis external event

“Derivation of design basis external event loading conditions: general considerations” subsection contains a POS item related to:

* **(Item 3.17)** Derivation of load–time functions for localized loads with substantial local effects resulting from external human induced events.

“Derivation of design basis external event and beyond design basis external event loading conditions for specific external events” subsection contains a POS item related to:

* **(Item 3.21)** Hazards screening and loading conditions

“Evaluation of beyond design basis external events: cliff edge effects” subsection contains a POS item related to:

* **(Item 3.23**) Methods for defining the beyond design basis external event loading conditions for assessing the margins and evaluation of cliff edge effects

*Refer to* ***Subsection 2.4*** *of the compliance assessment report [51] for more details.*

#### SSG 68 - Section 4 (Installation layout and design approach)

Requirements presented in the following Section are met by Westinghouse.

The AP1000 plant design meets the criteria set in **Section 4** of the reference document [21] by following applicable codes, industry standards and NRC recommendations as described throughout DCD Chapters 2 and 3.

Additional work might be required to ensure that the AP1000 plant design is compatible with the site conditions provided by the Owner and/or Licensee.

Several CWO items were identified in one subsection.

“Installation layout” subsection contains one CWO item related to:

* **(Item 4.6)** Utilization of a protective structures in instances in which locating an item important to safety inside a building structure is not practicable or even possible

POS items were identified in two subsections.

“Installation layout” subsection contains POS items related to:

* **(Item 4.4)** Demonstration of practicability of moving non-permanent equipment (used for fulfilment of a safety function) from storage locations (off the site and on the site) to connection points on the site
* **(Item 4.13)** Placing of flood sensitive equipment important to safety inside watertight compartments of buildings or at elevations above the level of the flood
* **(Item 4.17)** Combination of event loads with normal operational loads and loads from other extreme events for extreme events, which are more frequent than rare events.

“Approach to structural design” subsection contains POS items related to:

* **(Item 4.20)** Necessity to use hazard parameters to derive design basis external event and beyond design basis external event parameters for the design and evaluation process for each external event.
* **(Item 4.21)** Consistency between the derivation of the design basis parameters and the relevant loading scheme for the selected design basis external events
* **(Item 4.38)** Degradation involving corrosive chemicals or biological phenomena resulting from external events and requirements to protective measures.

*Refer to* ***Subsection 2.5*** *of the compliance assessment report [51] for more details.*

#### SSG 68 - Section 5 (Design provisions against external events)

Guidelines presented in **Section 3** of the reference document [21] are met by Westinghouse either “as stated” or via compliance with their objectives. Some of the guidelines show compliance with objectives with the AP1000 plant because the AP1000 plant design meets this criteria in a different fashion, so the main safety objective goal is achieved. This may also result from a different approach to the design methodology or terminology based on the NRC Regulations.

Specific sites are evaluated with respect to the AP1000 plant site envelope to assure site-specific safety capabilities, using the information provided by the Owner and/or Licensee.

The AP1000 plant is designed to handle natural hazards such as earthquakes, floods, and storms at the site. The Nuclear Island structures are designed to withstand the effects of hurricanes, floods, tornadoes, tsunamis, earthquakes and other events without the loss of capability to perform safety functions.

CWO items were identified in twelve subsections.

“External floods, including tsunamis” subsection contains CWO items related to:

* **(Item 5.2)** Scenarios to consider
* **(Item 5.4)** Consideration of dynamic and static effects of water as well as erosion phenomena
* **(Item 5.6)** Wind wave analysis
* **(Item 5.8)** Consideration of uplift and subsidence of the Earth’s surface for tsunamis induced by earthquakes
* **(Item 5.9)** The seiche hazard analysis
* **(Item 5.10)** Considerations for the design against river flooding
* **(Item 5.11)** The design relating to flooding due to local precipitation
* **(Item 5.12)** The parameters used to characterize floods due to the sudden release of impounded water
* **(Item 5.15)** Applicability of flooding due to local precipitation to all sites
* **(Item 5.17)** Derivation of design basis flood conditions
* **(Item 5.19)** Consideration of effects associated with low water levels, including drawdown, on items important to safety (including the ultimate heat sink)
* **(Item 5.23)** Inclusion of flood barriers when dry site concept cannot be applied

“Extreme winds” subsection contains a CWO item related to:

* **(Item 5.59)** Evaluation of the ultimate heat sink and associated transport systems and ensuring that any changes in water level caused by extreme winds will not prevent the transport and absorption of residual heat

”Other extreme meteorological conditions” subsection contains CWO items related to:

* **(Item 5.75)** Design of intake structures for the heat transport systems directly associated with the ultimate heat sink
* **(Item 5.77)** Measures to confirm that the facilities provided to transfer heat to the ultimate heat sink will retain their capability under extreme meteorological conditions

“Volcanism” subsection contains CWO items related to:

* **(Item 5.81)** Protection against some of the effects of volcanic phenomena provided by design envelope of the nuclear installation for external hazards
* **(Item 5.82)** Additional design features or site protective measures associated with effects of volcanic phenomena
* **(Item 5.83)** Consequences of tephra fallout on the physical loads and SSC operation
* **(Item 5.85)** Volcano generated missiles
* **(Item 5.86)** Hazards relating to gases and aerosols resulting from volcanic eruption
* **(Item 5.87)** Volcano induced flooding
* **(Item 5.88)** Volcanic earthquakes
* **(Item 5.89)** Non‑exclusionary volcanic hazards

“External fire” subsection contains CWO items related to:

* **(Item 5.95)** Combination of effects of an external fire originating from sources such as fuel storage, vehicles or natural vegetation with normal operating loads
* **(Item 5.105)** Ensuring adequate supply of air to all diesel generators and other emergency power sources necessary to perform safety functions

“External explosions” subsection contains a CWO item related to:

* **(Item 5.118)** Obtaining deflagration loads

“Toxic, flammable, corrosive and asphyxiant chemicals and their mixtures in air and liquids” subsection contains CWO item related to:

* **(Item 5.144)** Technical basis for the hazardous chemicals removal capability
* **(Item 5.149)** Supplementary control room

“Radiological hazards from other on‑site and collocated installations” subsection contains CWO items related to:

* **(Item 5.152)** The release of radioactive gases, liquids and aerosols from adjacent operating nuclear units or storage installations, from vehicles transporting new or spent fuel, and from other on‑site and off‑site sources
* **(Item 5.159)** Recommendations on the protection of operating personnel against asphyxiant and toxic gases

“Aircraft crash” subsection contains CWO items related to:

* **(Item 5.162)** Possible utilization of alternative paths (normally one train) to ensure the satisfactory performance of safety functions
* **(Item 5.176)** An alternative approach to assessing the effects of secondary missiles and debris
* **(Item 5.182)** Consideration of soil in analysis of the hazard from accidental aircraft crashes at a nuclear installation
* **(Item 5.183)** Consideration of masses of the structural members and the dead load of equipment in numerical model, actual live loads and representation of fluid stored in tanks or pools
* **(Item 5.184)** Damping in the global area
* **(Item 5.187)** Addressing local, global and vibration effects of the crash.
* **(Item 5.191)** Loads that have to be combined with the aircraft crash load

“Electromagnetic interference” subsection contains a CWO item related to:

* **(Item 5.194)** A distinction should be made between sources of electromagnetic interference that are off the site and those that originate within the installation

”Biological phenomena” subsection contains CWO items related to:

* **(Item 5.205)** Utilization of screens to prevent ingress of fish, seaweed, debris, etc. from entering cooling water system
* **(Item 5.207)** Adequate treatment of cooling water used in condensers and in heat transport systems directly associated with the ultimate heat sink
* **(Item 5.208)** Provision of frequent biological monitoring of the ultimate heat sink

“Hazards associated with floating bodies and hazardous liquids” subsection contains CWO items related to:

* **(Item 5.210)** Accounting for some components being outside the site boundary and, in some cases, spread over a wide area when it comes to the design of the ultimate heat sink and the water intake for the service water systems important to safety
* **(Item 5.211)** The collision of floating bodies with water intakes or with structures of the ultimate heat sink
* **(Item 5.214)** Consideration of effects of shipping accidents on the capability to fulfil the heat removal safety function (For sites with a safety related intake of water from navigable water bodies)
* **(Item 5.215)** Resilience of the water intake design against ship collision and oil spills or releases of corrosive fluids or particles
* **(Item 5.216)** Application of dynamic action (derived from the analysis) to the structures which intended to guarantee structural integrity for the cases involving debris and ice
* **(Item 5.219)** Prevention of fundamental safety functions loss
* **(Item 5.221)** Considerations related to preservation, to the maximum extent possible, of the vessel to avoid spillage or blockage of the water intake.
* **(Item 5.222)** Protective structure types that are commonly used in ports or waterways that may be adapted to protect water intakes and components of the ultimate heat sink
* **(Item 5.223)** Measures to maintain the supply of cooling water and ensure the capability of the ultimate heat sink where a potential direct collision with the intake structure is of concern
* **(Item 5.224)** Adequate measures to mitigate the effects of the potential spillage of liquids that could readily mix with the intake water and result in damage to the heat transport system or could seriously degrade the heat transfer capability
* **(Item 5.227)** Methods in the assessment for beyond design basis collisions and design basis collisions
* **(Item 5.228)** Definition of beyond design basis external events by increasing the size of the floating body and/or the impact velocity with respect to the design basis values

“Combination of hazards” subsection contains a CWO item related to:

* **(Item 5.231)** Conditions under which , external hazards can be combined with other extreme loads

POS items were identified in thirteen subsections.

“External floods, including tsunamis” subsection contains POS items related to:

* **(Item 5.1)** Recommendations on assessing the potential risk of flooding of a site due to diverse initiating causes and scenarios and considered phenomena.
* **(Item 5.2)** Scenarios to consider
* **(Item 5.3)** Consideration of the potential damage to SSCs important to safety due to the infiltration of water into internal areas of the installation as well as the resulting water pressure on walls and foundations that could challenge their structural capacity or stability
* **(Item 5.4)** Consideration of dynamic and static effects of water as well as erosion phenomena
* **(Item 5.5)** Storm surge analysis.
* **(Item 5.6)** Wind wave analysis
* **(Item 5.7)** The tsunami flooding analysis
* **(Item 5.8)** Consideration of uplift and subsidence of the Earth’s surface for tsunamis induced by earthquakes
* **(Item 5.9)** The seiche hazard analysis
* **(Item 5.10)** Considerations for the design against river flooding
* **(Item 5.12)** The parameters used to characterize floods due to the sudden release of impounded water
* **(Item 5.13)** The parameters describing bores and mechanically induced waves
* **(Item 5.15)** The design relating to flooding due to local precipitation
* **(Item 5.16)** The tidal range
* **(Item 5.17)** Derivation of design basis flood conditions
* **(Item 5.18)** Protection of SSCs important to safety from damage due to flooding
* **(Item 5.21)** Means for protection of SSCs important to safety from damage due to flooding
* **(Item 5.24)** Civil engineering structures (e.g. sea walls) that are permanent barriers for protecting SSCs important to safety against flooding
* **(Item 5.26)** External barriers and natural or artificial islands
* **(Item 5.27)** Infill necessary to raise the installation above the level of the design basis flood
* **(Item 5.28 and 5.29)** Flood monitoring system
* **(Item 5.30)** Effects associated with design loading conditions for a nuclear installation located at the coast
* **(Item 5.31)** The design of a nuclear installation against river floods
* **(Item 5.32)** River floods in cold climates
* **(Item 5.33)** The design of a nuclear installation against estuary floods

“Extreme winds” subsection contains a POS item related to:

* **(Item 5.37)** Determination of the design basis wind speed basing on hazard evaluation results
* **(Item 5.38)** Averaged wind speeds and topological effects
* **(Item 5.40)** Direction of extreme winds
* **(Item 5.42)** Form of derived structural loads
* **(Item 5.43)** Wind velocity and wind forces
* **(Item 5.44)** Static and dynamic wind loads
* **(Item 5.45)** Wind generated pressures influenced by interference effect
* **(Item 5.46)** The combinations of wind induced loads with other design loads
* **(Item 5.51)** Wind‑borne missile analysis
* **(Item 5.56)** Other parameters for the design of a nuclear installation against dust storms and sandstorms
* **(Item 5.57)** Considerations for the design against dust storms and sandstorms
* **(Item 5.58)** Other effects of extreme winds
* **(Item 5.63)** Assessment for beyond design basis wind
* **(Item 5.64)** The methods used in the assessment for beyond design basis wind

”Other extreme meteorological conditions” subsection contains POS items related to:

* **(Item 5.65)** Extreme meteorological conditions
* **(Item 5.66)** Consideration of effects of changes in river water temperature
* **(Item 5.67)** Damage due to the extreme meteorological conditions
* **(Item 5.68)** Damage caused by lightning
* **(Item 5.69)** Environmental parameters for extreme meteorological conditions
* **(Item 5.70)** Codes and standards for the design of nuclear installations in relation to the extreme meteorological conditions
* **(Item 5.71)** Consideration of effects of snow in design and safety analysis
* **(Item 5.72)** Consideration of the effect of extreme air temperatures and water temperatures on items important to safety in design and safety analysis
* **(Item 5.73)** Effects of lightning
* **(Item 5.74)** Special protection from lightning
* **(Item 5.76)** Effects of extreme weather conditions on make‑up supplies
* **(Item 5.77)** Measures to confirm that the facilities provided to transfer heat to the ultimate heat sink will retain their capability under extreme meteorological conditions

“Volcanism” subsection contains POS items related to:

* **(Item 5.79)** List of volcanic phenomena
* **(Item 5.80)** Measures available for the phenomena associated with volcanoes
* **(Item 5.81)** Protection against some of the effects of volcanic phenomena provided by design envelope of the nuclear installation for external hazards
* **(Item 5.82)** Additional design features or site protective measures associated with effects of volcanic phenomena
* **(Item 5.83)** Consequences of tephra fallout on the physical loads and SSC operation
* **(Item 5.84)** Massive flows
* **(Item 5.85)** Volcano generated missiles
* **(Item 5.86)** Hazards relating to gases and aerosols resulting from volcanic eruption
* **(Item 5.87)** Volcano induced flooding
* **(Item 5.88)** Volcanic earthquakes
* **(Item 5.89)** Non‑exclusionary volcanic hazards

“External fire” subsection contains POS items related to:

* **(Item 5.90)** Fire originated outside the site
* **(Item 5.93)** Fire resulting from aircraft crash
* **(Item 5.94)** Consideration of the characteristics of postulated fire and secondary effects in fire hazard evaluation
* **(Item 5.95)** Combination of effects of an external fire originating from sources such as fuel storage, vehicles or natural vegetation with normal operating loads
* **(Item 5.102)** Protection of the installation against fires that originate outside the site

“External explosions” subsection contains POS items related to:

* **(Item 5.108)** An analysis of the ability of the structures in a nuclear installation to resist the effects of a gas cloud explosion
* **(Item 5.109)** Effects of explosions
* **(Item 5.110)** Consideration of missiles produced by explosion in the design of installation
* **(Item 5.111)** Explosions during the processing, handling, transport or storage of potentially explosive substances outside safety related buildings
* **(Item 5.112)** Methods of determining design basis parameters to protect the nuclear installation against unacceptable damage by pressure waves from detonations
* **(Item 5.113)** A conservative estimate for the portion of the total mass assumed to detonate
* **(Item 5.114)** The potential for flame acceleration and overpressure generation due to obstacles in gas clouds
* **(Item 5.120)** Steps involved in design of structures to withstand the effects of detonations or deflagrations
* **(Item 5.121)** The minimum parameters used to define the response of a particular structure
* **(Item 5.122)** Local and global effects in evaluating blast effects
* **(Item 5.123)** External walls or roof elements directly exposed to explosion loads
* **(Item 5.124)** Justification for using simplified approaches to checking the ability of the primary load path to carry loads transferred from the exterior surfaces
* **(Item 5.125)** Vibratory loads induced in building structures by the explosion
* **(Item 5.127)** Protection against the effects of an external explosion by means of a suitable stand‑off distance between the explosion source and the target SSC
* **(Item 5.128)** Considerations in calculations of the distances necessary to provide protection by means of separation, usage of TNT equivalence data, evaluation of protection adequacy considering mobile sources on transport routes in the site vicinity and a sufficient number of plausible locations for the explosion

“Toxic, flammable, corrosive and asphyxiant chemicals and their mixtures in air and liquids” subsection contains POS items related to:

* **(Item 5.133)** The release of toxic, flammable, corrosive and asphyxiant chemicals and associated damage, including electrical and electronic equipment located outside buildings.
* **(Item 5.134)** Recommendations on evaluating hazards from the release of hazardous fluids at or near the installation
* **(Item 5.135)** Mean of calculation of the atmospheric dispersion of the released chemicals
* **(Item 5.136)** Used models and considerations
* **(Item 5.137)** Considerations and goal for the calculation of atmospheric dispersion for different scenarios
* **(Item 5.138)** Consideration of density of toxic, flammable, corrosive or asphyxiant gases and vapor clouds and its effects on vertical diffusion
* **(Item 5.139)** Beyond design basis releases
* **(Item 5.140)** Dispersion calculations once a toxic, flammable, corrosive, or asphyxiant gas or vapor cloud has been postulated.
* **(Item 5.142)** Assumptions for gas or vapor concentrations
* **(Item 5.144)** Technical basis for the hazardous chemicals removal capability
* **(Item 5.145)** Comparison of concentrations to the limit in national regulations and equipment specifications
* **(Item 5.147)** Detection systems at control room air intakes and protective systems for concentrations exceeding prescribed limits
* **(Item 5.148)** Provision of versatile detectors where multiple types of gas or vapour could be a hazard
* **(Item 5.149)** Requirements to the supplementary control room
* **(Item 5.150)** Ensuring the inspection intervals are sufficient to avoid safety systems impairment and suggested means of protection
* **(Item 5.151)** Considerations regarding methods in the assessment for beyond design basis
* **(Item 5.152)** The release of radioactive gases, liquids and aerosols from adjacent operating nuclear units or storage installations, from vehicles transporting new or spent fuel, and from other on‑site and off‑site sources
* **(Item 5.153)** Recommendations to be followed when identifying the external radioactive releases to be considered in the design of the installation
* **(Item 5.154)** Approach to define beyond design basis releases
* **(Item 5.156)** Calculation of potential concentration of radionuclides inside the installation in the case of the release of radioactive material to the atmosphere and determination of the extension time and the interaction time of the gas or vapor cloud
* **(Item 5.157)** Consideration of the effect on the installation and the possible exposure of operating personnel in scenarios in which radioactive material might enter the cooling water intake
* **(Item 5.159)** Recommendations on the protection of operating personnel against asphyxiant and toxic gases
* **(Item 5.160)** Methods in the assessment for beyond design basis releases

“Aircraft crash” subsection contains POS items related to:

* **(Item 5.161)** Recommendations on evaluating the hazard from an aircraft crash on the nuclear installation site
* **(Item 5.162)** Alternative paths in ensure the satisfactory performance of safety functions, iterations in SSC design and SSC classification
* **(Item 5.169)** Utilization of Load–time functions in considering a design basis external event
* **(Item 5.170)** Utilization of coupled analysis for determining dynamic interaction between the missile and the target
* **(Item 5.182)** Consideration of soil in analysis of the hazard from accidental aircraft crashes at a nuclear installation
* **(Item 5.183)** Consideration of masses of the structural members and the dead load of equipment in numerical model, actual live loads and representation of fluid stored in tanks or pools
* **(Item 5.184)** Damping in the global area
* **(Item 5.191)** Loads that have to be combined with the aircraft crash load

“Electromagnetic interference” subsection contains POS items related to:

* **(Item 5.193)** Protection against hazards relating to electromagnetic interference
* **(Item 5.196)** Electromagnetic pulses
* **(Item 5.198)** Protection against stationary and mobile sources of electromagnetic interference

”Biological phenomena” subsection contains POS items related to:

* **(Item 5.202)** Analysis of the environmental conditions and establishing a monitoring regime

“Hazards associated with floating bodies and hazardous liquids” subsection contains POS items related to:

* **(Item 5.211)** The collision of floating bodies with water intakes or with structures of the ultimate heat sink
* **(Item 5.212)** Evaluation of hazards from ship collisions and a vessel impact design basis
* **(Item 5.213)** Head‑on bow collisions and sideways collisions
* **(Item 5.214)** Consideration of effects of shipping accidents on the capability to fulfil the heat removal safety function (For sites with a safety related intake of water from navigable water bodies)
* **(Item 5.215)** Resilience of the water intake design against ship collision and oil spills or releases of corrosive fluids or particles
* **(Item 5.217)** Requirements to protective measures for coastal sites.
* **(Item 5.218)** Establishing preventive measures in cooperation with the navigation authorities and means to achieve prevention
* **(Item 5.220)** Requirements for protective structures and structures exposed to potential impacts
* **(Item 5.221)** Considerations related to preservation, to the maximum extent possible, of the vessel to avoid spillage or blockage of the water intake.
* **(Item 5.222)** Protective structure types that are commonly used in ports or waterways that may be adapted to protect water intakes and components of the ultimate heat sink
* **(Item 5.223)** Measures to maintain the supply of cooling water and ensure the capability of the ultimate heat sink where a potential direct collision with the intake structure is of concern
* **(Item 5.224)** Adequate measures to mitigate the effects of the potential spillage of liquids that could readily mix with the intake water and result in damage to the heat transport system or could seriously degrade the heat transfer capability
* **(Item 5.227)** Methods in the assessment for beyond design basis collisions and design basis collisions
* **(Item 5.228)** Definition of beyond design basis external events by increasing the size of the floating body and/or the impact velocity with respect to the design basis values

“Other external hazards” subsection contains a POS item related to:

* **(Item 5.230)** Using a combination of hazards for the assessment of beyond design basis external events in case hazards for which a specific beyond design basis external event has not been defined

“Combination of hazards” subsection contains a CWO item related to:

* **(Item 5.231)** Conditions under which , external hazards can be combined with other extreme loads

*Refer to* ***Subsection 2.6*** *of the compliance assessment report [51] for more details.*

#### SSG 68 - Section 6 (Safety design provisions for nuclear installations other than nuclear power plants)

**Section 6** of the reference document [21] does not apply to nuclear power plant applications and therefore is not assessed for the AP1000 plant.

#### SSG 68 - Section 7 (Application of the management system to the design of a nuclear installation against external events)

Guidelines presented in **Section 7** of the reference document [21] are met by Westinghouse.

Westinghouse has established appropriate plans and programs to support the design, construction and commissioning process of new-built AP1000 plants. The Westinghouse program for quality assurance during the plant design, construction, and operation phases is described in DCD Chapter 17 - Quality Assurance.

Detailed design is considered after provisions of the site-specific project input data by the Owner.

POS items, identified in the **Section 7** are related to:

* **(Item 7.1)** Management system
* **(Item 7.2)** Activities included to design process

*Refer to* ***Subsection 2.8*** *of the compliance assessment report [51] for more details.*

### Non Compliance

No “Non Compliances” have been identified in the assessment [51].

## SSG-69 EQUIPMENT QUALIFICATION FOR NUCLEAR INSTALLATIONS

Following summary reflects results of the assessment conducted against reference [22] as part of the compliance assessment report [52] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [52] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-69: Equipment Qualification for Nuclear Installations” [22] is expected to be demonstrated either via compliance with the guidelines “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “IAEA Specific Safety Guide No. SSG-69: Equipment Qualification for Nuclear Installations” has been assigned to a “**Low Risk**” category.

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### SSG 69 - Section 2 (Concepts and process of equipment qualification)

Guidelines, presented in **Section 2** of the reference document [22], that address basic concepts of the EQ, equipment qualification process, qualified life, Quality Management System, documentation and training for equipment qualification are met by Westinghouse “as stated”.

Guidelines addressing licensing, cooperation with regulatory bodies, preservation of the status of qualified components, monitoring of environmental conditions during operations, maintenance of the equipment qualification program during plant life, development and maintenance of a management system, maintaining the documentation and training of its personnel to support the maintenance of the equipment qualification program while the equipment is in service fall under responsibility of the Owner and/or Licensee.

#### SSG 69 - Section 3 (Design inputs for equipment qualification)

Guidelines in **Section 3** of the reference document [22] that address design inputs for equipment qualification and identification of equipment performance are met by Westinghouse “as stated”.

Guidelines addressing identification of service conditions is met by Westinghouse, however part of this requirement which addresses plant specific conditions during operation are responsibility of the Owner and/or Licensee.

The AP1000 plant is designed as a Standard Design that can be applied in different regions. As such, a preliminary suitability assessment requirement is no longer applicable to future AP1000 projects.

#### SSG 69 - Section 4 (Establishing equipment qualification)

Guidelines in **Section 4** of the reference document [22] are met by Westinghouse either “as stated” or via compliance with their objectives.

“Compliant with objective” (CWO) items were identified in the “qualification by type testing” subsection and are related to:

* **(Item 4.11)** An evaluation to determine how many test specimens need to be tested to ensure an accurate representation of the performance of the equipment to be qualified.
* **(Item 4.30)** The simulation of radiation ageing.

Type testing methods used by Westinghouse fully comply with the guidelines, except for one item for which is slightly different from the guideline but is consistent with its objective. This CWO item, however, has no effect on the plant design, and therefore does not affect the risk factor.

Guidelines addressing qualification by analysis and combined methods are met by Westinghouse “as stated”.

*Refer to* ***Subsection 2.2.3*** *of the compliance assessment report [52] for more details.*

#### SSG 69 - Section 5 (Preservation of equipment qualification)

**Section 5** of the reference document [22] is predominantly responsibility of Owner and/or Licensee.

Guidelines addressing aging, protective barriers, marking and inspection of procured qualified equipment are met by Westinghouse “as stated”.

Guidelines addressing monitoring during operation, monitoring the condition of qualified equipment, periodic surveillance of qualified equipment, maintenance relating to qualified equipment and protective barriers for qualified equipment, reassessment due to changes of operating condition, the storage of qualified equipment with a defined shelf life and replacement of qualified equipment are responsibility of the Owner and/or Licensee.

#### SSG 69 - Section 6 (Evaluation of the effectiveness of the equipment qualification programmes and processes)

Guidelines, presented in **Section 6** of the reference document [22], that address evaluation of the effectiveness of the equipment qualification programme are mainly the responsibility of the Owner.

#### SSG 69 - Section 7 (Integration of equipment qualification into safety programmes and processes)

Guidelines addressing safety analysis report in **Section 7** of the reference document [22] are met by Westinghouse “as stated”.

The Westinghouse will supply necessary documentation related to interface between the equipment qualification programme and other programmes, however, programmes regarding licensing, long term operation etc. is full responsibility of the Owner. In addition, any modifications are also responsibility of the Owner and/or Licensee.

### Non Compliance

No “Non Compliances” have been identified in the assessment [52].

## SSG-70 OPERATIONAL LIMITS AND CONDITIONS AND OPERATING PROCEDURES FOR NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [23] as part of the compliance assessment report [53] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [53] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-70: Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants” [23] is expected to be demonstrated either via compliance with the guidelines “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “IAEA Specific Safety Guide No. SSG-70: Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants” has been assigned to a “**Low Risk**” category.

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### SSG 70 - Section 2 (The concept of operational limits and conditions and their development):

The recommended Operational Limits and Conditions (OLCs), that were developed by Westinghouse meet all the guidelines presented in **Section 2** of the reference document [23] “as stated ” and will be provided by Westinghouse.

Guidelines related to OLCs during plant operation life are responsibility of the Owner.

#### SSG 70 - Section 3 (Safety limits):

Guidelines that address safety limits in **Section 3** of the reference document [23] are met by Westinghouse “as stated”.

Obeying these limits during operation is the responsibility of the Owner.

#### SSG 70 - Section 4 (Limiting settings for safety systems):

Guidelines presented in **Section 4** of the reference document [23] are met by Westinghouse “as stated”.

It is a responsibility of the Owner to use them as recommended.

#### SSG 70 - Section 5 (Limits and conditions for normal operation):

Guidelines presented in **Section 5** of the reference document [23] are met by Westinghouse “as stated”.

It is the responsibility of the Owner to adopt these limits and safely operate the plant.

#### SSG 70 - Section 6 (Surveillance and testing requirements):

All the surveillance and testing requirements will be recommended by Westinghouse and meet guidelines presented **Section 6** of the reference document [23] .

The rest is the responsibility of the Owner.

One “Compliant with objective” (CWO) item was identified in **Section 6** and is related to:

* **(Item 6.4)** Coverage of activities to detect ageing and other forms of deterioration due to corrosion, fatigue and other mechanism by the surveillance requirements.

*Refer to* ***Subsection 2.1.5*** *of the compliance assessment report [53]* *for more details.*

#### SSG 70 - Section 7 (Operating procedures and guidelines):

All the operating procedures and guidelines will be provided by Westinghouse and these documents meet the guidelines presented in in **Section 7** of the reference document [23].

The rest is the responsibility of the Owner.

“Project of site-specific scope” (POS) items were identified in two subsections.

“Severe accident management guidelines” subsection contains POS related to:

* **(Item 7.28)** Addressing plant specific details as a part of the identification and selection of the most suitable actions to cope with severe accidents .

“Accidents at multiple unit sites” subsection contains POS related to:

* **(Item 7.32)** Multiple units being concurrently affected by an accident.
* **(Item 7.34)** Interconnections between units.

The procedures during commissioning require cooperation of both the Westinghouse and the Owner and/or Licensee.

*Refer to* ***Subsection 2.1.6*** *of the compliance assessment report [53]* *for more details.*

#### SSG 70 - Section 8 (Development of operating procedures):

Guidelines presented in **Section 8** of the reference document [23] are met by Westinghouse “as stated”.

It is the responsibility of the Owner and/or Licensee to adopt them.

#### SSG 70 - Section 9 (Compliance with operational limits and conditions and operating procedures):

Guidelines presented in **Section 9** of the reference document [23] are responsibility of the Owner and/or Licensee.

The generic plant design operating procedures and documentation are consistent with the generic plant Limits and Conditions.

One POS item was identified in the following section and is related to:

* **(Item 9.3)** Requirement regarding form of presenting OLCs for sites with multiple units.

*Refer to* ***Subsection 2.1.8*** *of the compliance assessment report [53]* *for more details.*

### Non Compliance

No “Non Compliances” have been identified in the assessment [53].

## SSG-73 CORE MANAGEMENT AND FUEL HANDLING FOR NUCLEAR POWER PLANTS

Following summary reflects results of the assessment conducted against reference [24] as part of the compliance assessment report [54] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [54] indicates that compliance with the guidelines presented in “IAEA Specific Safety Guide No. SSG-73: Core Management and Fuel Handling for Nuclear Power Plants” [24] is expected to be demonstrated either via compliance with the guidelines “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “IAEA Specific Safety Guide No. SSG-73: Core Management and Fuel Handling for Nuclear Power Plants” has been assigned to a “**Low Risk**” category.

### Roadmap of Items Specifically Discussed in the Guide Assessment

#### SSG 73 - Section 2 (Core management)

Guidelines presented in **Section 2** of the reference document [24] are met by Westinghouse either “as stated” or via compliance with their objectives. The rest of the guidelines are responsibility of the Owner.

“Compliant with objective” (CWO) items were identified in the “Computational methods for core calculations” subsection and are related to:

* **(Item 2.15)** Validation tests.
* **(Item 2.17)** Requirements to modifications to the software and databases used for core calculations.

*Refer to* ***Subsection 2.3*** *of the compliance assessment report [54] for more details.*

#### SSG 73 - Section 3 (Handling and storage of fresh fuel)

Most of the guidelines presented in **Section 3** of the reference document [24] are responsibility of the Owner.

Guidelines that fall under responsibility of Westinghouse are met “as stated”.

#### SSG 73 - Section 4 (Implementation of the refueling programme)

Guidelines presented in **Section 4** of the reference document [24] are met by Westinghouse “as stated”.

After the initial fuel loading for the AP1000 plant, the Owner has the overall responsibility for the refueling plan.

In addition, the Owner will be responsible for the foreign material exclusion program for the operating life of the plant.

#### SSG 73 - Section 5 (Handling, storage and inspection of irradiated fuel)

Guidelines presented in **Section 5** of the reference document [24] are met by Westinghouse either “as stated” or via compliance with their objectives.

Procedures for handling, movement and storage of the spent fuel are the responsibility of the Owner.

Fuel repair, activities associated with transport of casks or inspection of spent fuel or casks, personal radiation protection equipment, decontamination equipment, personal communication equipment, and fuel inspection equipment are the responsibility of the Owner as well.

One CWO item was identified in “storage of irradiated fuel” subsection and is related to:

* **(Item 5.20)** Plans for dealing with damaged or leaking fuel assemblies, and appropriate storage arrangements.

*Refer to* ***Subsection 2.6*** *of the compliance assessment report [54] for more details.*

#### SSG 73 - Section 6 (Handling and storage of core components)

Guidelines presented in **Section 6** of the reference document [24] fall under the Owner’s scope.

Adequate specified storage positions for storage of irradiated core components and tools are a part of the AP1000 plant design i.e. reactor vessel internals storage stands, control rod drive shaft storage racks and tool storage racks and other needed for core component shuffling in the Spent Fuel Pool.

Procedures and operation of storage, handling, inspection, testing, and maintenance activities after the initial fuel loading for the AP1000 plant is the responsibility of the Owner and should follow the refueling plans prepared.

Each refueling outage will have a specific plan to address that cycle’s core design requirements. This may be developed by Westinghouse or by the Owner depending on the scope of the refueling contract.

#### SSG 73 - Section 7 (Preparation of fuel for dispatch)

Procedures and actions for preparation, transport, and control of the spent fuel package off the site discussed in **Section 7** of the reference document [24] are the full responsibility of the Owner, therefore this section of IAEA SSG-73 is not assessed. Refer to the SSG-73 content for more details on the guidelines imposed on the Owner.

#### SSG 73 - Section 8 (Administrative and organizational aspects of Core management and fuel handling)

Administrative and organizational aspects of core management and fuel handling discussed in **Section 8** of the reference document [24] are the full responsibility of the Owner, therefore this section of IAEA SSG-73 is not assessed. Refer to the SSG-73 content for more details on the guidelines imposed on the Owner.

#### SSG 73 - Section 9 (Documentation for core management and Fuel handling activities)

Documentation for core management and fuel handling activities discussed in **Section 9** of the reference document [24] is the full responsibility of the Owner, therefore this section of IAEA SSG-73 is not assessed. Westinghouse will provide, as contracted, the design related information of the AP1000 standard plant in terms of core management and fuel handling activities, necessary to be included in such documentation. Refer to the IAEA SSG-73 content for more details on the guidelines imposed on the Owner.

### Non Compliance

No “Non Compliances” have been identified in the assessment [54].

## \*WENRA REPORT SAFETY OF NEW NPP DESIGNS

Following summary reflects results of the assessment conducted against reference [25] as part of the compliance assessment report [55] that is included in Attachment 1.

### High Level Risk Assessment Summary

Conducted compliance assessment [55] indicates that compliance with the requirements presented in “WENRA Report Safety of New NPP Designs (2013)” [25] is expected to be demonstrated either via compliance with the requirements “as stated” or via compliance with their objectives. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, “WENRA Report Safety of New NPP Designs (2013)” has been assigned to a “**Low Risk**” category.

### Roadmap of Items Specifically Discussed in the WENRA Report

#### Section 2 (WENRA safety objectives for new nuclear power plants)

**Section 2** of the reference document [25] contains explanatory statements.

#### Section 3 (Selected key safety issues)

the AP1000 plant design has been evaluated against WENRA common positions on specific safety issues (**Section 3** of the reference document [25]):

• Position 1 – defense-in-depth approach for new nuclear power plants,

• Position 2 – independence of the levels of defense-in-depth,

• Position 3 – multiple failure events,

• Position 4 – provisions to mitigate core melt and radiological consequences,

• Position 5 – practical elimination,

• Position 6 – external hazards,

• Position 7 – intentional crash of a commercial airplane.

Requirements presented in the Positions 1 to 7 are met by Westinghouse.

One “Compliant with objective” CWO item was identified for “**Position 3: multiple failures event”** and is addressed in subsection 2.25.4 of this report.

#### Section 4 (Lessons Learnt from the Fukushima Dai-ichi accident)

AP1000 Standard Design is aligned with the statements and recommendations presented in **Section 4** of the reference document [25].

Fulfilment of several statements will require site-specific inputs.

**Subsection 04.1** falls under "Project or site-specific scope” (POS) as all possible external hazards should be identified by the Owner and are dependent on a specific site.

Apart from identification of periodic safety reviews (04.1) external hazards (04.06), and emergency response center (04.6) fall under the scope of the Owner/Licensee.

#### Annex 1 (WENRA statement on safety objectives for new nuclear power plants, November 2010)

**Annex 1** of the reference document [25] elaborates on Objectives developed by WENRA.

Although the AP1000 plant design has taken U.S. Nuclear Regulatory Commission (NRC), American Society of Mechanical Engineers (ASME), American Nuclear Society (ANS), IAEA, and other international organizations’ standards into account, and thus a high level of compliance was expected, a thorough evaluation of the seven safety objectives proposed by the WENRA RHWG was performed against the AP1000 plant design and is presented in the compliance assessment report [55]. It is the conclusion of this assessment that the AP1000 plant design is in compliance with all WENRA objectives:

• O1 (normal operation, abnormal events, and prevention of accidents)

• O2 (accidents without core melt)

• O3 (accidents with core melt)

• O4 (independence among defense-in-depth levels)

• O5 (safety and security interfaces)

• O6 (radiation protection and waste management), and

• O7 (leadership and management for safety).

**Objectives** are fundamentally met by AP1000 design see assessment [55] sections 2 and 3. However part of the objectives contain requirements that are either site specific or also contain requirements for the License.

**Objectives 4, 6** and **7** have Owner/Licensee being responsible for specific parts of the imposed requirements.

**Objectives 4** and **6** also include requirements that are to be resolved on a site-specific basis.

### Non Compliance

No “Non Compliances” have been identified in the assessment [55].

### Identified Potential Risks To Be Addressed In Bulgaria Project

The only risk was identified in “**Position 3: multiple failure event**” and is caused by difference of approach in deriving requirements for diversity between AP1000 design and WENRA.

AP1000 plant requirements for diversity were derived from the aggressive PRA goals: the higher the frequency of an event, the more reliable the mitigation means needs to be in order to bring the corresponding CDF and LRF to the targeted order of magnitude. While this approach differs from a formal point of view from the more deterministic approach presented by WENRA, the end result is essentially the same and achieves a very robust level of diversity and redundancy in the design, with verification focused on the screening of higher frequency initiating events.

## ENERGY ACT

Following summary reflects results of the assessment conducted against reference [26] as part of the compliance assessment report [56] that is included in Attachment 1.

### High Level Risk Assessment

Conducted compliance assessment [56] indicates that compliance with the requirements presented in Bulgarian “Energy Act” [26] is expected to be fully demonstrated via compliance with the requirements “as stated”. Since no non-compliances have been identified and site-specific scope items are not expected to result in design changes or additional design analyses, Bulgarian “Energy Act” has been assigned to a “**Low Risk**” category.

Potential risks associated with the “Energy Act” are presented in “Identified Potential Risks To Be Addressed In Bulgaria Project” subsection.

AP1000 Standard Design or Reference Plant do not consider District Heating/Heat Generation. Nevertheless, in case heat production will be requested, requirements for the connection of the plant to the heating mains will apply. A specific study will need to be developed for this scenario. In case of heat production, AP1000 plant is expected to be connected to the existing KNPP heat supply infrastructure.

### Non Compliance

No “Non Compliances” have been identified in the assessment [56].

### Identified Potential Risks To Be Addressed In Bulgaria Project

In the potential request to provide Heat/District Heating by KNPP-NB a specific study of this option will need to be performed.

Additional risks including both standard design and cogeneration option comes from ordinances addressing electrical generation and heat supply and being referred to in **Article 83 (3)** and **Article 125 (3)** respectively. These ordinances established by Minister of Regional Development and Public Works and the Minister of Energy might contain additional design requirements that were not presented in the “Energy Act”. This is not assessable at this stage, and although not expected to impact the design, will need to be further investigated in the subsequent steps of the project.

# SUMMARY AND CONCLUSIONS

## ASSIGNED RISK CLASSES

As a result of conducted compliance assessments no high risk documents were identified.

Some documents were categorized as “**Medium Risk**” as additional design analyses or evaluations would be required to fully comply with specific requirements or guidelines presented in them.

Majority of the documents were assigned a “**Low Risk**” category. Despite the fact that some of the requirements were met as “compliance with objective” or a site-specific scope needs to be done, no additional analyses or design changes are anticipated.

**Table 3-1** presents the summary of risk classification for the received regulations, requirements and guidelines.

**Table 3‑1 Summary of assigned risks classes**

|  |  |  |
| --- | --- | --- |
| BGP-GW-GL-201 | \*Act on the Safe Use of Nuclear Energy | Low |
| BGP-GW-GL-202 | \*Regulation on Radiation Protection | Medium |
| BGP-GW-GL-203 | \*Regulation on Ensuring the Safety of Nuclear Power Plants | Medium |
| BGP-GW-GL-204 | \*Regulation on Ensuring the Safety in Spent Fuel Management | Medium |
| BGP-GW-GL-205 | \*Regulation on Safe Management of Radioactive Waste | Medium |
| APP-GW-G0R-004 | \*GSR Part 3 Radiation Protection and Safety of Radiation Sources | Medium |
| APP-GW-GL-704 | \*GSR Part 4 Safety Assessment for Facilities And Activities | Medium |
| APP-GW-GL-059 | \*SSR-2/1 Safety of Nuclear Power Plants: Design | Low |
| APP-GW-G0R-005 | SSG-30 Safety Classification of Structures, Systems and Components In Nuclear Power Plants | Low |
| APP-GW-G0R-006 | SSG-34 Design of Electrical Power Systems for Nuclear Power Plants | Medium |
| APP-GW-G0R-007 | SSG-39 Design of Instrumentation and Control Systems For Nuclear Power Plants | Medium |
| APP-GW-G0R-008 | SSG-52 Design of the Reactor Core for Nuclear power plants | Low |
| APP-GW-G0R-009 | SSG-53 Design of the Reactor Containment and Associated Systems For Nuclear Power Plants | Low |
| APP-GW-G0R-010 | SSG-54 Accident Management Programmes for Nuclear Power Plants | Low |
| APP-GW-G0R-011 | SSG-56 Design of the Reactor Coolant System and Associated Systems For Nuclear Power Plants | Low |
| APP-GW-G0R-003 | SSG-61 Format And Content of the Safety Analysis Report For Nuclear Power Plants | Medium |
| APP-GW-G0R-012 | SSG-62 Design of Auxiliary Systems and Supporting Systems For Nuclear Power Plants | Low |
| APP-GW-G0R-013 | SSG-63 Design of Fuel Handling and Storage Systems For Nuclear Power Plants | Low |
| APP-GW-G0R-014 | SSG-64 Protection Against Internal Hazards in the Design of Nuclear Power Plants | Medium |
| APP-GW-G0R-015 | SSG-67 Seismic Design for Nuclear Installations | Low |
| APP-GW-G0R-016 | SSG-68 Design of Nuclear Installations Against External Events Excluding Earthquakes | Low |
| APP-GW-G0R-017 | SSG-69 Equipment Qualification for Nuclear Installations | Low |
| APP-GW-G0R-018 | SSG-70 Operational Limits and Conditions and Operating Procedures For Nuclear Power Plants | Low |
| APP-GW-G0R-019 | SSG-73 Core Management and Fuel Handling For Nuclear Power Plants | Low |
| EPS-GW-GL-701 | \*WENRA Report Safety of New NPP Designs, RHWG, 2013 | Low |
| BGP-GW-GL-206 | Energy Act | Low |
| BGP-GW-GL-201 | \*Act on the Safe Use of Nuclear Energy | Low |

## Main risks to be addressed in Bulgaria project

This subsection covers main risks and topics for further discussion that were identified in the regulations, codes and standards marked by the customer as mandatory to comply with. Additionally, this subsection covers risks and topics for further discussion that resulted in “Medium Risk” category for some regulations, codes and standards outside of the “mandatory” scope.

### Risks and topics for further consideration

#### Site Assessment and External Hazards

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Articles 33 (5), 80 (5), 81, 83 (4, 6-7), 84 (2), 85, 86, 125), “Bulgarian Regulation Related to Ensuring the Safety in Spent Fuel Management” (Article 23 (Points 1-2)).***

The design of the AP1000 plant nuclear safety systems and engineered safety features includes allowances for natural environmental disturbances such as earthquakes, floods, and storms at the station site. The nuclear island structures, including Fuel Handling and Storage Area, are designed to withstand the effects of natural phenomena such as hurricanes, floods, tornadoes, tsunamis, and earthquakes without the loss of capability to perform safety functions.

Design for natural phenomena is based on the industry standards and the SSCs vital to the shutdown capability of the reactor are designed to withstand the envelope of probable natural phenomena described in DCD Chapter 2.

In the event that site-specific external events are identified for a specific site that are not bounded by the existing AP1000 plant design analyses, additional analysis is performed to demonstrate either acceptability of the design or if necessary and possible, the design will be reconciled for new conditions, depending on a site-specific data and site-specific regulatory expectations about recurrence intervals.

As described in DCD Section 2.2, site-specific analyses of accidents and events external to the nuclear plant that are potential hazards within the site vicinity are to be performed by the Owner.

Due to robust design solutions for AP1000, it is expected that there will be no major design changes needed due to site-specific characteristics and hazards. However, this may require new analyses to confirm that there are no hazards which could affect the safety of the AP1000. If a phenomena are identified for which the probability of a severe consequence is not enveloped by the standard design, specific changes to the AP1000 plant analysis or design will be identified.

#### Site Compatibility

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Articles 3 (1), 4 (3), 28).***

Additional Analyses Needed.

The compatibility of the site and site conditions are studied in report KZG-GW-GL-100 (Reference [9] of the compliance assessment report [33]) and related to seismic and geotechnics in report KZG-GE01-X7R-001 (Reference [10] of the compliance assessment report [33]). Full compatibility of the proposed units with the conditions on site will need to be demonstrated taking into consideration the recommendations in these reports.

Thus, as specified in previous sections recommendations about additional analyses are to be addressed. This includes some geotechnical, seismic, site characterization and clarification on external hazards (natural and technogenic).

#### Aircraft crash

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Articles 33 (5), 85, 86).***

Westinghouse has performed a rigorous assessment of the AP1000 plant design to demonstrate that the plant’s design features, and functional capabilities provide inherent protection against the effects of an aircraft impact, this assessment count with regulatory approvals that will need to be validated by BNRA.

#### Multiple NPP’s on site

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Articles 30 (2), 174).***

Selected site includes existing nuclear facilities. This may need further considerations in the design and safety analyses.

#### Subcriticality analysis in the spent fuel pool.

***Applicable to: “Bulgarian Regulation Related to Ensuring the Safety in Spent Fuel Management” (Article 113 (Point 8)).***

**Article 113**. **Point 8** of“Bulgarian Regulation Related to Ensuring the Safety in Spent Fuel Management” on credit for soluble boron for subcriticality analysis in the spent fuel pool. Generally, no soluble boron is required to maintain subcriticality below Keff<0.95 in the Spent Fuel Pool for most the credited accidents. In the case of a mislocated or misloaded maximum enrichment fresh fuel assembly, to maintainKeff<0.95, the analysis concludes that 800 ppm of boron is the maximum required soluble boron concentration for the most limiting condition.

Indeed, the criticality analyses performed for the AP1000 plant design and which support the certification of the standard design reflect two requirements:

* + - First, keff must be maintained less than or equal to 0.95 with the Spent Fuel Pool loaded with fuel of the maximum fuel assembly reactivity when crediting boron for normal operation as well as design basis events such as misloading or mislocated fuel.
    - Second, it must be shown that keff will remain below 1.0 in the unlikely event of a complete loss of boron.

The double contingency principle outlined in Section 4.2.2 of ANSI/ANS-8.1-1998;R2007;W2014, “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors” indicates that it should take two independent, unlikely, and concurrent events for a criticality to occur. This combination of events is considered outside of the design basis of the plant.

Because soluble boron is not credited to ensure that all normal operating conditions remain subcritical (keff< 1.0 at 0 ppm soluble boron), the soluble boron present in the pool can be credited to offset the reactivity impacts associated with accident conditions. For accidents that are independent of a soluble boron dilution event, the full minimum soluble boron requirement in the Technical Specifications can be credited. For accidents which are not independent of the soluble boron accident (they share a common mode initiator), soluble boron credit can be taken up to the minimum credible concentration identified in the boron dilution analysis.

Dilution of the spent fuel pool water to below this concentration is considered highly unlikely.:

* + - The Spent Fuel Pool is initially filled with water at a nominal boron concentration of 2700 ppm.
* For accidental conditions added to a severe spent fuel pool inadvertent dilution such as in the case of a mislocated or misloaded maximum enrichment fresh fuel assembly, to maintain Keff<0.95, the analysis concludes that 800 ppm of boron is the maximum required soluble boron concentration for the most limiting condition of a misloaded assembly in region 2 (and 650 ppm in the case of a mislocated assembly). Also, in region 1 in the case of a mislocated assembly 250 ppm are credited.
  + - Demineralized water can be added for makeup purposes (e.g., to replace evaporation losses from the demineralized water transfer and storage system). Boron concentration in the Spent Fuel Pool during operation is monitored via manual sampling occurring at a minimum of every 7 days. If the sampling shows any decrease in boron concentration outside Technical Specification limits, immediate actions are required to be taken. If needed, boron is added to the Spent Fuel Pool from the chemical and volume control system (CVS).
    - Dilution of the Spent Fuel Pool cannot be caused by a single design basis external or internal event, such as flooding. Furthermore, it is not expected that a beyond design basis (BDB) external flood event could cause a boron dilution of the SFP. Indeed the top of the Spent Fuel Pool is located at elevation 135 ft (110.7 m) compared to the design basis flooding which is defined as elevation 100 ft (100 m). This provides a very large margin even against BDB flooding.
    - Therefore, the only credible initiator of a dilution in the Spent Fuel Pool is operator error such as opening manual valves. There are different ways of detecting a dilution event. Alarms are provided in the control room for a high level in the Spent Fuel Pool, water detected in normally empty tanks, high levels in normally full tanks as the Spent Fuel Pool water spills into the adjacent tanks, and low levels in the dilution sources. There will be either high-ranked alarms or multiple alarms for any Spent Fuel Pool dilution events. In addition, these specific alarm combinations are unique for this event case, and the operators will repeatedly see this event at their simulator training. Taken together, these items make it unlikely that the operators will not interpret the alarms correctly.

It should also be noted that the water in the spent fuel pool will become intimately mixed with the water in the reactor coolant system during fuel handling and refueling operations. As such, the spent fuel pool must remain borated to ensure subcriticality in the reactor even if no credit is taken for soluble boron in the spent fuel pool criticality analyses.

However, if this accidental condition must be considered in coincidence with the severe inadvertent boron dilution in the spent fuel additional measures can be taken in the design.

Additional measures may include elimination of the number of available spaces in Rack 2 storage; including caps on some positions in the spent fuel pool which will not allow the insertion of elements in such positions; or if so desired, but not envisaged, proceed to a redesign of Racks in Region 2 similar to the Racks in Region 1 (so as not to need soluble boron).

Input from the Customer for future discussions of this topic is Included as Attachment 1 to [34].

#### Radiological Analyses needed for Planned Exposure and Emergency Exposure.

***Applicable to: “Bulgarian Regulation on Radiation Protection” (Articles 11, 13, 15, 18(1-2), 98(3), 103(1), 135 (1)); “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Articles 2 (1), 4 (3), 43), “Bulgarian Regulation Related to Ensuring the Safety in Spent Fuel Management” (Article 4 (Points 1-4), “Bulgarian Regulation Related to Safe Management of Radioactive Waste” (Article 8), “IAEA Safety Standards Series, General Safety Requirements No. GSR Part 3 Radiation Protection and Safety of Radiation Sources” (Requirement 12, 19, 29, 30).***

***A complete set of radiological impact analysis against dose limits and dose constrains for the public and workers are needed. See assessments.***

**For planned Exposure (Normal Operation)**

The AP1000 was designed attending the US-NRC regulatory framework. This regulatory framework is mainly collected in the following: Title 10, Part 20, of the Code of Federal Regulations (10 CFR Part 20), "Standards for Protection Against Radiation"; 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; and Appendix I to 10 CFR Part 50, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

In relation to Assuring that Occupational Radiation Exposures Are As-Low-As-Reasonably Achievable in AP1000 design:

* AP1000 design compliance with 10 CFR 20 is confirmed for the design and operation of the facility following the guidelines of Regulatory Guides 1.8, 8.8, and 8.10. Compliance with Regulatory Guides is addressed as discussed in DCD [28] subsection 12.1.3 The design of AP1000 meets the guidelines of Regulatory Guide 8.8, Sections C.2 and C.4, which address facility, equipment and instrumentation design features. Features of the plant that are examples of compliance with Regulatory Guide 8.8 are delineated in Section 12.3.
* As explained in DCD [28] subsection 12.1.3 the operational considerations of ALARA, as well as the related policies, programs and operational procedures that grant compliance with occupational exposure are to be addressed under the responsibility of the Licensee (the Nuclear Power Plant Operator), this includes the operational considerations included in Standard Review Plan (NUREG-0800), Regulatory Guides that will be addressed include: 8.2, 8.7, 8.9, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38.

It must be noted that NRC Regulation on Radiation Protection has not been updated to the ICRP Publication 103, reference [11] of [32], see attachment included to this assessment with a more detailed explanation. However, the US-NRC believes that NRC's current regulatory framework continues to adequately protect the health and safety of nuclear industry workers, the public, and the environment.

**The AP1000 Standard Design** contained in the DCD [28] and Vogtle Reference plant for Bulgaria Project **Licensing Basis are based on this US-NRC Regulatory Framework. However, this will not be considered as Non Compliance for the project,** since the extensive use of ALARA considerations in the Design, and existing analysis, shows that the reduction of some limits is achievable with the current design once Operational Radiation Programs are set in place.

A more detailed Assessment can be found in [32].

**For Emergency Exposure (Accidental Conditions)**

Although the NRC requirements seem to be less strict, this is not necessarily true due to the conservatisms that are included in the U.S NRC approach (mainly the requirement that the source term is based on a deterministic core melt accident, a significantly more conservative source term). Note that the U.S NRC requires the dose assessment for LOCAs to postulate a source term consistent with full core melt, despite the safety analysis prediction of limited fuel rod failures and the maintenance of coolable geometry. The US analysis requirements, which are thus significantly more conservative, have correspondingly higher limits than in other countries and in some cases; this is the reason that the DCD results would not comply with the limits prescribed in other countries, but the reason, as explained above, is simply due to the different licensing methodology and terminology used in dose analyses in the US.-specific analyses.

A second set of analyses have been performed in support to licensing or pre-licensing efforts in other geographies, for example in Czech Republic or in the United Kingdom, especially during the licensing process in the United Kingdom, the UK regulator requested that Westinghouse calculate the radiological consequences of design basis faults using methods and assumptions consistent with relevant UK practice, which tend to be more realistic than U.S. methods. Dose calculations include contributions from inhalation and immersion and also from the activity deposited on the ground. Doses are calculated not only at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ) and in the MCR, but also to any individual onsite. Doses at all locations are calculated for the duration of releases and are reported for the adult age groups. The contribution to the doses from the activity deposited on the ground is based on a full year of exposure to the contaminated ground.

These assessment calculations provide confidence that the AP1000 plant can meet the dose limits provided in Bulgarian regulations by performing additional calculations based on appropriate assumptions for licensing in Bulgaria and having appropriate radiation programs in place.

According to **Article 4 ( Point 4)** of *“Bulgarian Regulation on Ensuring the Safety of Spent Nuclear Fuel Management”* frequency for large radioactive releases to the environment for which it is necessary to take urgent protective measures for the public shall not be greater than 1.10-6 events per facility in one year. The AP1000 Spent Fuel Pool has lower large radioactive release frequencies. Current analyses for Fuel Damage in the Pool are bellow this limit, however an updated Spent Fuel Pool analysis will need to be performed at a later stage.

**According to Article 8 (position 1) of “*Bulgarian Regulation on Safe Management of Radioactive Waste*”,** the individual effective dose for the respective critical group of members of the public resulting from RAW management activities and/or following the normal operation of all nuclear facilities located on a single site shall not exceed 0.15 m in a year for new facilities and 0.25 mSv in a year for existing facilities;

Feasibility of these limits was discussed in Feasibility Study subsections II-2.1.1 Evaluation Criteria and II-2.1.2 Evaluation of the Capacity of the Site (Reference [6] of the compliance assessment report [35]). However, specific analyses may need to further substantiate this area of the plant design.

#### Effluent and Radiation monitoring and control

***Applicable to: “Bulgarian Regulation on Radiation Protection” (Articles 56 (2), 100), “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Article 234 (2)).***

The radiation monitoring system plus the portable equipment is sufficient to monitor the relevant radiological parameters. However some adjustments of setpoint of these monitors might be requested by the specific requirements for the Bulgaria project.

AP1000 plant is designed in a way that all liquid and gaseous release points are monitored using continuous radiation monitoring system RMS. AP1000 Effluent monitoring for Bulgaria shall be in agreement with local requirements, thus most likely it should conform with Recommendation 2004/2/Euratom which provides guidance to EU countries on the reporting of discharges of radioactive nuclides. Thus Some minor design changes might be requested for some plant and effluent monitors, however since they are considered as minor, thus this change has been maintained as a medium risk in the affected regulations since the small impact of this design change.

#### Requirements Arising from Obligations of the Republic of Bulgaria under International Treaties Ratified, Promulgated and in Force for the Republic of Bulgaria.

***Applicable to: “Bulgarian Regulation Related to Ensuring the Safety in Spent Fuel Management” (Article 34).***

**Article 34** of the Regulation specifies that the design of the facilities shall ensure the fulfilment of the requirements related to the control, storage, on-site transportation and physical protection of nuclear material arising from the obligations of the Republic of Bulgaria under international treaties ratified, promulgated and in force for the Republic of Bulgaria.

The AP1000 plant design is generally expected to support compliance to most of the stated requirements and regulations (either directly or through sites specific modifications). Some of the requirements are not in the design scope (e.g. transportation) or are the responsibility of the Owner. These include design requirements related to Physical Protection, and Safeguards.

No design changes are currently expected to be required in the AP1000 standard design to meet obligations arising from international treaties signed by Bulgaria. As previously stated additions to ensure these requirements might however be needed; but this will have to be confirmed.

#### Physical Protection

***Applicable to: “Bulgarian Act on the Safe Use of Nuclear Energy” (Chapter 7).***

The Design Basis Threat will need to be reviewed to determine AP1000 plant compliance.

Convention on the Physical Protection of Nuclear Material (CPPNM), the need for a Physical Protection Program that needs to accompany Licensee’s Request, and the need to assess and adapt to the Design Basis Threat.

**Article 112 (2)** states: The system of physical protection of nuclear facilities, as well as the system of physical protection of nuclear material in transport, shall be designed and the effectiveness thereof shall be evaluated in accordance with the design basis threat.

The AP1000 plant design supports this requirement by the plant physical protection system. The AP1000 plant physical protection system is designed to protect the site against malevolent attempts to commit radiological sabotage using elements of the US specific DBT in accordance with the rules laid out in 10 CFR 73.55 as clarified in Reg. Guide 5.69. It is also designed to support on site security in the assessment and detection of threats as well as supporting threat neutralization (armed officer engagements) as directed by 10 CRF 73.55.

The four functional objectives of the physical protection system: deterrence, detection, delay (denial), and response are designed to support a variety of different strategies. The physical protection system is designed to provide detection capability at critical locations throughout the plant, provide delay features to deter intruders, and provide the ability to efficiently respond to the threat using whatever means are allowed by regulation.

Specific characteristics and attributes defined within Design Basis Threat must be reviewed to determine if there are any potential impacts to the physical security system or any other elements of the AP1000 plant structure are impacted.

**Article 113(4**) requires the issuance of a regulation that shall contain the requirements for Physical Protection. This regulation seems to be in the BNRA website under the title “Regulation for the Provision of Physical Protection of Nuclear Facilities, Nuclear Material and Radioactive Material, adopted by CM Decree № 283/19.10.2015, promulgated, SG No. 82/23.10.2015”. Reference [8] of the compliance assessment report [31]. This likely includes the requirements for the owner to be included in the procedures and design. Westinghouse considers this requirement as not assessable at this stage of the project since it is available only in Bulgarian language and has not been selected among the 26 regulations to be analyzed for this stage of the project. This regulation will need to be assessed at later stages by the owner to determine if there are specific needs related to physical protection that need to be included in the plant. It also states the Confidential Nature of the Physical Protection Information in agreement with that given by the Classified Information Protection Act.

#### Application of Safeguards

***Applicable to: “Bulgarian Act on the Safe Use of Nuclear Energy” (Chapter 9), “IAEA Specific Safety Guide No. SSG-63: Design of Fuel Handling and Storage Systems for Nuclear Power Plant (Requirement 2.16).***

The current AP1000 plant design and general infrastructure is expected to support the addition of technical safeguards, as approved by Euratom and IAEA, to fully measure, monitor and record nuclear material subject to safeguards both via on- site inspections and remote monitoring; however, these systems are not currently provided in the design.

Westinghouse preparation to be involved in the Safeguards by Design Dialog that need to involve (IAEA, Bulgarian Authorities, NPP Operator, and Vendor), as well as support the data needed to prepare IAEA Form N-72 – IAEA Design Information Questionnaire Research and Power Reactors (Reference [11] of the compliance assessment report [31]).

The current AP1000 plant design does not address this beyond the general plant layout and basic physical barriers. The physical layout includes doors, walls, and other physical barriers and the physical protection system provides a backbone for monitoring their inviolability and integrity.

It has to be noted that Article 126 requires the issuance of a regulation which shall establish the terms and the procedure for collection and provision of information and for the keeping of registers on the activities comprehended in the application of safeguards. This regulation seems to be in the BNRA website only in Bulgarian Language under the title “Regulation on the Application of the Safeguards under the Treaty on the Non-proliferation of Nuclear weapons, adopted by CM Decree № 244/27.10.2017, promulgated, SG No. 88/3.11.2017” (Reference [12] of the compliance assessment report [31]). This regulation has not been selected among the 26 regulations to be analyzed for this stage of the project. Thus, it is out of the scope of this assessment.

#### Egress Studies

***Applicable to: “Bulgarian Regulation Related to Ensuring the Safety in Spent Fuel Management” (Article 30 (2)), “IAEA Specific Safety Guide No. SSG-64 "Protection against Internal Hazards in the Design of Nuclear Power Plants” (Item II-22).***

AP1000 plant has an egress study for the nuclear island, that takes into account the potential evacuation route, on the understanding of the unique nature of a nuclear power plant facilities dominated by equipment areas and where regularly occupied areas are ancillary.

Hence, AP1000 plant is to be in compliance with the pertinent Occupational Safety and Health Administration (OSHA) and National Fire Protection Association (NFPA) standards unless compliance interferes with the plant’s ability to implement or maintain nuclear safety, support nuclear safeguards requirements, or meet licensed design features due to space limitations. Areas where AP1000 plant design deviates from NFPA 804, 101 and OSHA are justified and documented.

As appropriate, site-specific studies will need to be performed to review applicable Bulgarian life-safety regulations/standards and assess the specific compliance of the AP1000 NPP design, as well as define and justify necessary exemptions to those requirements pursuant to the requirements of nuclear safety and security.

#### Remote shutdown station

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Articles 114 (3), 211 (1)).***

The remote shutdown workstation is provided for control of the plant in the case of an evacuation of the main control room (mainly due to fire). The remote shutdown station is not needed for other internal and external events.

Additionally, a secondary diverse actuation is in a diverse spatially separated location (not in the same zone of the plant) to actuate key safety functions such as ADS Stage 4 actuation, IRWST injection and containment recirculation actuation. The secondary DAS panel is powered by an independent local battery. The secondary DAS panel is located sufficiently far from the Main Control Room (MCR) and Remote Shutdown Workstation), its location has been selected as to provide additional protection so that it is very unlikely that it could be affected by internal events such as fire, internal flooding, or external events such as flooding (thus providing additional protection from these events).

This is considered complying with the objective, hence, no design changes to AP1000 standard design is expected. This topic is proposed to be follow-up to ensure this approach and discard potential additional analyses or design change. At this stage this is considered a Medium Risk.

#### Containment isolation for instrument lines

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Article 130 (1)) and “IAEA Specific Safety Guide No. SSG-53 "Design of the Reactor Containment and Associated Systems for Nuclear Power Plants” (Requirements 4.157 and 4.158).***

Containment Isolation related to four instrumentation lines is demonstrated according to Regulatory Guide 1.11. and alternate criteria.

No design changes to AP1000 standard design or additional analyses are expected.

#### Fire response classes (A1, A2)

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Article 139 (2)).***

Fire response class A1 or A2 is required. This has been classified as “NAS” (Not Assessable). This might need to be studied to understand it potential impact. At this moment it is understood as “Low Risk”, since AP1000 Fire protection approach is expected to be maintained.

#### Development of Safety Assessment Report (SAR)

***Applicable to: “Bulgarian Regulation Related to Ensuring the Safety in Spent Fuel Management” (Article 6).***

Development of a plant-specific SAR (and all its mandatory content) is the responsibility of the Owner, however Westinghouse provides the inputs to the Owner as appropriate and contracted, on AP1000 design, construction and operation items based on experience as the plant designer.

#### Decommissioning

***Applicable to: “Bulgarian Regulation Related to Ensuring the Safety in Spent Fuel Management” (Articles 16 (Point 4), 80 (Point 2), 96 (Point 1), 110 (Point 3)), “IAEA Specific Safety Guide SSG-61 “Format and Content of the Safety Analysis Report for Nuclear Power Plants” (Chapter 21), “IAEA Safety Standards Series, General Safety Requirements No. GSR Part 4, Safety Assessment for Facilities and Activities” (Item 1.8, 4.16, 4.18, 4.37, 4.42), “IAEA Safety Standard No. SSR-2/1 - Safety of Nuclear Power Plants: Design” (Requirement 12).***

The DCD does not contain information on decommissioning. At later stages of the project a specific decommissioning plan might be requested and provided by Westinghouse upon the further agreements.

The AP1000 plant is designed to be amenable to decommissioning, with features that both facilitate decommissioning and minimize scope and cost. These features include the following:

- Minimization of radioactive waste generation;

- Minimization of activation of metallic components and contamination of plant systems, structures, and components (SSCs), including material selection that prevents large quantities of long-lived radionuclides from forming (such as using low cobalt steel);

- Control of activation (e.g., controlling primary circuit water chemistry);

- Physical and procedural methods that prevent the spread of contamination; Consideration of decommissioning when proposing plant modifications;

- Identification of reasonably practicable changes to the facility to facilitate or accelerate the decommissioning process.

Westinghouse has developed and provided extensive decommissioning feasibility studies, specifically in support of the AP1000 licensing review in the United Kingdom, where decommissioning assessment for the AP1000 plant was performed in the context of the UK Generic Design Assessment (GDA) to demonstrate that:

1. It is feasible to decommission an AP1000 plant using current technology, and

2. Decommissioning issues were appropriately considered in the overall design.

Additional Information for Decommissioning might have to be produced specially taking into account the requirements with regard to the ISAR contents in APPENDIX № 1 to Article 40, Paragraph 1, Subparagraph 1, “a” of the Bulgarian Regulation on the Procedure for Issuing Licenses and Permits for the Safe Use of Nuclear Energy.

#### Defense-in-depth

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Articles 41 (4), 57 (1), 112 (3))***

Defense-in-depth is also widely discussed in WENRA Report Safety of new NPP designs, RHWG, 2013 and SSR-2/1 Safety of Nuclear Power Plants: Design (2.12 to 2.14 and 4.9 to 4.13A)

Related to SSCs as stated in Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants [33].

There are SSCs which are used both during transients and during accidents with fuel melting, such as Passive Containment Cooling System. However, operation of these systems during transients does not affect the operation during accidents with fuel melting.

Due to the passive features with high reliability in AP1000 design, it is considered that objective of this requirement is met. This is confirmed by deterministic and probabilistic safety analyses (DCD sections 15 and 19).

No design changes to AP1000 standard design or additional analyses are expected.

#### Multiple failures

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Article 49 (2)).***

Common cause failures are considered in AP1000 design by introducing diversified safety functions and SSCs to strengthen the defense-in-depth concept, e.g., diverse means for core cooling, diversity in automatic depressurization and diversity in actuation system.

Common cause analysis is included in the AP1000 plant PRA as stated in the AP1000 plant DCD Section 19.29. The PRA was used to define where and to what degree diversity needed to be incorporated into the AP1000 plant SSCs.

No design changes to AP1000 standard design or additional analyses are expected.

#### Safety classification

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Article 54 (2-4)), “IAEA Specific Safety Guide SSG-30: Safety Classification of Structures, Systems and Components In Nuclear Power Plants (In the Safety Classification Process).***

Safety classification principles for AP1000 design is presented in DCD section 3.2. The classification system provides a means of identifying the extent to which structures, systems, and components are related to safety-related and seismic requirements. The classification of SSCs is slightly different than required in this article, e.g., safety class C includes components, which provides safety support functions to Class A, B and C SSCs.

The AP1000 classification system provides a means of identifying the extent to which structures, systems, and components are related to safety-related and seismic requirements. The classification system provides an easily recognizable means of identifying the extent to which structures, systems, and components are related to ANS nuclear safety classification, NRC quality groups, ASME Code, Section III classification, seismic category, and other applicable industry standards, as shown in DCD table 3.2-3.

No design changes to AP1000 standard design or additional analyses are expected.

#### Support functions own protections

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Article 137 (4)).***

Due to the passive features of the AP1000 design, the number of supporting safety systems needed for accident management is limited, since main safety functions are performed by passive systems. Thus, active components are not classified as safety related to perform safety functions (they are included in the Defense in Depth type of Systems, thus not safety-related), for the previous reason the need to preclude their own protections is not as relevant and is to be analyzed only for Probabilistic Analyses.

No design changes to AP1000 standard design or additional analyses are expected.

#### Waste categorization

***Applicable to: “Bulgarian Regulation Related to Safe Management of Radioactive Waste” (Article 6).***

Waste categorization in Bulgaria requires consideration in the site-specific operational programs for radwaste management. This should also specifically apply in the case of needing to develop a SRTF.

#### Probabilistic Risk Analysis (PRA) need for Additional Analyses

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Articles 44 (2), 49 (6), 79 (1-4)).***

Additional Analyses Needed.

External hazards are considered in the AP1000 design by structural design solutions and passive design features. Similarly, spent fuel pool design is designed so that structural integrity is always confirmed and spent fuel pool cooling is always ensured. For these reasons, it is expected that there are no design changes needed in AP1000.

Probabilistic safety analyses are documented in DCD section 19, and it confirms that AP1000 design is robust against different initiating events and hazards with high safety margin. However, these topics may require further development needs in the PRA models and analyses. In addition, site-specific data may require adjustments in the PRA models.

Thus PRA might need additional scopes and updates that might include apart to updates to At Power Scope (Internal Events, Internal Flood, Internal Fire, Seismic) scopes like External Hazards, Low Power and Shutdown, Spent Fuel Pool, Multi Unit, Level 3.

#### Management of large quantities of liquid RAW generated in accident conditions.

***Applicable to:*** ***“Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Article 147 (4).***

**Article** **147 (4)** of the “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants”specifies that “The design shall specify the way for management of large quantities of liquid RAW generated in accident conditions”. This has been classified as owner requirement, since it does not affect AP1000 Standard Design.

However the extent of this requirement could potentially add some additional scope that is currently not considered. This needs to be fully understood.

#### Cogeneration (District heating)

***Applicable to: “Bulgarian Energy Act” (Article 84 (6)), “IAEA Safety Standard No. SSR-2/1 - Safety of Nuclear Power Plants: Design” (Requirement 35), “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Section X).***

AP1000 Standard Design or Reference Plant do not consider District Heating/Heat Generation. Nevertheless, in case heat production were requested, this might be feasible but will request a specific scope to be developed.

#### Commissioning tests

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Article 173 (1-2)).***

Commissioning tests will be provided as presented in DCD section 14. Test program will cover all necessary tests but with different test structure.

No design changes to AP1000 standard design or additional analyses are expected.

#### Inspection, maintenance, testing

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Articles 168 (1), 169 (1-3), 170 (1-2), 171, 172 (1-2), 173 (3-4), 217 (1-3), 218 (3), 221 (1-2), 226 (1), 227 (1-2)).***

Specific/Additional Documentation might need to be produced.

AP1000 plant meets the design related to taking into consideration inspection and maintenance programs. Nonetheless the licensee shall prepare and implement documented programs of maintenance, testing, surveillance, and inspection of SSCs important to safety.

Westinghouse will provide input to the designer for maintenance, testing, surveillance, and inspection developed for AP1000 plant SSCs for the Owner to develop their programs.

The initial test program is described in DCD chapter 14. Detailed commissioning program will be specified more in detail at later stages.

#### Operating Procedures, Emergency procedures & SAMGs

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Articles 188 (2-3), 189 (1-2), 190 (1-3), 191 (1-2), 192 (1-2), 203 (2-3), 206 (2)), “IAEA Specific Safety SSG-54: Accident Management Programmes for Nuclear Power Plants"***

Specific/Additional Documentation might need to be produced. No design Changes to Standard Design expected.

Emergency procedures and SAMGs will be prepared at later stages. Standard AP1000 Procedures will be used as input.

#### Chemistry program

***Applicable to: “Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants” (Article 213 (1-2)).***

Specific/Additional Documentation might need to be produced. No design Changes to Standard Design expected.

The chemistry program shall be prepared at later stages. It will be based on AP1000 standard Chemistry Manual, procedures, and specifications.

#### Lightning and surge protection

***Applicable to: “IAEA Specific Safety Guide No. SSG-34 "Design of Electrical Power Systems for Nuclear Power Plants” (Items 187,188,190).***

According to Articles 187 and 190: “internal lightning protection will normally include shielding and surge arresters to protect against both the induced high voltage caused by the lightning current and high transferred voltage. High transferred voltages are caused by voltage differences between the ground and parts of the external lightning protection system and the associated grounding connections. The internal protection grounding should be connected to the rest of the lightning grounding in such a manner that it protects personnel and equipment against high transferred potentials.”.

For AP1000 plant internal surge protection is provided for some power circuits. The extent of protection needs further evaluation.

In addition, according to Article 188: “To protect the safety power system from induced voltages, safety classified raceways and cables should not be located close to the outer walls of buildings.”

In AP1000 plant the Class 1 safety power system equipment is located internally in the auxiliary building. A portion of the cabling is routed internally. The extent of internal routing needs further evaluation in case this guidance is pursued.

#### I&C Impacts

***Applicable to: “IAEA Specific Safety Guide No. SSG-39 " Design of Instrumentation and Control Systems for Nuclear Power Plants”.***

**Common cause failures**

This risk is also related to the risk addressed in **Subsection 3.2.1.18.**

Statement presented in **(Item 4.31)** recommends that “Justification should be provided for any identified common cause failures that are not considered in this assessment”. This recommendation is quite ambiguous. In general, in AP1000 design identified common cause failures had been considered in the AP1000 design, and evaluated in the PRA.

**(Item 4.32 and 4.34)** Recommendation 4.32 states that an analysis should be done of the consequences of each postulated initiating event within the scope of safety analysis in combination with the common cause failures that would prevent a protection system from performing the necessary safety functions. 4.34 states that if one of the analyses referred in 4.32 determines that a postulated initiating event in combination with a common cause failure of a protection system results in unacceptable consequences, the design should be modified.

It needs to be noted that:

Common cause failures are largely considered in AP1000 design by introducing diverse means to achieve safety functions and diverse SSCs to strengthen the defense-in-depth concept.

Additionally, Common cause analysis is included in the AP1000 plant PRA as stated in the AP1000 plant DCD Section 19.29 and it was used to define where and to what degree diversity needed to be incorporated into the AP1000 plant SSCs.

Nonetheless, in some cases further evaluation of this topic might be required.

At this time, this is considered a medium risk since no significant design changes are anticipated, however this might require more detailed analyses.

**(Item 6.14)** states that failures resulting from errors in design, maintenance, operations or manufacturing are not included in the analysis of compliance with the single failure criterion. Known errors should be properly addressed by means of the management system. The effects of unknown errors cannot be predicted and, thus, the single failure criterion is not a useful tool for understanding the effects of such errors on a safety group.

AP1000 is especially resilient to common cause failure. A paramount salient feature of the AP1000 design is the use of passive systems. In addition to redundancy, the passive features also provide diversity. this approach is very different from conventional nuclear power plants in which failures of active safety functions lead directly to core damage and therefore the designs must provide more independent systems.

An example of diversity is passive feed and bleed cooling provides a backup diverse means to the passive residual heat removal heat exchanger (PRHR HX) of removing decay heat from the reactor coolant system. In addition to the use of diverse mechanical components, these diverse decay heat removal features are both actuated by the protection (1E) I&C system (PMS) as well as a diverse defense-in-depth (DiD) I&C system (DAS).

Furthermore, the electrical power requirements for these features are also diverse. The PRHR HX is self-actuating and does not require electrical power. The passive feed and bleed capability uses dc power from safety batteries.

As explained in the example, The I&C safety related equipment performing reactor trip and ESF actuation functions (from PMS system), their related sensors, and the reactor trip switchgear are, for the most part, four-way redundant. This redundancy permits the use of bypass logic so that individual channel(s) out of service can be accommodated by the operating portions of the protection system while still performing maintenance 2oo3 (2-out-of-3) or 2oo4 (2-out-of-4) logic.

The DAS provides a diverse backup to the protection and safety monitoring system (PMS) in the case of a common-mode failure of PMS

The design Probabilistic Risk Assessment (PRA) for the standard AP1000 plant is APP-GW-GL-022 Revision 8, Chapter 29 describes Common-Cause Analysis and Attachment 29A provides Common-Cause Analysis Guidelines. The methodology, as used in the Reference Plant Vogtle PRA, for CCF has been peer reviewed and work towards meeting the requirements of Capability Category II of the ASME/ANS RA-Sa-2009 and Regulatory Guide 1.200, Revision 2.

This guideline might result in low or medium risk, depending on level of effort needed to further investigate this item.

**Item (6.68).** Requires that “The non-systematic failure modes of I&C components and systems should be known and documented”. This requirement has been considered for AP1000 safety-related system (PMS), failure modes and effects analysis (FMEA) was performed, see DCD Subsection 7.2.3. Therefore, there might potentially be a need for improved analysis for other important to safety systems.

**(Item 7.6)** claims that the consequences of common cause failure in sensors should be included in the analysis described in **(Items 4.30–4.34).** This might require further evaluation and analysis, however, no changes to the AP1000 plant design is expected.

**(Item 7.7)** states that no identified vulnerability to common cause failure of sensing devices should have the potential of denying operators the information and parameters that they need to control accidents and mitigate their consequences. This is per the related topics and other risks above and might potentially require further evaluation and analysis, however, no changes to the AP1000 plant design is expected.

All the mentioned above items are contributing to the “Medium Risk”.

**Items (7.95-7.99)** addressing “Independence of data communications” are also related to the Common Cause Failure discussed in other risk areas. Even though risk associated with those provisions is “Low”, it might end in some additional analyses requested depending of their interpretation.

**Decommissioning**

Following risk is linked to the risks described in **Subsection 3.2.1.16** of this document which addresses decommissioning.

Typical I&C development life cycle activities and interfaces with human factors engineering and computer security programs presented Figure 1 in **(Item 2.19)** of the reference document includes “Decommissioning”. DCD and UFSAR do not contain decommissioning provisions. However, Westinghouse can provide Owner with relevant inputs depending on the agreed project scope.

***Topics discussed below are not contributing to the increased risks level, however were identified as the topics that might need further consideration.***

**Content of design basis for instrumentation and control systems**

**(Item 3.16)** states that the **(Items 3.7-3.16)** “may be specified in either the overall I&C design basis or the design basis for individual systems. For some items, it might be appropriate to specify generic requirements in the overall I&C design basis and to provide more detail in the design basis for individual systems. In any case, the design bases for the overall I&C and for the individual systems should be consistent with each other, and the relationship and interfaces between the different design bases should be readily understandable.”

This statement addresses the guidelines or recommendations on the applicability of certain requirements, however, it does not fully address the underlying requirements themselves.

AP1000 Complies With the Objective of the expressed design basis however a different methodology and terminology is used. Hence, this might require further evaluation and possibly a more detailed decomposition of Requirements.

**Development verification and Validation and Life Cycle Processes**

**(Items 2.14 to 2.16).** The AP1000 plant design for I&C systems includes provisions for Validation and Verification V&V practices, and activities, See DCD 7.1.4.2, These include IEEE 1074-1995; “IEEE Standard for Developing Software Life Cycle Processes” and US-NRC RG 1.173 “Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants”. These standards Comply with the Objective set in this cited IAEA guides. However, if these IAEA guides were required to be met, further investigation and analyses might be needed, but no design change to AP1000 standard design is anticipated.

**Ambiguity of the general statements and guidelines presented in Specific Safety Guides.**

**(Items 7.151-154)** that are addressing “Software tools” are general recommendations and guidelines and thus cannot be assessed against specific provisions of the AP1000 I&C design. Because of the ambiguity of general statements and guidelines, there’s a likelihood for further inquiry, however this is currently considered a low risk.

**Human Factors**

Low risk is associated with **(Items 8.51-8.56)** and is related to human factors engineering for instrumentation and control systems. Those guidelines state that the design of the human–machine interface required for the supervisory control of safety systems should apply the principles of defense in depth and that the I&C system should provide operators with the information necessary to detect changes in system status, to diagnose the situation, to operate the system (when necessary) and to verify manual or automatic actions. In addition, mentioned above guideline items state that a satisfactory design will take into account the cognitive processing capabilities of operators as well as process-related time constraints, should ensure that the longest time from operating any control to when the input is acknowledged by the control system is acceptable to the operators. As well as that operator tasks can be performed within the time specified by system requirements.

This assessment properly addressed the impacted design topics in AP1000 design, however, based on the general assumptions of the original text (per the SSG-39), it's possible further inquiries might be raised, this is however considered a low risk.

**Software design**

Low risk is associated with guidelines presented in **(Items 9.21-9.25)** that are addressing simplicity of design of software for the safety systems and lower safety class systems. Those guidelines also state that software design architecture should be structured to allow for future modification, maintenance and upgrades, the software architecture should be hierarchical to provide graded levels of abstraction and that use of information hiding, where possible, is encouraged to enable piecewise review and verification and to aid modification.

Those guidelines might require further investigation based on the expectations of these items, regarding interactions between safety systems and non-safety (lower class) systems, and because the low level of technical details that it's expected. However, no design changes to AP1000 software design are expected.

**Predeveloped software**

Low risk is associated with “Predeveloped software” subsection containing **(Items 9.96-9.98).** The DCD does not include low level details about predeveloped software or expected behaviors of underlying software tools or components, used in the development of control application systems and software for the AP1000 I&C platforms. Thus, further investigation might be required based on potential inquiries.

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# REFERENCES

1. Bulgarian Act on the Safe Use of Nuclear Energy. Promulgated, State Gazette No. 63/28.06.2002, amended and supplemented, SG No. 120/29.12.2002, SG No. 70/10.08.2004, effective 1.01.2005, amended, SG No. 76/20.09.2005, effective 1.01.2007, SG No. 88/4.11.2005, SG No. 105/29.12.2005, effective 1.01.2006, SG No. 30/11.04.2006, effective 12.07.2006, SG No. 11/2.02.2007, amended and supplemented, SG No. 109/20.12.2007, effective 1.01.2008, amended, SG No. 36/4.04.2008, SG No. 67/29.07.2008, amended and supplemented, SG No. 42/5.06.2009, amended, SG No. 74/15.09.2009, effective 15.09.2009, amended and supplemented, SG No. 80/12.10.2010, amended, SG No. 87/5.11.2010, SG No. 88/9.11.2010, effective 1.01.2011, SG No. 97/10.12.2010, effective 10.12.2010, SG No. 26/29.03.2011, effective 30.06.2012, SG No. 38/18.05.2012, effective 1.07.2012, SG No. 82/26.10.2012, effective 26.11.2012, amended and supplemented, SG No. 15/15.02.2013, effective 1.01.2014, amended, SG No. 66/26.07.2013, effective 26.07.2013, SG No. 68/2.08.2013, effective 2.08.2013, SG No. 98/28.11.2014, effective 28.11.2014, SG No. 14/20.02.2015, SG No. 58/18.07.2017, effective 18.07.2017, SG No. 99/12.12.2017, effective 1.01.2018, amended and supplemented, SG No. 102/22.12.2017, effective 1.01.2018, supplemented, SG No. 103/28.12.2017, effective 1.01.2018, amended, SG No. 7/19.01.2018, SG No. 77/ 18.09.2018, effective 1.01.2019 amended and supplemented SG No. 17/25.02.2020.
2. Bulgarian Regulation on Radiation Protection. Promulgated, State Gazette SG No. 16/2018; amended and supplemented, State Gazette SG No. 110/2020.
3. Bulgarian Regulation on Ensuring the Safety of Nuclear Power Plants. Approved with a CM Letter No.245, dated 21.09.2016, promulgated SG, issue 76/ 30.09.2016, amended, issue 37/4.05.2018.
4. Bulgarian Regulation on Ensuring the Safety in Spent Fuel Management. Adopted by CM Decree No. 196 of 02 August 2004, promulgated, SG No. 71 of 13 August 2004, amended SG No. 76 of 30 August 2013, amended SG No. 4 of 09 January 2018, in force as of 09 January 2018, No. 37 of 04 May 2018.
5. Bulgarian Regulation on Safe Management of Radioactive Waste. adopted by CM Decree No. 185 of 23 August 2013, promulgated, SG No. 76 of 30 August 2013, amended SG No. 4 of 09 January 2018, in force as of 09 January 2018, No. 37 of 04 May 2018.
6. IAEA GSR Part 3 Radiation Protection and Safety of Radiation Sources, 2014.
7. IAEA GSR Part 4 Safety Assessment for Facilities And Activities, 2016.
8. IAEA SSR-2/1 Safety of Nuclear Power Plants: Design, 2016.
9. IAEA SSG-30 Safety Classification of Structures, Systems and Components In Nuclear Power Plants, 2014.
10. IAEA SSG-34 Design of Electrical Power Systems for Nuclear Power Plants, 2016.
11. IAEA SSG-39 Design of Instrumentation and Control Systems For Nuclear Power Plants, 2016.
12. IAEA SSG-52 Design of the Reactor Core for Nuclear power plants, 2019.
13. IAEA SSG-53 Design of the Reactor Containment and Associated Systems For Nuclear Power Plants, 2019.
14. IAEA SSG-54 Accident Management Programmes for Nuclear Power Plants, 2019.
15. IAEA SSG-56 Design of the Reactor Coolant System and Associated Systems For Nuclear Power Plants, 2020.
16. IAEA SSG-61 Format And Content of the Safety Analysis Report For Nuclear Power Plants, 2021.
17. IAEA SSG-62 Design of Auxiliary Systems and Supporting Systems For Nuclear Power Plants, 2020.
18. IAEA SSG-63 Design of Fuel Handling and Storage Systems For Nuclear Power Plants, 2020.
19. IAEA SSG-64 Protection Against Internal Hazards in the Design of Nuclear Power Plants, 2021.
20. IAEA SSG-67 Seismic Design for Nuclear Installations, 2021.
21. IAEA SSG-68 Design of Nuclear Installations Against External Events Excluding Earthquakes, 2021.
22. IAEA SSG-69 Equipment Qualification for Nuclear Installations, 2021.
23. IAEA SSG-70 Operational Limits and Conditions and Operating Procedures For Nuclear Power Plants, 2022.
24. IAEA SSG-73 Core Management and Fuel Handling For Nuclear Power Plants, 2022.
25. WENRA Report Safety of New NPP Designs, RHWG, 2013.
26. Bulgarian Energy Act. Promulgated, SG No. 107/9.12.2003, amended, SG No. 18/5.03.2004, effective 5.03.2004, amended and supplemented, SG No. 18/25.02.2005, effective 20.01.2005, amended, SG No. 95/29.11.2005, effective 1.03.2006, SG No. 30/11.04.2006, effective 12.07.2006, amended and supplemented, SG No. 65/11.08.2006, effective 11.08.2006, SG No. 74/8.09.2006, effective 8.09.2006, amended, SG No. 49/19.06.2007, amended and supplemented, SG No. 55/6.07.2007, effective 6.07.2007, amended, SG No. 59/20.07.2007, effective 1.03.2008, SG No. 36/4.04.2008, amended and supplemented, SG No. 43/29.04.2008, supplemented, SG No. 98/14.11.2008, effective 14.11.2008, amended, SG No. 35/12.05.2009, effective 12.05.2009, amended and supplemented, SG No. 41/2.06.2009, SG No. 42/5.06.2009, amended, SG No. 82/16.10.2009, effective 16.10.2009, SG No. 103/29.12.2009, amended and supplemented, SG No. 54/16.07.2010, effective 16.07.2010, amended, SG No. 97/10.12.2010, effective 10.12.2010, amended and supplemented, SG No. 35/3.05.2011, effective 3.05.2011, supplemented, SG No. 47/21.06.2011, effective 21.06.2011, amended, SG No. 38/18.05.2012, effective 1.07.2012, amended and supplemented, SG No. 54/17.07.2012, effective 17.07.2012, amended, SG No. 82/26.10.2012, effective 26.11.2012, SG No. 15/15.02.2013, effective 1.01.2014, supplemented, SG No. 20/28.02.2013, effective 28.02.2013, SG No. 23/8.03.2013, effective 8.03.2013, amended and supplemented, SG No. 59/5.07.2013, effective 5.07.2013, amended, SG No. 66/26.07.2013, effective 26.07.2013, SG No. 98/28.11.2014, effective 28.11.2014, SG No. 14/20.02.2015, amended and supplemented, SG No. 17/6.03.2015, effective 6.03.2015, SG No. 35/15.05.2015, effective 15.05.2015, supplemented, SG No. 48/27.06.2015, effective 30.06.2015, amended and supplemented, SG No. 56/24.07.2015, effective 24.07.2015, supplemented, SG No. 42/3.06.2016, amended and supplemented, SG No. 47/21.06.2016, SG No. 105/30.12.2016, supplemented, SG No. 51/27.06.2017, amended, SG No. 58/18.07.2017, effective 18.07.2017, amended and supplemented, SG No. 102/22.12.2017, effective 1.01.2018, supplemented, SG No. 103/28.12.2017, effective 1.01.2018, amended, SG No. 7/19.01.2018, amended and supplemented, SG No. 38/8.05.2018, effective 8.05.2018, amended, SG No. 57/10.07.2018, effective 1.07.2018, amended and supplemented, SG No. 64/3.08.2018, effective 3.08.2018, SG No. 77/18.09.2018, effective 1.01.2019, SG No. 83/9.10.2018, SG No. 91/2.11.2018, amended, SG No. 103/13.12.2018, effective 13.12.2018, amended and supplemented, SG No. 17/26.02.2019, SG No. 41/21.05.2019, effective 21.05.2019, SG No. 79/8.10.2019, effective 8.10.2019, supplemented, SG No. 25/20.03.2020, SG No. 38/24.04.2020, effective 24.04.2020, amended and supplemented, SG No. 57/26.06.2020, effective 26.06.2020, SG No. 9/2.02.2021, effective 2.02.2021, SG No. 21/12.03.2021, effective 12.03.2021, SG No. 8/28.01.2022, effective 1.01.2022, SG No. 9/1.02.2022, effective 1.02.2022, amended, SG No. 99/13.12.2022, effective 1.12.2022, SG No. 102/23.12.2022, effective 1.01.2023
27. L.KNP\_WEC\_230003, 8th of June 2023
28. APP-GW-GL-700, Rev. 19, “AP1000 Design Control Document,” Westinghouse Electric Company LLC. (Publicly available at: www.nrc.gov/docs/ML1117/ML11171A500.html)
29. Vogtle, Units 3 and 4, Rev. 11 to Updated Final Safety Analysis Report. NRC Accession Number Link: ML22179A145 (6462 page(s)). Date Released: Wednesday, June 29, 2022. (Publicly available at: https://www.nrc.gov/docs/ML2217/ML22179A145.html)
30. APP-GW-GL-022, Rev. 8, “AP1000 Probabilistic Risk Assessment”, Westinghouse Electric Company
31. BGP-GW-GL-201 Rev. A, Assessment of the AP1000 Plant for the Bulgarian Act on the Safe Use of Nuclear Energy
32. BGP-GW-GL-202 Rev. A, Assessment of the AP1000 Plant for the Bulgarian Regulation on Radiation Protection
33. BGP-GW-GL-203 Rev. A, Assessment of the AP1000 Plant for the Bulgarian Regulation Related to Ensuring Safety of Nuclear Power Plants
34. BGP-GW-GL-204 Rev. C, Assessment of the AP1000 Plant for the Bulgarian Regulation Related to Ensuring the Safety in Spent Fuel Management
35. BGP-GW-GL-205 Rev. A, Assessment of the AP1000 Plant for the Bulgarian Regulation Related to Safe Management of Radioactive Waste
36. APP-GW-G0R-004 Rev. A, Assessment of the AP1000 Plant to IAEA Safety Standards Series, General Safety Requirements No. GSR Part 3 Radiation Protection and Safety of Radiation Sources
37. APP-GW-GL-704 Rev. 2, Assessment of the AP1000 Plant to IAEA Safety Standards Series, General Safety Requirements No. GSR Part 4, Safety Assessment for Facilities and Activities
38. APP-GW-GL-059 Rev. 2, AP1000 Plant Design Comparison to IAEA Safety Standard No. SSR-2/1 - Safety of Nuclear Power Plants: Design
39. APP-GW-G0R-005 Rev. B, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-30 "Safety Classification of Structures, Systems and Components in Nuclear Power Plants"
40. APP-GW-G0R-006 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-34 "Design of Electrical Power Systems for Nuclear Power Plants”
41. APP-GW-G0R-007 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-39 "Design of Instrumentation and Control Systems for Nuclear Power Plants”
42. APP-GW-G0R-008 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-52 "Design of the Reactor Core for Nuclear Power Plants”
43. APP-GW-G0R-009 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-53 "Design of the Reactor Containment and Associated Systems for Nuclear Power Plants”
44. APP-GW-G0R-010 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-54 "Accident Management Programmes for Nuclear Power Plants”
45. APP-GW-G0R-011 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-56 " Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants”
46. APP-GW-G0R-003 Rev. A, Compliance Assessment of the IAEA Specific Safety Guide SSG-61 “Format and Content of the Safety Analysis Report for Nuclear Power Plants”
47. APP-GW-G0R-012 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-62 "Design of Auxiliary Systems and Supporting Systems for Nuclear Power Plants"
48. APP-GW-G0R-013 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-63 "Design of Fuel Handling and Storage Systems for Nuclear Power Plant"
49. APP-GW-G0R-014 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-64 "Protection against Internal Hazards in the Design of Nuclear Power Plants”
50. APP-GW-G0R-015 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-67 " Seismic Design for Nuclear Installations”
51. APP-GW-G0R-016 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-68 "Design of Nuclear Installations Against External Events Excluding Earthquakes”
52. APP-GW-G0R-017 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-69 "Equipment Qualification for Nuclear Installations"
53. APP-GW-G0R-018 Rev. A, Assessment of the AP1000® Plant to IAEA Specific Safety Guide No. SSG-70 "Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants"
54. APP-GW-G0R-019 Rev. A, Assessment of the AP1000 Plant to IAEA Specific Safety Guide No. SSG-73 "Core Management and Fuel Handling for Nuclear Power Plants”
55. EPS-GW-GL-701 Rev. C, AP1000 Plant Evaluation of Western European Nuclear Regulators’ Association Safety Objectives for New Power Reactors
56. BGP-GW-GL-206 Rev. A, Assessment of the AP1000 Plant for the Bulgarian Energy Act
57. APP-GW-G0R-001 Revision B. Compliance Assessment of the AP1000 Plant to EUR Revision E Key Issues. June 2022
58. L.WEC\_KNP\_230019, “Meeting Minutes from Completion of FEED Deliverables D.02.04 Meeting Minutes of Video conference workshop 1 on applicable regulations, regulatory guides, codes and standards for the Kozloduy Unit 7 new build project.” 8 September 2023.1
59. L.WEC\_KNP\_230025, “Action Items from Deliverable 02.04 Workshop”. 18 October 2023
60. BGP-GW-GEH-001, Revision A, “Preliminary Radioactive Waste Information for Bulgaria”.