European Nuclear Safety Regulators Group

EU Peer Review Report of the Türkiye Stress Tests

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1 INTRODUCTION AND BACKGROUND

On 11 March 2011, a magnitude 9.0 earthquake struck some 80 km off Japan's Tohoku coast. The ensuing tsunami and the subsequent accident at the Fukushima Daiichi Nuclear Power Plant (NPP) triggered the core melt of three reactors at the site. It was the worst emergency at a nuclear power plant since the Chernobyl disaster in 1986.

The analysis of the Fukushima accident revealed substantial, well-known and recurring technical issues: natural phenomena of a critical nature not being considered, faulty design, insufficient backup systems, failure to introduce safety improvements to operating reactors, human error, inadequate contingency plans, confusion in the response to a severe accident and poor communications. These points are clearly described in the International Atomic Energy Agency's (IAEA) comprehensive report on Fukushima published in September 2015[1].

[1] https://www.iaea.org/newscenter/news/iaea-releases-director-generals-report-on-fukushima-daiichi-accident

2 EU – STRESS TESTS AND FOLLOW-UP

- ***Mandate***

Against the backdrop of Fukushima, and based upon a mandate given by the European Council at its meeting of 24-25 March 2011, the European Commission (EC) – together with the **European Nuclear Safety Regulators Group** (**ENSREG**) – launched in 2011 an EU-wide comprehensive risk and safety re-assessments of all EU NPPs. These individual re-assessments are referred to as **stress tests (STs)**.

In its request, the European Council specified that the stress tests must first be carried out at national level and that they would then be complemented by a European **peer review (PR)**.

- ***Methodology***

The European Council invited the EC and ENSREG to develop the scope and detailed rules for the stress tests with the support of the **Western European Nuclear Regulators' Association (WENRA).** WENRA drafted the preliminary stress test specifications. A consensus was reached on these specifications, known as the **EU-STs specifications**, by ENSREG and the EC on 24 May 2011[2].

The specifications for the peer review of these EU-STs as well as a working paper on the transparency aspects of the STs[3] were agreed later at the ENSREG meeting of 11 October 2011.

The EU-STs specifications, which were the basis of the safety track of the stress tests, **set out three main areas (topics) to be assessed**: **extreme natural events** (earthquake, flooding, extreme weather conditions), **response of the plants to prolonged loss of electric power and/or loss of the ultimate heat sink** (irrespective of the initiating cause) and **severe accident management**.

The assessments were organised in three phases:

- Self-assessments by nuclear licensees. Licensees were asked to submit ST reports covering all their nuclear power plants (NPP) to the national regulators.
- National review of the self-assessments. The National regulator reviewed the ST reports supplied by the licensees and prepared a **national report (NR**);
- European peer review of national reports.

The peer review teams were composed of nuclear safety experts from EU Member States, Switzerland, Ukraine and from the Commission, with observers from three countries (Croatia, USA and Japan) and the International Atomic Energy Agency (IAEA).

A considerable effort was made to mobilise the necessary human resources to analyse the safety of all NPPs and spent fuel storage facilities of all 17 countries in a short time. In each of the 17 countries the review team conducted a NPP visit. A total of 43 reactor units were visited on the relevant sites during the originally-scheduled visits of March 2012 (approximately 30% of all units in operation). These plant visits confirmed the findings of the prior analyses and in some cases led to additional recommendations.

Additional visits were carried out at eight reactor sites by the peer review teams in September 2012, with the aim of gaining additional insights into the different reactor types, discussing how the identified improvements would be implemented, and alleviating concerns about installations in border areas. This means that all operating reactor types in Europe have been visited by peer reviewers.

While the stress tests confirmed the high standards of nuclear safety in the EU, the reports also identified a number of improvements that could enhance safety. To ensure an appropriate follow-up, Member States developed national actions plans (NAcPs) to address the recommendations.

[2] http://www.ensreg.eu/node/289/

[3] http://www.ensreg.eu/node/349/

- ***Transparency and public involvement***

During its meeting of 24-25 March 2011, the European Council mandated that the outcome of the stress tests and information on any subsequent selected safety improvement measures be made available to the public. Full transparency was therefore a key requirement for the EU-STs and its follow-up activities right from the start. The possibility for the public to become involved, by raising questions on the NRs and later the NAcPs, and to have access to all reports of the reviews conducted, illustrates the extent to which transparency has been achieved.

Several public meetings also took place in 2012 to present the process for conducting the stress tests and the major outcomes.

All NRs and NAcPs, as well as many licensee reports, are accessible to the public on the ENSREG website[4].

- ***Invitation to neighbouring countries to take part in the EU-STs***

The events in Japan underlined the vital importance of nuclear safety, which should be addressed by the European Union (EU) and its neighbouring countries together as an absolute policy priority, and the need to continuously re-evaluate nuclear safety.

A meeting was held on 23 June 2011 between Commissioner Oettinger and Deputy Ministers and senior representatives of the Ministries of Energy and national authorities responsible for nuclear energy of the Republic of Armenia, the Republic of Belarus, the Republic of Croatia, the Russian Federation, the Swiss Confederation, the Republic of Türkiye and Ukraine. Its aim was to invite these counties to take part in the EU stress tests and improve the safety of their nuclear installations. The outcome of this meeting was that the attending countries, in cooperation with the EU[5]:

- Confirmed their willingness to undertake on a voluntary basis comprehensive risk and safety assessments ('stress tests'), taking into account the specifications agreed by the European Commission and the European Nuclear Safety Regulators Group (ENSREG) on 24 May 2011. The need for a consistent approach on nuclear safety by all countries making use of nuclear energy was reinforced by their shared vision which highlights the potential cross-border nature of nuclear accidents.

- Agreed to commit nuclear operators to carry out self-assessments of their nuclear power plants, as well as to invite national regulatory bodies to present national reports, and to make use of a transparent peer review system, enhancing the credibility and accountability of the comprehensive risk and safety assessments.

- Agreed to engage on a multilateral level and with the IAEA in discussions on strong and common safety standards and international peer reviews.

Two countries - **Switzerland and Ukraine** - fully participated in the stress test process along with the EU Member States in 2012, and contributed to the national action plan (NAcP) peer reviews in 2013 and 2015.

Some neighbouring countries such as Armenia, Belarus and **Türkiye** have expressed their interest in participating in the same peer review process at an appropriate point in the future. The EC has always indicated its willingness to support, together with ENSREG, the peer review process of any country that indicates its readiness to be take part. The peer review process took place in Armenia in 2015–2016, in Belarus in 2017–2021 and **in Türkiye in 2022–2024** (as covered by this document).

[4] http://www.ensreg.eu/EU-Stress-Tests

[5] https://ec.europa.eu/energy/sites/ener/files/documents/20110623_stress_test_joint_declaration_eu_neighbouring_countries.pdf

- ***Follow-up***

Member States developed national actions plans (NAcPs) that then underwent the EU level peer review process. The first NAcP peer review workshop was organised by ENSREG in April 2013. The workshop:

- identified specific country actions and timescales for actions to improve nuclear safety in nuclear reactors;
- highlighted the importance of the 'defence in depth' principle whereby the safety of nuclear plants is assured in the case of an accident by a number of independent layers of safety actions;
- recognised the importance of periodic safety reviews (PSR) for continuous improvement in the field of nuclear safety;
- highlighted the need to maintain 'containment integrity' under severe accident conditions;
- committed to presenting an updated NAcP report by December 2014 and participating in a follow-up peer review workshop in April 2015.

The second NAcP peer review workshop was held in April 2015 to discuss the updated NAcPs, measures taken to improve the safety of nuclear power plants, and schedule changes since the first reports. Special attention was paid to the technical reasons for changing any of the safety improvement measures proposed, as well as to reviewing the studies and analyses carried out since the first workshop.

The workshop identified that a high proportion of actions listed in the NAcPs had been completed under the oversight of the respective national regulatory authorities. It concluded that most of the countries were making adequate progress on their NAcPs, with all participating countries strongly committed to fully implementing their respective improvement measures under the oversight of the regulatory authorities. Despite these positive improvements, **in November 2015 ENSREG issued a statement on this topic**[6] where it considered that 'the rate of safety upgrade implementation should be strengthened to target agreed implementation deadlines, taking into account other safety priorities and quality requirements'.

As a follow-up to the completion of the pending actions in the NAcPs, **ENSREG members committed to update and publish periodically (every 2 years starting from 2017) a status report from each country** on the implementation of its NAcP until the time of its completion. These updated NAcPs were published on the ENSREG website in January 2018[7].

[6] http://www.ensreg.eu/document/ensreg-statement-progress-implementation-post-fukushima-national-action-plans-nacps

[7] http://www.ensreg.eu/EU-Stress-Tests/Country-Specific-Reports

3 TÜRKIYE – CURRENT STATUS and STRESS TEST PROCESS

- ***Nuclear power plant in Akkuyu, Türkiye***

Türkiye started negotiations with the Russian Federation to build a nuclear power plant (NPP) at the Akkuyu site in Türkiye in February 2010. The negotiations concluded with an Intergovernmental Agreement based on a 'build-own-operate' model. The Agreement was signed on 12 May 2010. Based on the Agreement, the Akkuyu Nuclear Power Plant Electricity Generation Joint Stock Company (Akkuyu Project Company, APC), responsible for the construction and operation of 4 Water Water Energetic Reactor (Vodo- Vodyanoi Energetichesky Reaktor, VVER), was established. Each unit is expected to produce 1200 MW of power. The predecessor of the Nuclear Regulatory Authority of the Republic of Türkiye (NDK), Turkish Atomic Energy Authority (TAEK), recognised APC as the owner (referred to below as 'the Applicant') on February 7, 2011.

The Akkuyu site on the Mediterranean coast was granted a site license for building an NPP in 1976. In 2011, this site was allocated to APC as specified in the Agreement. On 9 February 2017, TAEK approved the site parameters, and in March 2017, APC applied for a construction licence of Akkuyu NPP unit 1.

The design of the Akkuyu NPP – VVER 1200 / V-509 – is the result of a development process of the USSR-designed VVER- (Vodo-Vodyanoi Energetichesky Reaktor-) type pressurised water reactor (PWR) family. The operating experience in the VVER-type plants is equivalent to around 1 300 reactor years, many of which took place in the VVER-440 power plants in Russia and eastern Europe as well as in the VVER-1000s operating in Czechia, Bulgaria, China, India, Russia and Ukraine.

A construction licence was issued in 2018 for unit 1, and works on the unit started in April 2018. A construction licence for unit 2 was issued in 2019, and works started in April 2020. A construction licence for unit 3 was issued in November 2020, with works on the unit starting in 2021. In 2021, the construction licence for unit 4 was issued, with works starting the following year. In November 2023, a commissioning permit for non-nuclear activities of unit 1 was granted.

Türkiye initially announced officially that its goal was to put unit 1 into service in 2023, unit 2 in 2024, unit 3 in 2025 and unit 4 in 2026. The project is currently delayed by around 2 years. In November 2023, Akkuyu NPP unit 1 was granted a commissioning permit by the Turkish Nuclear Regulatory Board. The first batch of fresh nuclear fuel was delivered to the Akkuyu NPP site in April 2023.

- ***Mandate to perform a stress test peer review in Türkiye***

In the wake of the Fukushima accident in 2011, Europe took the lead in carrying out comprehensive risk and safety assessments (stress tests) of nuclear power plants (NPPs) to assess their ability to withstand extreme external events.

By joining the Joint Declaration on comprehensive risk and safety assessments of nuclear plants (stress tests) in June 2011, Türkiye confirmed its willingness to carry out such assessments on a voluntary basis, in line with the specifications agreed by the European Commission and the European Nuclear Safety Regulators Group (ENSREG) on 24 May 2011.

The European Commission and ENSREG have always expressed their willingness to support any non-EU country that decides to carry out the same kind of peer review process, and this support has been extended to Türkiye.

- ***Stress tests in Türkiye in compliance with the European stress test process***

Since 2011, the European Commission's Directorate-General for Energy has been in regular contact with the Turkish Nuclear Safety Regulator to explain the EU stress tests peer review process and ensure that it could be conducted in Türkiye.

In May 2012 the Turkish Nuclear Regulatory Authority submitted Türkiye's national report to ENSREG for the stress test process. The report covered the 4 x 1200 VVER MWe Akkuyu project. In its 21st Plenary Meeting of November 2012, ENSREG decided to follow up on Türkiye's national report. Then, during ENSREG's 23rd Plenary Meeting of March 2013, it was decided that an appropriate mechanism as well as a peer review team would be set up.

Based on a decision from the 27th ENSREG Plenary Meeting of May 2014, Türkiye became an observer to ENSREG.

In 2019, Türkiye submitted an updated national stress test report, dated December 2018.

To ensure the peer review process ran smoothly, the Commission initiated early detailed discussions with ENSREG to guarantee that resources would be available to perform the peer review process in a timely manner. A tentative common understanding on the stress test peer review process based on a two-phase approach was reached with Türkiye in 2019, and subsequently approved by the ENSREG Plenary in November 2019.

ENSREG included Türkiye's peer review exercise in its work programme for 2021-2023[8] and its Board for stress tests in third countries initiated the peer review at its 8th meeting in February 2021.

Following discussions with Türkiye, an agreement was reached on a two-phase peer review. In the first phase, an ENSREG team would conduct a partial review of the relevant issues. The second phase of the peer review would take place when the design was sufficiently mature. The reason behind this approach was that construction was at an early stage, and workload needed to be balanced out due to a reorganisation of regulatory structures in Türkiye and a spike in licensing work.

To ensure consistency, the EU stress test peer review process in Türkiye was conducted in line with the technical specifications prepared by ENSREG in May 2011 for previous applications of the process. It adhered to the 'openness and transparency' principle adopted by ENSREG in December 2011, as demonstrated by the publication of Türkiye's national stress test report -the core element of the peer review, on the ENSREG website[9].

To ensure the smooth implementation of the process, all practical details of the peer review were compiled into a single document on 'practical arrangements', which was agreed and approved by the stress test Board and by Türkiye's Nuclear Regulatory Authority.

The objective of the stress test peer review is to promote continuous nuclear safety improvements in Türkiye, by providing an international, independent and complementary assessment to ensure that no important issues have been overlooked on any of the topics within the scope of the stress test. As with the conclusions of previous stress test reviews, the peer review team highlighted areas for further improvement and flagged good practice identified in earlier peer reviews of national reports (from 2012 onwards)[10] to the Turkish national regulator and the licensee/operator for their consideration.

- ***Peer Review Board***

The Peer Review Board corresponds to the ENSREG Board for stress tests in third countries. The Board composition was as follows:

Chairperson:

- Andreas Molin, Austria – until 16 September 2021
- Frank Hardeman, Belgium – from 17 September 2021 until 20 March 2024

[8] https://www.ensreg.eu/document/ensreg-work-programme-2021-23

[9] https://www.ensreg.eu/Türkiye-0

[10] http://www.ensreg.eu/NODE/513

- Petteri Tiippana, Finland– from 21 March 2024

Members:

- Tsanko Bachiyski, Bulgaria – until 20 March 2024
- Sylvie Cadet Mercier, France
- Massimo Garribba, Directorate-General for Energy, European Commission
- Patrick Majerus, Luxembourg – from 16 September 2021 until 15 August 2024
- Cristina Les (Spain) – from 25 October 2024
- Dimitris Mitrakos – from 25 October 2024
- Ms Eszter Rétfalvi – from 25 October 2024

The Board's secretariat tasks were carried out by the Euratom Policy Coordination Unit of the Directorate-General for Energy, European Commission.

The detailed roles and appointment process of the Board and the peer review team are set out in the 'Peer Review Practical Arrangements' that have been approved by the Turkish Nuclear Regulatory Authority.

- ***Peer review team***

In July 2021, ENSREG's Chairperson asked ENSREG members and observers to nominate experts to be part of a peer review team (PRT) for Türkiye. In addition to ENSREG members, nominations were also sought from countries that had already participated to the EU stress test process in the past (e.g. Switzerland and Ukraine).

Based on the proposals received and drawing from a pool of experts that had participated in previous stress test peer reviews, the Board for stress tests in Third Countries selected the team. It comprised one team leader (Petteri Tiippana of Finland), one rapporteur, a leader for Topic 1 (Extreme external initiating events), a leader for Topic 2 (Safety functions and design issues) and a leader for Topic 3 (Severe accident management), as well as experts for each of the topics. The initial PRT was composed of 13 experts nominated by EU and non-EU countries (two by Austria, one by Finland, two by France, one by Greece, one by Hungary, one by Germany, two by Switzerland and one by Ukraine). The PRT also included two European Commission officials. However, some changes were made to the PRT during the peer review exercise: a French and a Swiss member stepped down, and three members (nominated by Finland, Germany and Slovakia) were appointed to replace them.

The PRT also included a rapporteur from the Commission appointed by the ENSREG Board for stress tests in third countries and three observers: one from Cyprus, one from the IAEA and one from Iran. The IAEA changed their nomination during the exercise.

- ***Independence***

The peer review was carried out in an independent and structured manner by the selected experts in the PRT.

The experts drew on information sources provided by various stakeholders, such as the regulator, the licensee, TSOs, non-governmental organisations (NGOs), etc., in addition to the core element of the peer review, namely Türkiye's national stress test report.

- ***Türkiye peer review process – timeline***

The main activities and timeframe of the peer review exercise include:

- '**Practical arrangements'** agreed with Türkiye on 6 August 2021

- Call for **PRT expert nominations** issued on 20 September 2021
- **PRT leader appointed** by the ENSREG Board for stress tests in third countries on 16 September 2021
- A **public consultation on Türkiye's national stress test report** published on the ENSREG website from Monday 21 October 2021 to 19 November 2021
- **PRT meetings** organised virtually between 15 October 2021 and 24 May 2024
- **Document review visit to NDK in Ankara, Türkiye** on 16-20 May 2022, comprising a document review by the PRT and a series of meetings with NDK and the operator
- Summary of **preliminary findings** presented to NDK on 20 May 2022
- **List of preliminary findings** prepared by the PRT and **sent to NDK on 27 July 2022**
- **Written replies from NDK** to the PRT questions provided **on 25 January 2023**
- **Draft peer review report presented to NDK on 16 May 2024 (Topics 1 and 3) and 20 May 2024 (Topic 2)**
- **Akkuyu NPP site visit on 26-31 May 2024, including a visit by the PRT to the Akkuyu NPP, plant walkdowns and a series of meetings with NDK and the operator**
- Final summary of findings presented to NDK on 31 May 2024
- Final draft peer review report sent to NDK on 13 June 2024 for comment
- NRK's comments on the final draft peer review report sent to PRT on 25 July 2024
- Validation by NDK (with final comments) of the text of the peer review report on 24 October 2024.

- ***Transparency and public involvement***

The PRT was conscious that full transparency, combined with the opportunity for wider civil society involvement, would significantly contribute to the Türkiye ST process being recognised by the public and other stakeholders as a reliable and trustworthy reference on the status and adequacy of nuclear safety in Türkiye. Consequently, the EC and ENSREG, in close collaboration with Türkiye counterparts ensured that the PRT of the Türkiye STs was guided from the beginning by the principles of openness and transparency, similar to those applied in Europe for the earlier STs and associated follow-up process.

The Turkish Nuclear Regulatory Authority was informed about the EU transparency objectives and requirements and advised on how it might engage the public by organising a structured and comprehensive information and public communications process.

The goal of these activities was to inform all stakeholders as objectively and comprehensively as possible on each aspect of the process and to help gather the views of stakeholders on the key nuclear safety-related issues and how they were being dealt with in the course of the PR.

The Türkiye national stress test report, the core element of the peer review, was published on the ENSREG website for public consultation from 21 October 2021 to 19 November 2021. During this public consultation comments/questions were received from one source, Mr Apostolos Panagiotou, Professor Emeritus of Nuclear Physics, University of Athens, Greece.

The questions and comments were deemed to be outside the scope of the ST exercise but were nevertheless taken into consideration by the PRT.

- ***Peer review team report***

The **main outcome of this peer review exercise** is this **PRT report**. The structure of this report is similar to that of the reports published for the countries that participated in the EU-STs in 2012. According to the 2012 ST template the report covers the following topics:

- General quality of national report and national assessment
- Plant assessment relative to earthquake, flooding and other extreme weather conditions
- Plant assessment relative to loss of electrical power and loss of ultimate heat sink
- Plant assessment relative to severe accident management.

The PRT report outlines potential improvements and good practices identified during its review exercise in Türkiye, with the aim of continuously improving safety.

4 GENERAL QUALITY OF NATIONAL REPORT AND NATIONAL ASSESSMENTS

- ***Compliance of the national report with the topics defined in the EU stress tests specifications***

The PRT considers that Türkiye's national report was drafted in line with the requirements for EU stress tests. The Team notes that completing the EU stress test process takes a significant amount of effort, particularly for a country developing a new nuclear power programme.

However, there were some constraints in making all information available to the PRT, particularly during the desktop review phase. This was mainly due to the Akkuyu NPP being under construction, but also for information security reasons. However, the Turkish counterparts answered all questions and provided answers during the peer review process to the extent possible. For information security reasons, some of the information was only presented to the Team during the visit to Ankara in May 2022 and during the site visit to the Akkuyu NPP in May 2024. However, the PRT was later able to clarify outstanding points during discussions with counterparts, at the workshop in Ankara and during the site visit to the Akkuyu NPP.

- ***Adequacy of the information supplied, consistency with the EU stress tests specifications***

Topic 1

For earthquakes, external flooding and extreme weather, the national report (NR) provided basic information on the regulatory bases, technical background, methodology used for screening and characterisation of the hazards. This information was supported by numerous documents provided during the desktop review. The volume and content of these documents were extremely valuable for the peer review. Additional information and explanations were provided during two site visits and video conferences on specific topics.

Topic 2

For loss of safety functions and ultimate heat sink, the NR provided sufficient information about the regulatory bases, technical background and methodology used for assessing station blackout (SBO) and loss of ultimate heat sink (LUHS) events. However, at times the NR did not provide sufficient information on the relevant technical specifications and how they were applied to the design of the Akkuyu NPP. Gaining a clear understanding of this issue required further questions that were later clarified by NDK, and additional technical discussions prior to and during the PRT’s visits to Ankara and the Akkuyu NPP.

Topic 3

For severe accident management (SAM), the information provided in the NR is in line with the EU stress test specifications, although more detailed information was subsequently requested, including during the site visits to Ankara and to the Akkuyu NPP. Areas warranting additional information included the specific legislation covering SAM, the selected approach, the stage of development and plans for future implementation of emergency operating procedures (EOPs) and SAMGs, independence between design provisions implemented at different levels of defence, SAM strategies, operational characteristics, practical elimination and the functioning of novel design solutions.

- ***Adequacy of the assessment of compliance of the plants with their current licensing/safety case basis for the events within the scope of the stress tests***

Topic 1

The plant's compliance with their licensing basis for seismic hazards has been found to be adequate, apart from a small number of systems, structures, and components (SSCs) which do not meet the seismic design basis requirements. The plant with the licensing basis for external flooding and extreme weather has been found to be adequate.

Topic 2

For events within the scope of the stress tests, the NR describes the extent to which plants comply with their licensing/safety case bases. Some detailed information on the applicable regulations not presented in the NR were later delivered during the PRT process.

Information on the application of the defence in depth (DiD) concept and approach is set out in a specific document. International standards must be adhered to when setting a DiD concept, such as the IAEA SF, SSR 2/1, 75 INSAG 10 and 75 INSAG 12, the WENRA report 'Safety of new NPP designs' and requirements as formulated in the Russian standard NP-001-97 OPB-88/97 'General regulations on ensuring safety of nuclear power plant' as well as Turkish requirements. The DiD concept aims to address the safety aspects arising from the lessons learned exercise following the Fukushima Daichii accident and needs to be reviewed by NDK.

Topic 3

The plant’s compliance with the licensing basis for severe accidents has been found to be generally adequate. However, Turkish side mostly uses Russian rules and regulations, some of which are somewhat obsolete. NDK is working on developing a national legal basis.

- ***Adequacy of the assessment of the robustness of the plants: situations taken into account to evaluate margins***

  Topic 1

For seismic hazards, NDK requires that seismic category 1 equipment be able to withstand earthquakes with ground motion parameters 40% higher than the seismic design basis. The margin must ensure that the SSCs for the prevention or mitigation of the consequences of severe fuel damage caused by seismic events exceeding the DBE are duly qualified to carry out their functions. The reason for setting the seismic demand level exactly 40% higher than the DBE is not clear to the PRT. The 40% margin may or may not be sufficient to achieve the WENRA safety objectives for a new NPP, i.e. to render large or early releases extremely unlikely with a high degree of confidence (see Appendix 1, Table of Findings, finding nr. T1-00). For the 40% seismic margins, it has been found that the margins have not been reached for a significant number of SSCs important to safety according to current information.

For flooding hazards, it seems that events exceeding the severity of the design basis events were not taken into account and safety margins have not been specified. This is particularly the case for tsunami flooding. The situation is similar for extreme weather cases.

Topic 2

The PRT found that the assessment work carried out on loss of off-site power (supplement power transmission line), on-site power supply, station blackout or heat removal is in general acceptable. However, later in this report the PRT highlights the need for further assessment work in a number of areas it considers necessary to enhance the robustness of the design.

Topic 3

For severe accident management the majority of the scenarios were considered. However, the design relies heavily on a few passive features, common to most scenarios, and some active features, dedicated to other DiD levels. This approach has given rise to some doubts on containment cooling and depressurization strategies, with a possible cliff edge effect. As mentioned in the PRT's recommendations, these strategies should be carefully analysed and, if appropriate, the relevant measures should be developed and implemented.

- ***Regulatory treatment applied to the actions and conclusions presented in the national report***

4..1 General aspects

Topic 1

The national report (NR) lists the following safety improvements proposed by NDK to increase the plant's seismic robustness:

- Additional mobile diesel generator set (ADGS) including organisational and engineering arrangements.
- Development of operating procedures to transfer the NPP into a safe state after a seismic event.
- Improving the robustness of the plant against secondary earthquake effects (integrity of bank protection structures) by developing a procedure for regular inspection.

For flooding, the NR does not consider any measures to be necessary. For extreme weather conditions, the NR envisages that the condition of critical buildings be monitored after extreme events by a dedicated company and qualified inspectors. Surveys should be carried out according to an approved programme using visual inspection and instrumentation methods. No additional design provisions are considered to be needed to improve the stability of the Akkuyu NPP against extreme weather conditions.

Topic 2

As stated in the NR, NDK formulated the following findings to improve the reliability of the safety of the Akkuyu NPP:

- Consider the possibility of supplementing the design with supporting alternative equipment (mobile pumps, DG) with standbys, which can be connected to either of the two safety trains and to the equipment of normal operation systems important for safety. Or, as an alternative, consider the possibility of setting up an additional intermediate substation supplying power in a crisis situation to critical equipment of the power unit subject to voltage in the off-site power grid (both 10 kV or 0.4 kV switchgears and directly consumers).

- Make organisational and engineering provisions for connection (delivery, deployment) of planned alternative power supply equipment taking into account possible damage of the site distribution network access infrastructure.
- Pay attention to seismic resistance of the planned alternative power supply equipment; it shall have greater seismic resistance than regular power supply systems.
- Develop relevant operating procedures to maintain availability of the equipment required to transfer the reactor plant into safe state after an earthquake.
- Develop accident management (EOP, BDBAMG, SAMG), emergency preparedness and response (personnel protection plan, public protection plan) documentation.
- Develop a procedure for regular inspections of bank protection structures, breakwater dike, water intake facility, tunnels for essential service pipelines to improve resistance of the power plant against secondary effects of an earthquake (namely the integrity of bank protection structures).
- Pay attention to the development of the infrastructure to improve emergency response (alternative routes of materials / personnel delivery during accidents, evacuation routes).

Topic 3

NDK made significant progress during last 2 years in the field of regulatory review capabilities and dedication to safety. The NR identified a number of further measures, some of which are already being implemented. For example, an analytical justification for SAMGs is being developed and subjected to regulatory review. ADGs, for long-term blackout of the site (more than 72h), are dedicated to each unit, like the SAM equipment set in general.

4..2 Periodic safety review (PSR)

Periodic safety reviews (PSRs) are a required by Turkish legislation. In accordance with Article 19 of the Regulation on Authorisation of Nuclear Facilities (published in March 2023), periodic safety review should be carried out by the licensee after receiving the operation license at most every 10 years.

5 PLANT ASSESSMENT RELATIVE TO EARTHQUAKES, FLOODING AND OTHER EXTREME WEATHER CONDITIONS

Different from the past stress test peer reviews in the EU, Taiwan and Armenia, the plant under review is a new nuclear power plant (NPP). This fact has important implications for the review process, as the safety expectations for new NPPs are higher than those for existing plants. The latest ENSREG stress tests for the new Belarusian NPP in Astravets accounted for this fact by considering Europe's and the Western European Nuclear Regulators' Association's (WENRA) safety expectations for new reactors. The same benchmarks are applied in the stress tests on the Akkuyu NPP.

WENRA (2013[11]) stipulates that for new NPPs '*accidents with core melt which would lead to early or large releases have to be practically eliminated'.* The Vienna Declaration on Nuclear Safety formulates the same objective for new NPPs, although it does not refer to the notion of practical elimination. This Declaration was adopted by Türkiye and Russia. In this context, the possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if they can be considered with a high degree of confidence to be extremely unlikely to arise (WENRA 2019[12]). As regards the practical elimination of early or large releases, WENRA further specifies: '*For that reason, rare and severe external hazards, which may be additional to the general design basis, unless screened out (…), need to be taken into account in the overall safety analysis.'* It also states that '*Rare and severe external hazards are additional to the general design basis, and represent more challenging or less frequent events. This is a similar situation to that between Design Basis Conditions (DBC) and Design Extension Conditions (DEC); they need to be considered in the design but the analysis could be realistic rather than conservative.'*

These safety expectations require a broader and more extensive consideration of external hazards in plant design and the consideration of events with occurrence probabilities well below 10^{-4} per year in the safety demonstration.

- ***Description of current situation of plants in the country with respect to earthquakes***

5..1 Design basis earthquake (DBE)

5..1.1 Regulatory basis for safety assessment and regulatory oversight

National requirements and regulations are summarised in Chapter 1.2 of the national report (NR). Regulations are based on the Law on Turkish Atomic Energy Authority, which originally established the Turkish Atomic Energy Authority as regulator. In 2018[13], responsibilities were transferred to the Nuclear Regulatory Authority (NDK). Up to 2018, the Decree on Licensing of Nuclear Installations and the Directive on Principles of Licensing of Nuclear Power Plants formed the legal basis for regulatory action. Issues insufficiently addressed by Turkish regulations are covered by requiring compliance with IAEA Safety Fundamentals and Safety Requirements.

Requirements for the design basis: The Turkish Regulation[14] defines two levels of design basis: S1 (operating basis earthquake) and S2 (safe shutdown earthquake) as follows:

'S1: Maximum earthquake ground motion level which reasonably can be expected to be experienced at the site area once during the operating life of the plant and carried on normal operation,

[11] WENRA (2013), Report on safety of new NPP designs.
[12] WENRA (2019), Report Practical Elimination Applied to New NPP Designs – Key Elements and Expectations.
[13] Decree Laws No 702, 703; Presidential Decree No 4 on the organisation of authorities and institutions linked and related to Ministries (Articles 785-792).
[14] Citation in the NR: Regulation on Nuclear Power Plant Sites, 2009.

S2: Earthquake ground motion level that corresponds directly safety limits and maximum earthquake potential that can affect the site.'

S1 is to be determined as a minimum half of S2. The minimum acceptable level for S2 is 0.15 g.

The seismic design of the Akkuyu NPP complies with the Russian standards NP-031-01[15], MP 1.5.2.05.999.0027-2011[16] and MR 1.5.2.05.999.0025-2011[17] (NR, Chapters 2.1.1, 2.1.3). The documents are part of the licensing basis. The NR states compliance of the Russian regulations with the national requirements and IAEA Safety Standards, including seismic safety standards. NP-031-01 was developed taking into account the superseded guidelines IAEA 50-SG-D15[18] and 50-SG-S1 (Rev. 1)[19]. It defines two levels of earthquake:

1. Operating basis earthquake (OBE) (called 'design earthquake' in NP-031-01): 1 000-year return period, 5% exceedance probability in 50 years;
2. Safe shutdown earthquake (SSE) (called 'maximum credible earthquake' in NP-031-01): 10 000-year return period, 0.5% exceedance probability in 50 years.

These definitions comply with WENRA requirements for the design basis earthquake (SSE), international practice and IAEA guidance.

The peer review team (PRT) assumes that the return periods for S1 and S2 (1 000 and 10 000 years respectively) are only stipulated in the Akkuyu licensing basis and not specified in the Turkish regulations.

Safety expectations for fuel damage frequency and large release frequency: During the Phase 1 country visit in 2022, NDK reported on numerical values accepted by the regulator for fuel damage frequency (FDF), considering fuel damage in the reactor core, the spent fuel pool and the fresh fuel storage, and the large release frequency (LRF). Values were stated with FDF < 10^{-5} and LRF < 10^{-7} per year. The FDF < 10^{-5} per year is an acceptance criterion stipulated in the licensing basis and in the licence conditions. LRF < 10^{-7} per year was to be understood as a target value specified in the licence conditions similar to Russian regulations. NDK further explained that Russian regulations do not identify 'early' releases.

The acceptance criterion of FDF < 10^{-5} per year is in line with the safety expectations for new NPPs in European countries[20]. Restricting the safety expectations for radiological releases to large releases (LRF) is different to WENRA countries and the Vienna Declaration on Nuclear Safety that refer to large or early releases (LERF). Values accepted for LERF of new NPPs in WENRA countries range from 10^{-6} to 10^{-7} per year.

The PRT recommends that NDK formulate acceptance criteria (rather than '*targets'*) for the occurrence probability of large releases (LRF) and also include safety expectations for early releases (LERF).

5..1.2 Derivation of design basis earthquake

Design parameters for the design basis earthquake (called SSE in the NR and S2 in the Turkish regulations) were determined by probabilistic and deterministic approaches. Structures, systems and components (SSCs) of the Akkuyu NPP are designed based on the following peak ground accelerations for the SSE (NR, p. 48):

DBE (SSE) peak horizontal ground acceleration **PGA_{SSE-h} = 0.388 g** (10 000-year return period);

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[15] NP-031-01: Standards for Design of Seismic Resistant Nuclear Power Plant. Russian Federation, 2002.
[16] MP 1.5.2.05.999.0027-2011: Seismic Design Standards for Nuclear Power Plants. Guidelines. Russian Federation, 2011.
[17] MR 1.5.2.05.999.0025-2011: Seismic Analysis and Design of Nuclear Power Plants. Guidelines. Russian Federation, 2011. (outdated).
[18] IAEA (1992), Seismic Design and Qualification for Nuclear Power Plants. (superseded).
[19] IAEA (1991), Earthquakes and Associated Topics in Relation to Nuclear Power Plant Siting. (superseded).
[20] Nine WENRA countries require FDF < 10^{-5}, three countries require FDF values between 10^{-5} and 10^{-6}.

DBE (SSE) peak vertical ground acceleration **PGA_{SSE-v} = 0.295 g** (10 000-year return period).

Seismic category 1 equipment is also tested for earthquakes with ground motion parameters 40% higher than the SSE. The 1.4×SSE seismic analysis is made using realistic approaches.

Ground motion of the OBE was – fixed to 50% of the SSE, which is equal to PGA_{OBE-h} = 0.194 g and PGA_{OBE-v} = 0.147 g.

During the peer review, NDK explained that the DBE is based on the Seismic Hazard Assessment Report by ENVY/KOERI (2013)[21]. The report was used by the designer as the basis for their preliminary design and the resulting values for the SSE were adopted as the DBE. Ground motion values for exceedance probabilities < 10^{-4}/year were apparently not calculated.

According to the NR and to information obtained from NDK, three additional probabilistic seismic hazard assessment (PSHA) studies were performed. The studies, however, did not change the seismic design basis parameters:

(1) A PSHA by WorleyParsons (2014)[22], based on revised seismic source characterisations and seismic hazard assessment (SHA) methodology, was not considered for reasons and justifications provided in AKU-BDD0132, Rev. B04, 20.01.2017.

(2) A study[23] on the input to the seismic PSA provides the hazard curve to the PSA study 2021 (see Chapter 5.1.2.2).

(3) The Site Parameters Report (AKU-BDD0132 Rev. B04 20.01.2017, p. 6.6-1 to 6.6-176) includes a detailed description of a PSHA (referred to as PSHA 2017) completed after the definition of the DBE. The study is based on the PSHA by ENVY/KOERI (2013) but with minor modifications to seismic source zones, different weights in the logic tree and the use of different software. Ground motion values were calculated for exceedance probabilities down to 10^{-5}/year. The PSHA revealed the following ground motion parameters (p. 6-6.62):

Peak horizontal ground acceleration **$PGA_{10\text{-}4\text{-}h}$ = 0.359 g** (occurrence probability 10^{-4}/year);
Peak horizontal ground acceleration **$PGA_{10\text{-}5\text{-}h}$ = 0.662 g** (occurrence probability 10^{-5}/year).

(4) For a first version of the seismic PSA for power unit 1, a new set of hazard curves for occurrence probabilities extending down to 10^{-8}/year were calculated (PSA 2021). These hazard curves seem less demanding than those used in PSHA 2017 (see Appendix 1, Table of Findings, finding no. T1-08). It was explained that the approach was less conservative than the one used in PSA 2017.

5..1.3 Main requirements applied to this specific area

The NR regards the seismic hazard assessment to be in line with IAEA SSG-9 (2010; superseded by IAEA SSG-9 Rev. 1, 2022).

The main requirements applied to the seismic design of the Akkuyu NPP follow the Russian standards NP-031-01[Error! Bookmark not defined.], MP 1.5.2.05.999.0027-2011[Error! Bookmark not defined.] and MR 1.5.2.05.999.0025-2011[Error! Bookmark not defined.] (NR, Chapters 2.1.1 and 2.1.3). According to the NR, these regulations are in line with IAEA Safety Standards, including seismic safety standards. NP-031-01 was developed taking into account the superseded guidelines IAEA 50-SG-D15 and IAEA 50-SG-S1 (Rev. 1).

21 ENVY/KOERI (2013), Seismic Hazard Assessment of the Akkuyu NPP Site (Rev.03), March 2013.
22 WorleyParsons (2014), Standalone Seismic Hazard Assessment Report. TNPP-00-SV-REP-EN-0122-R1. 29 April 2014.
23 WorleyParsons (2014), Input to Seismic PSA Sigma Truncation. Report TNPP-00-SV-REP-EN-0128-R1.
02 September 2014.

5..1.4 Technical background for requirements, safety assessment and regulatory oversight (deterministic approach, PSA, operational experience feedback)

(1) The DBE is based on the Seismic Hazard Assessment Report by ENVY/KOERI (2013)[24]. It combines PSHA and Deterministic Seismic Hazard Analysis (DSHA) studies by the Kandilli Observatory and Earthquake Research Institute (KOERI), under contract to ENVY Inc., Paul C. Rizzo Associates, Inc. (PCR) and WorleyParsons Nuclear Services JSC (WP). The study is said to comply with IAEA SSG-9. It uses a logic tree approach with four source zone models and eight ground motion prediction equations (GMPEs) for active shallow regions and subduction zones (ENVY/ KOERI 2013; WP 2012[25], PCR 2012[26]). ENVY/KOERI 2013 was reviewed by the Institute of Physics of the Earth. Hazard calculation used EZ-FRISK™ software (Risk Inc.). The PSHA study, based on input data and models (seismic zones, source parameters, GMPEs, soil conditions etc.) prepared by four different groups and internationally reviewed, is regarded as good practice.

With respect to the combined use of PSHA and DSHA for seismic hazard assessment, the NR states that the final results of the three independent PSHA and DSHA studies differed from each other by 10% or less.

(2) The technical background of PSHA 2017 is described in detail in the Site Parameters Report AKU-BDD0132 Rev. B04 20.01.2017, further explained during the site visit. The study is not used for the definition of the DBE. Information is provided on source zone models, faults considered, associated M_{max}, GR parameter, GMPEs, the design of logic trees, logic tree weights etc. PSHA considers: (a) Five source zone models with the NPP located in a background seismicity zone. It is assumed that no seismicity can originate from the area within a radius of 5 km around the site. (b) The logic tree accounts for different completeness periods of the earthquake catalogues. The minimum magnitude is M_{min} = 3.5. (c) Three plus five different GMPEs for subduction zone ground motion models (Cyprus Subduction Zone) and shallow crustal seismicity, respectively. (d) Three models for site conditions described by different values of Vs30 (approx. 960 – 1300 m/s). Hazard deaggregation shows that the highest hazard contribution derives from the background zone encompassing the site. Seismic sources at distances of 0-50 km with magnitudes of M < 6.5 are by far the most important contributors.

The consolidated PSHA study based on input data and models prepared by four different groups (seismic zones, source parameters, GMPEs, soil conditions etc.) is regarded as a good practice. The logic tree appears robust and hazard curves incorporate epistemic uncertainty in the input data. The methodology used to calculate seismic hazard values is a strong safety feature of the Akkuyu NPP.

With respect to the DBE, NDK stated that the DBE spectra of ENVY/KOERI (2013) and PSHA 2017 are almost identical, with the ENVY/KOERI (2013) spectra being slightly more conservative. NDK further explained that the main differences between the PSHA of ENVY/KOERI (2013) and PSHA 2017 are minor modifications to seismic source zones, different weights assigned to the five source models in the logic tree, and the use of different software in the PSHA. NDK regarded these differences inconsequential, as indicated by the comparison of the DBE spectra.

PRT assessment. The high hazard contribution from near-site sources shown by hazard deaggregation calls for a careful assessment of potential seismogenic sources at distances up to 25 km or even 50 km from the site. The most important currently known potential active faults within this distance are the SW segment of the Ecemiş (Namrun) fault and the offshore Kozan (Anamur-Silifike) fault. The hazard contribution of the Ecemiş (Namrun) fault is treated differently in the source models but appears to be considered by all except by the Rizzo 1 model, which has a relatively high total weight of about 0.3 in the source zone logic tree. The Kozan (Anamur-Silifike) fault is not distinguished as a separate structure. It is contained in the background zone in all source zone models.

[24] ENVY/ KOERI (2013), Seismic Hazard Assessment of the Akkuyu NPP Site (Rev.03), March 2013.
[25] WP (2012), Independent Review of the Seismic Hazard Assessment of Akkuyu NPP Site. Interim PSHA Report, WorleyParsons Nuclear Services JSC, 23 April 2012.
[26] PCR (2012), Interim Probabilistic Seismic Hazard Analysis Report: Akkuyu Nuclear Power Plant – Draft, prepared by Paul C. Rizzo Associates, Inc., 7 May 2012.

To adequately account for the high hazard contribution from near-site sources, the PRT recommends extending systematic geomorphological, geophysical and, if appropriate, paleoseismological investigations to faults at distances of up to 50 km from the site to exclude their activity or assess their seismogenic potential. Particular attention should be paid to the location and possible activity of the Kozan (Anamur-Silifike) fault, including in the offshore area.

5..1.5 Provisions to protect the plant against DBE

According to the NR, the safety system design is based on a single failure criterion and principles of redundancy, diversity, independence and physical separation. The PRT took note that, according to the Preliminary Safety Analysis Report[27], the Akkuyu NPP is resistant to seismic impacts of the DBE (SSE) level of PGA_{SSE-h} = 0.388 g.

Protection against the DBE is provided by the classification and qualification of structures, systems and components (SSCs) in accordance with NP-031-01[Error! Bookmark not defined.] and designing SSCs to resist DBE loads. Seismic category I SSCs of the Akkuyu NPP are designed for DBE loads that assume damping and effective stiffness compatible with those defined in ASCE 4-98[28] and ASCE 43-05[29]. The PRT has no information on codes or documents for the seismic design of SSCs. NDK explained that the seismic loads of the SSCs were determined using detailed 3D FE models that consider SSI excited by three-component accelerograms defined on the free surface.

SSCs classified as seismic category II are required to resist the OBE (PGA_{OBE-h} = 0.194 g). According to the Russian code NP-031-01, Seismic category II is assigned to SSCs whose failure can lead to an interruption to electric power generation. Protection of safety-related SSCs against the secondary effects of earthquakes up to DBE is generally achieved by physical separation measures such as layout solutions, aseismic joints, and spacing of different seismic category systems. As regards remaining possible interaction between SSCs, complementary approaches are necessary either to check the robustness of higher seismic category SSCs under additional loads caused by the impact or failure of lower category SSCs, or to upgrade the design of lower category SSCs considering external loads to be covered in the design of adjoining higher category SSCs.
Protection against earthquake-induced fire is provided by firefighting systems that are qualified according to the seismic safety class of the rooms they are installed in. The fire brigade building is classified as seismic category 3. Regardless of this classification, it is designed to withstand PGA_{SSE} = 0.388 g. The PRT notes that severe accident management accounts for the fire brigade also in DEC conditions. It is therefore recommended to upgrade the fire brigade building to at least 1.4×PGASSE-h = 0.543 g. The accessibility of roads after an earthquake should not hinder the response of the fire brigade and other emergency responders.

5..1.6 Conclusions on adequacy of the design basis

Defining the design seismic basis for a return period of 10 000 years is in agreement with current international practices, IAEA guidelines and the WENRA 2021 Safety Reference Levels.

Selecting the DBE ground motion parameters on the basis of a PSHA performed by three independent groups and subjecting the results to external review is regarded as good practice.

The PRT notes that the reliability of hazard results heavily depends on the correct assessment of potential seismic sources in the near region (0-25 km) and region around the site, calling for a careful reassessment of potential seismogenic sources at distances up to 50 km from the site. Particular attention should be paid to the location and possible activity of the Kozan (Anamur-Silifike) fault. The results of these reassessments may necessitate modifying the PSHA calculations and may lead to

[27] Akkuyu NPP Unit 1, Preliminary Safety Analysis Report, Akkuyu Nuclear JSC, 2017.
[28] ASCE 4-98, Seismic Analysis of Safety-Related Nuclear Structures and Commentary, American Society of Civil Engineers, 1998.
[29] ASCE 43-05, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities American Society of Civil Engineers, 2005.

updated ground motion parameters relevant for the seismic design basis and safety margins beyond the DBE.

5..1.7 Compliance of plant(s) with current requirements for the design basis

For a new NPP, the design procedure, the design and qualification (testing) standards used form the basis for assessing compliance with design basis requirements. The plant is designed for a maximum horizontal ground acceleration of $PGA_{SSE\text{-}h}$ = 0.388 g. According to the NR and the Preliminary Safety Analysis Report**Error! Bookmark not defined.**, the plant design fulfils the design basis requirements.

The NR and documents provided by NDK during the desk review provide comprehensive information on the seismic resistance of seismic category I and II SSCs expressed by their High confidence low probability of failure (HCLPF) values. It was explained that the HCLPF values were obtained in the PSA 2017 study using the separation of variables (SoV) method and a conservative approach. All values specified for SSCs of seismic category I exceed the design value of the DBE ground motion of $PGA_{SSE\text{-}h}$ = 0.388 g, except for the pipeline tunnels for emergency power supply system cables and essential service water systems 11UQZ and 12UQZ, for which a HCLPF value of 0.38 g is listed. The smallest margins are listed for tunnels and for the fastening of cable runs with 0.41 g. These HCLPF values correspond to earthquake recurrence intervals only slightly higher than 10 000 years. Remarkably, the peak horizontal ground acceleration for the occurrence probability of 10^{-5}/year is $PGA_{10\text{-}5\text{-}h}$ = 0.662 g according to PSHA 2017.

Based on the available documentation, the PRT concludes that requirements for the design basis are not fulfilled by the emergency power supply systems (11UQZ, 12UQZ). This is regarded as a serious deficiency as these systems should be designed to be functional in severe accident scenarios, including scenarios initiated by earthquakes exceeding the DBE.

5..2 Assessment of robustness of plants beyond the design basis

The PRT notes that the WENRA Statement on Safety Objectives for New NPPs[30] calls for an extension of the safety demonstration for new plants, which is inconsistent with the distinction between ‘design’ and ‘beyond design’. The design philosophy for new NPPs has evolved and there is an expectation to address in the initial design what was ‘beyond design’ for existing reactors. WENRA (2013)[31] states that situations that are considered as ‘beyond design’ for existing plants, such as postulated multiple failure events and core melt accidents, called Design Extension Conditions in IAEA SSR-2/1, should be considered in the design of new plants. Design provisions must therefore be available to prevent accidents from escalating into core melt conditions (defence in depth/DiD Level 3) and to control accidents with core melt so that off-site releases are limited (DiD Level 4).

The adequacy of safety margins is assessed against this background.

5..2.1 Approach used for safety margins assessment

Expectations on the size of seismic margins

The reason for setting the seismic demand level exactly 40%[32] higher than the DBE is not clear from the NR. In response to the PRT, NDK stated that the seismic demand level at exactly 40% safety margin was established by the licensee as a requirement of its design contractors in accordance with the European Utility Requirements in consideration of a beyond design basis earthquake (BDBE). The PRT

[30] WENRA (2010), Statement on Safety Objectives for New Nuclear Power Plants.
[31] WENRA (2013), Report on safety of new NPP designs.
[32] The PRT notes that the general seismic margin of 40% is lower than the margins used in other countries for new NPPs, e.g., France, 1.5; USA, 1.67.

concludes that NDK did not formulate its own expectations to ensure adequate safety margins. The value of $1.4 \times PGA_{SSE\text{-}h} = 0.543$ g is fixed in the licensing basis.

In this context, it is appropriate to quote paragraph 3.27 of the IAEA Specific Safety Guide SSG-67[33] , which states that in addition to the earthquake levels SL-1 and SL-2, defined and determined for design purposes, a more severe earthquake level – derived from the hazard evaluation of the site – should be considered. For this earthquake level, referred to as the 'beyond design basis earthquake', the following applies: (a) the design should provide adequate seismic margins for those SSCs ultimately required to prevent core damage and an early radioactive release or large radioactive release; (b) the design should provide adequate seismic margins to the safety classified SSCs credited for mitigatory measures for Level 4 of the defence in depth concept; (c) it should be demonstrated that cliff edge effects are avoided within the uncertainty associated with the definition of SL-2. Furthermore, a footnote states that for low to moderate seismicity where the seismic margin is used to assess the robustness of the design, some countries define a factor of 1.4, 1.5 or 1.67. The PRT notes that the licensee has chosen the lowest factor (1.4) from the factors listed for nuclear facilities located in regions of low to moderate seismicity.

The safety margins must ensure that the SSCs for the prevention or mitigation of the consequences of severe fuel damage caused by seismic events exceeding the DBE are duly qualified to perform their relevant safety functions. Due to the significant uncertainties associated with a BDBE, the 40% margin may or may not be sufficient to meet this requirement and achieve the safety objectives for a new NPP, i.e. to make large or early releases extremely unlikely with a high degree of confidence. The actual margin required to achieve these objectives should be determined by: (1) the general safety expectation expressed by a core damage frequency or fuel damage frequency, and a large or early release frequency (LERF) defined in the regulatory framework; (2) the hazard progression beyond the DBE defined by hazard curves extending to sufficiently low occurrence probabilities (e.g., 10^{-7} per year); (3) the safety function of the SSC in question, i.e. the assessment of whether or not a margin is sufficient, must take into account the DiD level (e.g. Level 3 or 4) at which a specific SSC is required to prevent or mitigate accident scenarios. For example, according to IAEA TECDOC-1791[34], the practice in countries with low-to-medium seismicity considers an increase of about 50 % above DBE levels for the evaluation of the BDBE.

Comprehensive lists of SSCs required to fulfil the fundamental safety functions in order to prevent core melt and early or large releases were not available for the review. During the site visit, the PRT was informed that such lists existed and were under review by NDK. The list is in keeping with the list of seismic category 1 SSCs (related to seismic category I).

On fuel damage frequency (FDF) and large release frequency (LRF), NDK provided the following safety expectations: the numerical values accepted by NDK are FDF < 10^{-5} (considering fuel damage in the reactor core, the spent fuel pool and the fresh fuel storage) and LRF < 10^{-7} per year. The FDF < 10^{-5} per year value is an acceptance criterion specified in the licensing basis and licence conditions. LRF < 10^{-7} per year was to be understood as a target value. NDK further explained that 'early' releases are not considered in the regulations.

In response to the PRT, NDK stated that since the determination of the early and large release is not specified in the NDK regulatory documents, the final assessment of their probability is not possible. The PRT notes that the lack of regulatory basis for LERF is inconsistent with WENRA, which requires that LERF be practically eliminated. The PRT also noted that the required margin of $1.4 \times PGA_{SSE} = 0.543$ g is significantly smaller than the peak horizontal ground acceleration of earthquakes with a 10^{-5} annual probability of occurrence, which is $PGA_{10\text{-}5\text{-}h} = 0.662$ g according to PSHA 2017.

[33] IAEA SSG-67 (2021), 'Seismic Design for Nuclear Installations', Vienna.

[34] IAEA, 'Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants', IAEA-TECDOC-1791, 2016.

The PRT therefore recommends that NDK/the licensee verify that the chosen 40% margin is sufficient to eliminate cliff edge effects and make early or large releases extremely unlikely with a high degree of confidence (i.e. practically eliminate early or large releases) in compliance with the IAEA Specific Safety Guide SSG-67.

Methods used for determining seismic margins

The HCLPF values obtained from the fragility assessment by the separation of variables (SoV) method were evaluated for a reference earthquake RE = 1.4×PGA_{SSE} = 0.543 g.

NDK's written replies to the PRT and verbal information from Russian and NDK experts explained the applied methodology. In particular, the PRT's findings and recommendations are as follows:

- On the evaluation of the seismic resistance of SSCs, Russian practice/experience was used to define energy absorption factors Fμ in the design and fragility assessment. In the fragility assessment, Fμ was calculated in accordance with the SoV fragility assessment method based on median limits for nonlinear distortion. The evaluation of the energy absorption factor Fμ is critical for the fragility assessment and the derived safety margins.

  On the derivation of the factors Fμ and the related reduction factor Ke[35], NDK stated: '*The seismic design analyses of buildings and structures of the Akkuyu NPP to SSE-SL2 (1.4SSE) and DBE (SL1) level were performed in accordance with NP-031-01, taking into account the reduction factors of 0.625 for buildings of the seismic category I and 0.5 – for the buildings of seismic category II, taking into account the development of inelastic deformations. Thus, this safety factor is assumed to be equal to one and is not considered further in the analysis of seismic damage to buildings and structures*.'

  From the statement above, it is clear that in the SSE design, the licensee used a Ke value of 0.625 (Fμ = 1.6) for buildings of seismic resistance category I, and Ke = 0.50 (i.e. Fμ = 2.0) for buildings of seismic resistance category II. In the seismic fragility assessment, the licensee then used a value of Ke = 1.00 for buildings of both seismic resistance categories.

  The PRT notes that the Russian practice for evaluating and applying the energy absorption factor Fμ in the design basis domain (SSE) is inconsistent with EU and international practice. IAEA[36], ASCE 43-05 and ASCE 43-19 require that structures in seismic category 1 be designed to exhibit linear behaviour, i.e. Ke = 1.0. The obsolete version of the IAEA guide[37] states that structures in seismic categories 1 and 3 may be designed to exhibit nonlinear behaviour provided that their acceptance criteria are met with a safety margin consistent with the seismic categorisation. WENRA (2014)[38] requires that protection against design basis events apply reasonable conservatism in providing safety margins in the design.

  The PRT recommends that NDK review the energy absorption factors derived by the licensee and used in the basic design (SSE) because of their decisive contribution to the seismic fragility of safety-relevant SSCs.

- On floor-response spectra, the application of the energy absorption factor is only applied in the design of the structures but not the development of the in-structure and floor-response spectra. According to NDK, the floor-response spectra used in the design of electric equipment conservatively envelops the structural nodes at all floor elevations throughout the building height. The floor-response spectra used in the design of other equipment envelop the structural nodes of each respective floor. The design floor-response spectra were determined

[35] Coefficient Ke, defined in the Russian standard NP-031-01, is the reciprocal of the energy absorption factor Fμ.

[36] IAEA 2021, SSG 67, Seismic Design for Nuclear Installations.

[37] IAEA 2003 NS-G-1.6, Seismic Design and Qualification for Nuclear Installations.

[38] WENRA Safety Reference Levels for Existing Reactors, Issue T, Reference Level T5.3 (b).

without consideration for ground motion incoherence. The PRT appreciates this approach as it creates additional seismic margins.

The design floor-response spectra in the vertical direction do not take into account the effective out-of-plane stiffness reduction of the reinforced concrete slabs due to cracking. According to NDK, the Russian standard NP-031-01 does not include provisions for this effect.

Similarly, the effective out-of-plane stiffness reduction of the reinforced concrete walls due to cracking should be considered, if applicable, in the horizontal design response spectra for component design and their anchorages.

The PRT recommends that NDK/the licensee verify that a realistic out-of-plane stiffness reduction of the reinforced concrete slabs and walls due to cracking would not adversely affect the floor-response spectra used in the design and fragility assessment. This verification should prevent the unsafe design of components and/or their anchorage.

- As regards the anchorage of SSCs, in the meetings with Russian and NDK experts in 2022, the PRT pointed out that the governing failure mode of a component is frequently its anchorage. The assumption of an anchorage with a rigid baseplate as described in EN 1992-4: 2018[39] is not conservative for slender baseplate geometries (often the case) and can lead to an unsafe assessment of the equipment anchorage.

  The PRT recommends that NDK/the licensee verify that the design and fragility assessment of anchorages with slender baseplate geometries are not affected by an unrealistic rigid baseplate assumption leading to unsafe design.

- The PRT noted a lack of reference for the seismic qualification procedure by testing equipment/components. PRT recommends that NDK/the licensee ensure that the Preliminary Safety Analysis Report includes a reference to the standard or procedure-defining seismic qualification by testing for design and beyond design seismic demand that is going to be used for both the DBE and beyond the DBE to ensure enough margins and avoid cliff edge effects beyond the DBE.

5..2.2 Main results on safety margins and cliff edge effects

Seismic margins

The following assessment compares the HCLPF values of SSCs important to safety with the margin $1.4\times PGA_{SSE\text{-}h} = 0.543$ g for which, according to NDK, it should be ensured that the NPP can be transferred into safe state and the release of radioactive substances into the environment is prevented. The value $1.4\times PGA_{SSE\text{-}h} = 0.543$ g is fixed in the licensing basis. As explained above, this 40% margin may or may not be sufficient to achieve the WENRA safety objectives for a new NPP[40].

Table 3 of the NR and a report provided during the desktop review[41] list the HCLPF capacities of seismic category I SSCs. The NR concludes that seismic loads from earthquakes up to $1.4\times PGA_{SSE\text{-}h} = 0.543$ g do not exceed the robustness threshold for the SSCs and that safe shutdown of the NPP is ensured. Radioactive releases out of the containment as a result of a $1.4\times PGA_{SSE\text{-}h}$ ground motion are considered impossible.

Regardless of this statement, the PRT identified a number of seismic category I SSCs with HCLPF values smaller than the required value of $1.4x PGA_{SSE\text{-}h} = 0.543$ g[42]:

[39] CEN, TC 250 – Structural Eurocodes, Eurocode 2, ‘Design of concrete structures – Part 4: Design of fastenings for use in concrete’, EN 1992-4:2018, September 2018.

[40] The objective of practical elimination of early or large releases.

[41] Probabilistic Safety Assessment Volume 1 Level 2 Probabilistic Safety Assessment Book 16 Fuel damage frequency assessment for seismic events: AKU-VAB0101-BAA0016.

[42] Numbers in bold highlight HCLPF < 0.543 g.

11UQZ and 12UQZ (tunnels for emergency power supply system cables and service water systems): **0.38 g**
12BMF (switchgear): **0.50 g**
11QKB (cold supply): **0.49 g** (pump)
11SAC (11UBN EDG building): **0.46 g** (ventilation unit, fan)
11SAE (10UAZ 400 kV power output system bus duct tunnel): **0.51 g** (ventilation unit)
11SAD (ventilation system 12UBNEDG building): **0.46 g** (fans)
11XKA10 (generator set): **0.49 g**
SSCs of the safety systems' power supply (see below)

Seismic margins for safety systems required in DiD Levels 3 and 4 are stated as follows:

Containment and containment integrity:

10UJA (inner containment): 0.55 g (loss of robustness of certain bearing structures)
10UJA (inner containment): 0.68 g (loss of containment tightness due to cracking)
10UJA (outer containment): not analysed; designer expects higher stability than for inner containment
JMT and JMU (passive hydrogen recombiners in the containment): 0.75 g
JMN system (sprinkler protecting inner containment from overpressure): 1.25 g

Emergency shutdown:

Control rods: 2.5 g. The PRT considered this value implausibly high. During the country visit, the designer explained that the value of 2.5 g applies to the reactor internals and the HCLPF of the control rods was below the DBE ground motion of PGA_{SSE-h} = 0.388 g. However, shutdown during a DBE was said to be ensured within the required time (4 seconds) after a trigger value (and initiation of an automatic reactor scram) is detected at OBE level (0.194 g). The designer explained that shaking table experiments with SSE time histories showed that all control rods could be inserted in less than 3 seconds after initiation of the automatic scram and before the arrival of the strong ground motion phase. The PRT questions the assumption that the time window between scram initiation and the arrival of the strong ground motion phase is long enough for all expected earthquakes to ensure full rod insertion. This particularly applies to near-field earthquakes with magnitudes between 6 and 7 with short duration. Also, the triggering value of 0.194 g may already be part of the strong ground motion phase.

12JND (emergency boron injection system): 0.66 g

Failure of safety systems' power supply:

Fastening of cable runs: **0.41 g**
11UQZ, 12UQZ (tunnels for emergency power supply system cables and service water systems): **0.38 g**
Fastenings of electrical cabinets etc.: **0.52 g**
Reinforced concrete structures of EDG building: 0.72 g
11BTA Battery fastening to racks: **0.42 g**

Heat removal from core (emergency core cooling):

11JNB (emergency cooling SG): 0.59 g (SG emergency cooldown heat exchanger)
Fastening of heat exchanger: 0.61 g
Hanger supports for pipelines and steam lines: 0.63 g
Limiting value for heat removal from the core via JNB: 0.63 g.
JNJ 3rd level hydro accumulators HA3 in annulus, including piping: 1.01 g
JNJ HA2 in the containment, including piping and check valve: 1.01 g

Core catcher:

Core catcher: not analysed

JMN (for cooling the corium in the core catcher and feeding containment sprinklers): 1.31 g; piping of JMN pump: 1.25 g; availability of cooling water from SFP: 0.57 g (see below)

Spent fuel pool (SFP)

SFP (leak tightness): 0.57 g. The value is significantly lower than the fragility stated for the JMN pump used for cooling the corium and supporting sprinklers for containment cooling; the pump is fed by water from the SFP.

During the site visit, the licensee informed the PRT that HCLPF values will be recalculated, SMA will be performed and the seismic PSA will be updated when all SSCs are in place.

The PRT notes that HCLPF values below $1.4\times PGA_{SSE\text{-}h} = 0.543$ g are not in line with the margins required according to the licensing basis.

The PRT recommends NDK not to accept the margins of SSCs important to safety with HCLPF < 0.543g and take necessary action to ensure compliance.

The PRT expresses concern about the calculated seismic fragilities, i.e. the HCLPF values calculated by the designer. It recommends that NDK arrange for an in-depth review of the HCLPF values.

The PRT questions the data reported on seismic resistance of the reactor core and the reactor shutdown system (full control rod insertion within the required time window). It recommends reviewing the fragility calculation of the reactor core (needed for both SMA and seismic PSA) and the licensee’s approach to ensuring safe shutdown of the reactor by inserting control rods during seismic events. With respect to the latter, the PRT draws attention to IAEA SSG-67[43]: ‘Seismic category 1 includes the items that need to remain functional during and/or after the occurrence of the SL-2 design basis earthquake. An item in seismic category 1 should maintain its functionality and/or structural integrity (depending on functional requirements) during and/or after the occurrence of the SL-2 design basis earthquake, and an adequate seismic margin should be provided to avoid cliff edge effects. Seismic category 1 should include the following items: (x) Items whose failure could directly or indirectly cause accident conditions; (x) Items that are necessary for shutting down a reactor and maintaining a reactor in a safe shutdown condition, including the removal of decay heat.’

Secondary effects of earthquakes more severe than the DBE

The results of the current seismic margins evaluation of SSCs (HCLPF procedure) presented in Table 3 of the NR and in documents received during the desktop review do not mention secondary effects due to possible interaction between SSCs in the event of an earthquake more severe than the DBE, i.e. ‘beyond design basis earthquake’. Depending on secondary effects and impacts (failure modes related to resistance, displacement) due to their interaction, the seismic margins of possibly interacting SSCs may be overestimated when compared to $1.4\text{x}PGA_{SSE\text{-}h} = 0.543$ g.

The PRT therefore recommends that the licensee update the seismic margins evaluation of the safety-related SSCs needed for the control of accidents caused by earthquakes exceeding the DBE and the safety analysis to take into account secondary effects of earthquakes such as additional loads (spatial interaction, pounding, failure or collapse) due to possible interaction and displacements with adjoining SSCs. Potential impacts not only between category I SSCs, but also between category I and non-category I SSCs should be considered. The potential for impact should be considered for SSE and for all earthquake levels above the design basis.

Range of earthquakes leading to severe fuel damage

The NR gives values of the mean annual fuel damage frequency (FDF) derived from a seismic PSA. Calculations discretised the mean seismic hazard curve for the site ground motion parameter PGA, used for developing the component fragilities of SSCs, into 8 intervals between 0.2 g (occurrence probability ≈ 4×10^{-4}/year) and 0.6 g (occurrence probability ≈ 5×10^{-6}/year) with steps of 0.05 g. The

[43] IAEA SSG-67 (2021), Seismic Design for Nuclear Installations, Vienna, paragraph 3.33.

total mean annual FDF of core and pool fuel is 5.59×10^{-6}; the individual FDF values calculated for the core fuel and the pool fuel are 4.64×10^{-6} and 1.84×10^{-6}, respectively. The stated values are below 10^{-5} per NPP unit per year as required by NP-001-97[44].

In general, the PRT agrees with the discretisation of the mean seismic hazard curve for the site into 8 intervals in order to evaluate the mean annual frequency of plant damage state and obtain the seismic risk quantification by convolution of the seismic hazard curve and the plant damage state fragility curve. The mean seismic hazard curve used for the seismic PSA is shown in Figure 7 of the NR. Remarkably, the shown hazard curve indicates significantly lower PGA values than the results of PSHA 2017 and 2021. In addition, the hazard curve only extends to an exceedance probability of 10^{-5}/year, which is not sufficient to evaluate and demonstrate the LRF < 10^{-7}/year[45].

The NR shows in Figure 8 which ground motion intervals contribute to FDF. The last bin (0.55 g-0.60 g) appears to be the largest contributor. However, since the plant's fragility curve is not shown in the report, it cannot be excluded that the contribution of the bin > 0.60 g to the risk is significant or even higher than the last bin (0.55 g-0.60 g). These observations suggest that because of the convolution of the plant damage state fragility curve with the truncated hazard curve, the assessed risk is incomplete, i.e. the CD, PD, and CD+PD values are incorrect (i.e. optimistic estimate).

During the site visit, the designer reported on a new seismic PSA completed in 2021. This study is based on the PSA 2021 hazard curve and extends the considered ground motion bins up to approx. 1 g. The resulting FDF was $7.4*10^{-6}$, which is approx. 30 % higher than the value stated in the NR. The newly calculated FDF corresponds to about 70-80% of the total FDF, including all risk contributors. NDK/the licensee stated that the total FDF was still below the value of 10^{-5} per year and in line with regulatory expectations.

The high contribution of seismic hazard of 70-80% to the total FDF (reactor core and fuel storage) calls for a particular focus on seismic issues.

Range of earthquakes leading to loss of containment integrity

According to the NR, the fragility assessment demonstrates that the main SSCs have sufficient margins to withstand $1.4\times PGA_{SSE\text{-}h} = 0.543$ g. According to the document Topic 1-44.pdf, '*in the framework of the fragility analysis, all structures inside the 10UJA building are considered jointly with the reactor building (inside containment)*'.

The inner containment is made of prestressed concrete with a 6 mm steel liner. The internal structures within the inner containment that support the Nuclear Steam Supply System are made of conventional reinforced concrete. The outer containment is also made of conventional reinforced concrete. It is designed to withstand aircraft crashes and other external hazards. The outer containment has no confinement function.

The HCLPF value of 0.55 g for the 10UJA building given in Table 5.6.1 of the document Topic 1-05.pdf for '*loss of strength in the building's individual bearing reinforced concrete structural components*' is most likely to be valid for a failure mode in structural members of the inner structure. This value is at the seismic demand level of $1.4\times PGA_{SSE\text{-}h} = 0.543$ g of the review earthquake and indicates a potential cliff edge effect. According to NDK, the HCLPF = 0.55 g in the PSA corresponds to a 10UJA building collapse that causes core damage and large releases. Furthermore, according to the NR, the HCLPF = 0.68 g was evaluated for the loss of leak tightness failure mode of the inner containment, i.e. only 23% above the PGA of the review earthquake and the fragility failure mode of the reactor building. NDK explained that in the PSA study this HCLPF value is used if there is no collapse of the 10UJA building

[44] NP-001-97: General Regulations on Ensuring Safety of Nuclear Power Plants, Gosatomnadzor of Russia, 14 November 1997.

[45] For PSA 2021, the hazard curves were extended to an exceedance probability of 10^{-7}/year.

(independent accident sequences). The PRT concluded that these critical margins may not be sufficient to ensure robust seismic safety of the new plant. In accordance with the post-Fukushima update of the defence in depth principle and to avoid a cliff edge effect, additional seismic margin may be required (see e.g.Gürpinar et al[46]).

The PRT recommends that NDK assess whether the safety margin of the inner structure and inner containment is sufficient to exclude a cliff edge effect with an occurrence probability in line with the safety objectives for new NPPs. The PRT further recommends that it review in detail the crucial HCLPF values of the inner structure and inner containment, i.e. whether they are conservative with regard to possible internal force redistribution and activation of additional capacity. If the safety margin is not sufficient, NDK should indicate any provisions envisaged to eliminate the cliff edge effect and increase the robustness of the plant, e.g. modifications to the hardware, procedures or organisational provisions.

5..2.3 Strong safety features and areas for safety improvement identified in the process

The PRT acknowledges the following strong safety features:

- Dedication to perform modern PSHA based on data and models by four national/international groups and an independent review.
- Passive safety systems of VVER-1200/B-509, including the passive heat removal system, the emergency core cooling system (passive part), overpressure protection and hydrogen management systems, and the core catcher. It should be ensured that these systems are available in BDBA conditions subsequent earthquake exceeding the DBE ground motion.

5..2.4 Possible measures to increase robustness

The PRT recommends that NDK/the licensee verify that the chosen 40% margin is sufficient to eliminate cliff edge effects and to make early or large releases extremely unlikely with a high degree of confidence.

The seismic margin review identified a number of seismic category I SSCs with HCLPF values below the required value of $1.4\times PGA_{SSE\text{-}h}$ = 0.543 g. This mainly concerns a number of SSCs of the 11UQZ and 12UQZ safety systems' power supply that do not even meet the design basis requirements. It is recommended that NDK require compliance of these systems to meet the required margin of $1.4\times PGA_{SSE\ h}$ = 0.543 g.

NDK should review and confirm the correctness of assumptions made about the calculation of the seismic margins that are decisive for the reliability of the calculated HCLPF values. These include: (i) definitions of the energy absorption factor Fμ in the licensee's fragility assessment; (ii) out-of-plane stiffness reduction of the reinforced concrete slabs and walls due to cracking and its effect on the floor-response spectra; (iii) effects of the rigid baseplate assumption on anchorages with slender baseplate geometries. Depending on the results of the review, HCLPF values may need to be recalculated.

NDK should review and confirm the correctness of the mean annual FDF derived from the seismic PSA. The PRT is concerned about the applicability of the seismic hazard curve used to obtain the seismic risk quantification by convolution with the plant damage state fragility curve. First, the hazard curve underlying the PSA shows ground motion values significantly lower than those predicted by the 2017 and 2021 PHSAs. Second, it appears that the risk convolution is arbitrarily truncated, omitting the effects of earthquakes with annual occurrence probabilities of less than 5*10-6 (i.e. PGA > 0.60 g). Both issues may lead to underestimated FDF values (i.e. optimistic estimate). Depending on the results of the review, FDF values may need to be recalculated and compared to the regulatory requirements.

[46] Gürpinar, A. and Johnson, J., 'Considerations for Beyond Design Base External Hazards in Advance Reactor Safety Analysis', Transactions of the SMiRT 27, Yokohama, Japan, 2024.

It is recommended to develop a clear understanding of the required functionality of SSCs important to safety at different DiD levels and to ensure independence between SSCs' safety assigned to different DiD levels. WENRA (2013) stipulates that the ability of an SSC to perform the required safety functions must not be affected by the operation or failure of other SSCs needed at other DiD levels, and the ability to perform the required safety functions must not be affected by the occurrence of the effects resulting from the initiating earthquake.

5..2.5 Measures (including further studies) already decided or implemented by operators and/or required by regulators

The PRT appraises and supports the following measures proposed by NDK:

- Additional mobile diesel generator set (ADGS) for increasing the seismic robustness of the plant, including organisational and engineering arrangements for the ADGS connection (delivery, deployment), taking into account possible damage to the infrastructure due to a seismic event exceeding the DBE.
- Development of relevant operating procedures to maintain the availability of equipment required to transfer the NPP into a safe state after a seismic event.
- Improving the robustness of the plant against the secondary effects of an earthquake (in particular, the integrity of bank protection structures) by developing a procedure for regular inspections of the coastal engineering structures.

5..3 Peer review conclusions and recommendations specific to this area

The Akkuyu NPP is designed to withstand a design basis earthquake (DBE, also called SSE) with a 10 000-year return period and a peak horizontal ground acceleration of $PGA_{SSE\text{-}h}$ = 0.388 g. The DBE is derived from a PSHA that corresponds to the state of science. Both the definition of the DBE and seismic hazard assessment are in line with national and WENRA requirements. On the seismic hazard assessment, however, the PRT notes that the high hazard contribution from near-site sources calls for a careful assessment of potential seismogenic sources at distances of up to 50 km from the site. The most important potential active fault within this distance is the possible offshore continuation of the Kozan (Anamur-Silifike) fault.

Based on the available documentation, the PRT concludes that requirements for the design basis are fulfilled by all seismic category I SSCs except for the tunnels for emergency power supply system cables and service water systems (11UQZ, 12UQZ). This is regarded as a serious deficiency as these systems should be designed to be functional in severe accident scenarios, including scenarios initiated by earthquakes that exceed the DBE.

Seismic category 1 equipment is also required to withstand earthquakes with ground motion parameters 40% higher than the SSE, i.e. $1.4 \times PGA_{SSE\text{-}h}$ = 0.543 g. This value was proposed by the licensee, while the regulator seemingly did not formulate its own expectations to ensure adequate safety margins. The margin should ensure that the SSCs for the prevention or mitigation of the consequences of severe fuel damage caused by seismic events that exceed the DBE are duly qualified to perform their relevant safety functions. The 40% margin may or may not be sufficient to achieve the WENRA safety objectives for a new NPP, i.e. to render large or early releases extremely unlikely with a high degree of confidence. Further, the peer review revealed that a number of SSCs important to safety do not meet the margins proposed by the licensee. The PRT was informed during the site visit that a detailed fragility analysis may be carried out on a number of SSCs in the SEL as part of the 2024 PSA.

On the evaluation of the seismic resistance of SSCs, Russian practice was used in the design and the fragility assessment. The evaluation of the energy absorption factor Fμ is critical in both processes. The PRT notes that the Russian practice for evaluating and applying Fμ in the design basis domain is inconsistent with EU and international practice, leading to non-conservative design.

If the earthquake leads to core damage and loss of containment integrity, the PRT identified the following most critical issues:

1. Shutdown by insertion of control rods may not be conservatively ensured for the DBE (SSE) level. Shutdown is initiated by automatic scram at a trigger value of 0.194 g (OBE) using the time window between the automatic scram initiating their insertion and the arrival of the strong ground motion phase. The PRT understands that insertion of all control rods in the strong ground motion phase may not be possible. It has doubts that full insertion of the control rods is guaranteed for all possible earthquake scenarios.
2. The HCLPF value of 0.55 g for the 10UJA building is most likely to be valid for a failure mode in structural members of the inner structure. This value is at the the $1.4 \times PG_{ASSE\text{-}h}$ = 0.543 g level and indicates a potential cliff edge effect. According to NDK, the HCLPF = 0.55 g in the PSA corresponds to a 10UJA building collapse that causes core damage and large releases. PRT recommends that the fragility of this failure mode be further investigated.

During the site visit, information was provided on the updated seismic PSA in 2021, which showed an FDF (accounting for fuel in the reactor, fresh fuel and spent fuel) of 7.4×10^{-6} per year. This was said to be about 70-80% of the total FDF, including all contributors. The high contribution of seismic hazard of 70-80% to the total FDF calls for a particular focus on seismic issues.
As regards the regulatory basis for safety assessment, the PRT recommends:

- Reviewing if the seismic margin of 1.4xSSE with PGA = 1.4×SSE = 0.54 g is sufficient to ensure compliance with the safety objectives for new NPPs, i.e. to practically eliminate early or large releases.
- NDK formulate acceptance criteria (rather than ‘targets’) for the occurrence probability of large release frequencies and also include safety expectations for large or early release frequencies following the safety objectives for new NPPs in WENRA countries (see T1-00, ‘Introductory remarks’).

As regards the design basis earthquake (DBE), the PRT recommends:

- Investigating the location and possible activity of the Kozan (Anamur-Silifike) fault, possibly extending into the near region of the site in detail to increase the reliability of the PSHA result.
- Extending geomorphological, geophysical, paleoseismological (etc.) investigations to faults at distances of up to 25 km or even 50 km from the site to exclude their activity or assess their seismogenic potential.
- Increasing the sensitivity and accuracy of the local seismic monitoring network. The PRT recommends installing two additional broadband stations on the Mediterranean shore W and E of Akkuyu, respectively.
- Systematically observing and analysing the seismicity in the near region offshore of Akkuyu. Analyses should include efforts to locate hypocentres as accurately as possible and calculate fault plane solutions for stronger events.
- Reviewing the data produced during site investigations, such as geomorphological interpretations and the interpretation of offshore seismic reflection in the light of the results of seismic monitoring to ensure that no active faults extend into the near region of the site.
- Retrofitting SSC tunnels for emergency power supply system cables and service water systems (11UQZ, 12UQZ) that do not meet the seismic design basis requirements. This is regarded as a serious deficiency as these systems should be designed to be functional in design basis

accidents (DiD Level 3) to ensure that they do not propagate to severe accidents and severe accident scenarios (DiD Level 4), including scenarios initiated by earthquakes that exceed the DBE.

- Analysing and clarifying the differences between the hazard curves included in PSHA 2017 (Site Parameters Report) and the hazard curve PSA 2021 (used for the PSA). Typically, a mean hazard curve (duly considering the uncertainties) is used in the PSA. NDK should request a detailed justification for the approach used and the reason for deriving a 'less conservative' hazard curve for the PSA.

As regards the approach used for the safety margin assessment, the PRT recommends:

- Confirming, through independent review, the correctness of the assumptions made for the calculation of seismic margins: definitions of the energy absorption ($F\mu$), out-of-plane stiffness reduction of reinforced concrete slabs and walls, effects of rigid baseplate assumption on anchorages. Depending on the results, HCLPF values may have to be recalculated. NDK should ensure that the Preliminary Safety Analysis Report includes a reference to the standard or procedure for seismic qualification by testing that it will be used for both the DBE and beyond the DBE to ensure sufficient margins and avoid cliff edge effects beyond the DBE.
- Considering, in the evaluation of the seismic margins of safety-related SSCs needed in DEC for earthquakes more severe than the DBE, the possible indirect seismic effects by the spatial interaction, structural pounding, failure or collapse of adjoining SSCs. The interaction of SSCs with margins less than 1.4xSSE with safety-relevant SSCs should be considered in the evaluation of the seismic margins of SSCs required in DEC.

As regards safety margins, the PRT recommends:

- Not accepting the margins of SSCs important to safety with HCLPF < 0.543g and taking necessary action to ensure compliance. This primarily concerns a number of SSCs of the safety systems' power supply (fastenings of electric control cabinets, inverters, rectifiers, switchgear, transformers battery, fastenings to racks and cable runs etc.).
- Reviewing the fragility calculations of the reactor core (needed for both SMA and seismic PSA) and the licensee's approach to ensuring safe shutdown of the reactor by inserting control rods during seismic events. The PRT has doubts that the approach outlined by the designer (i.e. to use the time window between initiation of scram and the arrival of the strong ground motion phase to trigger and fully insert the control rods) conservatively ensures the control of reactivity for the DBE and ensures adequate margins for earthquakes that exceed the DBE.
- Evaluating whether the safety margins of the inner structure (0.55 g) and the inner containment loss of leak tightness (0.68 g) are sufficient to exclude a cliff edge effect in line with the safety objectives for new NPPs. The PRT considers the HCLPF of 0.68 g for loss of leak tightness of the inner containment, corresponding to ground motion with an occurrence probability of about 10^{-5}/year, to be too low. If the safety margins are insufficient, NDK should require any provisions envisaged to increase the robustness of the plant.
- Ensuring that firefighting systems that protect SSCs important to safety are seismically resistant at least to the level of the SSE (compare WENRA, 2021, Reference Level SV5.6). Fire protection of SSCs with DEC functions, in particular SSCs necessary to prevent early or large releases, should also be ensured for DEC earthquakes. The fire brigade building is designed to withstand $PGA_{SSE\text{-}h}$ = 0.388 g. The PRT notes that severe accident management accounts for the fire brigade also in DEC conditions (see Chapter 7). It is therefore recommended to upgrade the fire brigade building to at least $1.4 \times PGA_{SSE\text{-}h}$ = 0.543 g and ensure accessibility to access roads after an earthquake to enable the fire brigade and other emergency responders to respond.

- ***Description of current situation of plants in the country with respect to floods***

5..1 Design basis flood (DBF)

5..1.1 Regulatory basis for safety assessment and regulatory oversight

Akkuyu NPP flood studies have been conducted on the basis of Turkish regulations, IAEA requirements and Russian Federation regulations, in particular:

- Turkish Regulation, Decree on Licensing of Nuclear Installations
- Turkish Regulation on Nuclear Power Plant Sites
- Russian Regulation NP-064-05, Consideration of external natural and man-induced impacts on nuclear facilities
- IAEA SSR-2/1 Rev1., Safety of Nuclear Power Plant Design
- IAEA NS-R-3, Site Evaluation for Nuclear Installations [Note MN: superseded by IAEA SSR-1]
- IAEA SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations

The main regulatory basis used to define the DBF is Section 6 'Hydrological External Natural Event' of the 2009 Turkish Regulation on Nuclear Power Plant Sites. Article 18 stipulates:

- *'To determine design basis floods, probabilistic or deterministic methods shall be used, or if not possible, stochastic method. Uncertainties should be considered in the analysis.*
- *Hydrological and meteorological data over a minimum of 50 years should be collected for river banks.*
- *In addition to hydrological and meteorological events, the analysis of the events that may cause flooding in the region alone such as the failure of water retaining structures like dam break, as well as the analysis of flooding that may occur as a result of with combination of multiple events should be carried out.*
- *Parameters of tsunami or seiche that can affect the plant are determined via deterministic methods and whenever possible, these results should be verified with reviewing of historical records and seiche data on coastal region at site vicinity. Conservative approach is used in case of discrepancies.*
- *The nature and breaking mechanism of the waves and for the entire range of water elevations that are expected should be identified, and the hydrostatic and hydrodynamic loading on structures important to safety should be evaluated.'*

5..1.2 Derivation of DBF

<u>Local site characteristics</u>

The Akkuyu NPP is situated in a small bay open on its south-west side to the Mediterranean Sea, 100 km north of Cyprus. The site platform has been levelled on around 1 km^2 and is directly surrounded by hills up to 270 m high. The ridgeline of these hills constitutes a water divide that prevents potential floods beyond the ridgeline from spreading to the site area. The site shoreline is protected by a system of sea walls and dikes. There are no wetlands or natural water reservoirs in the site area.

<u>Flooding sources</u>

The following flooding scenarios have been taken into consideration to determine the DBF:

- Floods from streams and rivers. The most significant surface water source in the region of the site is the Sipahili River located about 7 km to the west. Due to the hydrological separation from the site valley by the water divide line, the run-off from the Sipahili River cannot reach the site. There are temporary streams flowing only during the cold season from November to February within the site valley, when most of the precipitation occurs. Potential site flooding by these temporary streams is discussed in Chapter 5.3.

- Potential dam failures. Due to the topography surrounding the site, no dam rupture could impact the Akkuyu NPP site. The nearest dam is the Gezende Dam, which is 53.3 km away.
- Storm surge, discussed below.
- Tsunami, which is discussed in detail below.
- Extreme precipitation (discussed in the extreme weather chapter).

According to the NR, landslide tsunamis and seiches in the semi-closed Mediterranean basin have been evaluated but present no significant hazard. Water receding during a tsunami is not discussed.

Tsunami DB

Among the potential flooding scenarios, a tsunami is considered to be the most critical. Chronicles and geological investigations in the Eastern Mediterranean show that tsunamis were observed up to at least 3 000 years before our current era due to high seismicity and volcanic eruptions.

The assessment of the tsunami hazards at the Akkuyu NPP site has been made in several steps. In 2013, a paleo-tsunami study and a detailed local bathymetry survey were conducted around the site.

In 2017, deterministic and probabilistic tsunami hazard assessments (DTHA-2017 and PTHA-2017) were developed as part of the Site Parameters Report. Both assessments were based on the modelling of 357 rupture scenarios spread over eight tsunamigenic zones. They computed the expected peak coastal amplitudes, taking the original native Akkuyu Bay morphology into account. The tsunamigenic zone generating the highest runup was 'RUN-4' located along the northern coast of Cyprus.

In 2018, a new report (THA-2018[47]) updated the modelling. It used the planned new morphology of the Akkuyu Bay and computed the tsunami parameters for six points around the offshore hydraulic engineering structures. This modelling update was provided only for the RUN-4 zone. It generated significantly lower runup values than those computed in 2017.

To take into account significant design modifications decided in the water inlet and outlet channels, a new DTHA report (DTHA 2023) was prepared. It modelled tsunami scenarios at 32 points around the site for the two most significant tsunamigenic sources (RUN-4 and RUN-7). Among all 112 scenarios modelled, DTHA 2023 concluded that the maximum modelled tsunami runup and minimum sea level are +11.73 m (at location PIB-04) and -8.89 m (at location PIB-11) respectively. Nevertheless, DTHA 2023 uses an unusual statistical combination of the modelled scenarios to conclude that the probable maximum tsunami runup and minimum sea level would be +5.36 m and -4.6 m respectively.

To define the tsunami DB, this calculated tsunami wave height is added to the following simultaneous adverse conditions (probability 1/year, total = +2.08 m): sea level rise due to global warming (+1.0 m), wind wave set-up (+0.08 m), tide (+0.15 m), storm wave set-up (+0.60 m), barometric effects (+0.1 m), seasonal fluctuations (+0.15 m).

DTHA 2023 concluded that the DB maximum tsunami runup and minimum sea level would be **+7.44 m** and **-5.30 m** respectively. However, it also concludes that another statistical combination of the tsunami scenarios could be used to increase the confidence level (mean+σ), which would lead to a maximum tsunami runup of +9.86 m and a minimum sea level of -7.15 m.

Considering the limitation of the current PTHA, the DB tsunami of the Akkuyu NPP is defined exclusively using a deterministic approach.

Storm surge DB

Calculations made concluded that the level of the probable maximum storm surge corresponding to a probability of 10^{-6}/year would be +7.54 m, which is lower than the site platform elevation and lower

[47] Akkuyu Nuclear Power Plant, 'Tsunami Analysis Report for Akkuyu NPP with the Designed Structures', Revision 1.02, 28/02/2018.

than the DBF tsunami. Site flooding due to storm surge is therefore considered excluded. The delivery of personnel and necessary equipment to the Akkuyu NPP remains in place during a storm surge.

5..1.3 Technical background for requirements, safety assessment and regulatory oversight (deterministic approach, PSA, operational experience feedback)

The characterisation of tsunami hazards is a complex task that encompasses numerous steps needed to limit the uncertainties. The following steps were conducted for the Akkuyu site.

A paleo-tsunami study of the Akkuyu site was conducted in 2013. Evidence of a past tsunami was found in trench ST-22 at Arkum. It consisted of a green sandy silty clay unit with a rich and diverse benthic and pelagic foraminifera and bivalve fauna, and is a strong candidate for a tsunami deposit for the following reasons: (i) sharp basal boundary and differing lithology and fauna from the neighbouring units; (ii) fining upward grain size; (iii) mixed, diverse benthic and pelagic foraminifer fauna with the benthic fauna indicating diverse depth and bottom water conditions; and (iv) broken bivalve shells indicating high energy environments. This deposit was found 2-3 m above sea level and 1650 m from the coastline in the delta of the Göksu River. It has a calibrated radiocarbon age of AD 990-1270, which likely corresponds to the tsunami event of AD 1036-1037 reported by the Armenian chronicler Matthew of Edessa. Unfortunately, source parameters of this tsunami event do not exist in the most widely used earthquake and tsunami catalogues.

A detailed local bathymetry survey was conducted in 2013, up to approximately 3 km around the Akkuyu site. The tsunami modelling later used both this detailed local bathymetry and the rougher bathymetry available for deeper water.

In 2017, the tsunami catalogue was compiled using existing sources. Tsunamigenic zones were identified based on probable epicentres. Among these, eight tsunamigenic zones were prioritised owing to their expected significance for the Akkuyu site. An attempt was then made to model the tsunami that led to the paleo-tsunami deposits observed at Arkum.

In 2017, deterministic and a probabilistic tsunami hazard assessments (DTHA-2017 and PTHA-2017) were developed as part of the Site Parameter Report. Both assessments were based on the modelling of 357 rupture scenarios spread over eight tsunamigenic zones. In 2018, a new report (THA-2018) provided an update of the modelling. It used the planned new morphology of Akkuyu Bay and computed the tsunami parameters at six points around the offshore hydraulic engineering structures. In 2023, a new DTHA provided an update of the modelling. The approach currently followed to define the DB tsunami corresponds to neither a standard DTHA nor to a PTHA.

5..1.4 Provisions to protect the plant against DBF

The plant platform has been levelled at an elevation of +10.50 m and this is the main design provision against tsunamis and storm surges.

The site platform has been expanded in the bay. The inlet and outlet channels are bordered by dikes, which consist of permeable structures made of rock material of various sizes (mass of 1-400 kg), covered by large rocks and topped by 30-tonne antifer concrete blocks piled in two layers. These structures are permeable and are meant to reduce the energy of the waves. The stability of these structures against tsunami-induced erosion has been checked and found to be adequate. The seaward perimeter of the +10.50 m flat site platform consists of parts with sloped bank protection structures covered by antifer blocks (example: East corner), parts with sloped shores (example: NW 'harbour/bay'), parts with vertical walls (example: pumping station) and a lower platform (the +3m South quay).

The essential service water system (water intake pipe, pumps, etc.) is designed to remain operable for the current DBF tsunami value of +7.44 m. The stability against tsunami-induced hydrodynamic impacts on the dikes and offshore structures was assessed and judged to be satisfactory.

The Akkuyu NPP has a standard seaside design to prevent clogging of the pumping station by flotsam, fish, jellyfish, seaweed, etc. This design consists of a net in the water inlet channel and rotating filters in the pumping station to protect the intake of the essential service water system.

5..1.5 Conclusions on adequacy of the design basis

The plant has applied a sound and formalised methodology to screen and characterise external flooding hazards. Among other steps, the PRT sees the regional paleo-tsunami study and the detailed local bathymetry survey of the Akkuyu site as commendable practices to reduce uncertainties.

For earthquake-induced tsunamis, the international consensus is that the hazard should be assessed by using either a deterministic hazard analysis or a probabilistic hazard analysis, or preferably both methods (see for instance IAEA safety guide SSG-18). In a DTHA, the DB tsunami is defined as the maximum water level from the range of all tsunami heights from conservative numerical calculations for all the possible seismogenic sources. PTHAs are globally analogous to PSHAs, but their use is not yet widespread. Results of the PTHAs are typically displayed as the annual frequency of exceedance of runup height values obtained through a logic tree approach. The plant decided to exclusively use a deterministic method to assess the DB tsunami as the existing PTHA lacks data points and therefore accuracy and reliability. The PRT agrees on this.

The method used at the plant to define the DB tsunami corresponds neither to a standard DTHA nor to a PTHA.

To improve adequacy of the DBF definition, the PRT recommends the following:

- Remove the unrealistic WorleyParsons values from the tsunami DB assessment because of their significant impact on the key DB tsunami demand metrics, i.e. maximum and minimum sea water levels, and maximum flow velocities.
- Thoroughly review the differences between the rupture parameters used in the DTHA 2017 and DTHA 2023 studies and correct them as necessary. Any modification to the rupture parameters should be covered by a careful and sound justification.
- Revise the methodology used for defining the DB tsunami to ensure it follows established international practice.

5..1.1 Compliance of plant(s) with current requirements for design basis

The NR states that the NPP complies with the regulatory requirements on protection against flooding hazards.

The plant is built on a flat platform (+10.50 m) higher than the current DBF level (+7.44 m, to be revised in line with the PRT recommendations). The plant considers that this sufficiently protects SSCs from external flooding. However, a large number of SSCs important to safety are partly or fully located in basements below the +10.50 m level, i.e. in basements below the DB flood level.

According to international practices[48], all SSCs necessary to prevent core damage, early radioactive releases or large radioactive releases should be located at elevations higher than the design basis flood for new nuclear installations. Alternatively, adequately engineered safety features (e.g. watertight doors) should be in place to protect these SSCs. For new nuclear installations, a dry site is preferred over a site protected by permanent external barriers. If the dry site concept cannot be applied to all SSCs important to safety, the layout should include permanent flood barriers with appropriate design bases and adequate margins.

The fact that some SSCs important to safety are located below the DBF leads us to conclude that the Akkuyu NPP does not qualify as a ‘dry site’. This stresses the importance of the ‘volumetric protection’ of all safety-related buildings, including galleries leading to these buildings. Due to the status of

[48] IAEA, 2021. Design of Nuclear Installations Against External Events Excluding Earthquakes, SSG-68.

construction, the PRT was not able to witness the implementation of the volumetric protection during the site visit.

5..2 Assessment of robustness of plants beyond the design basis

5..2.1 Approach used for safety margins assessment

The approach used in the NR to assess the margins above the DBF consists of incrementally increasing the flood level until it reaches levels at which adverse effects could occur.

There is currently no acceptance criteria for the margin to be provided beyond the DB tsunami and to avoid cliff edge effects.

5..2.2 Main results on safety margins and cliff edge effects

If the water level increases above the current DBF value of +7.44 m (to be updated based on the PRT recommendations), the following is expected:

- Some safety-related systems are located in basement rooms, i.e. below the +10.50 m elevation. It is assumed that these rooms cannot be flooded when the water level remains below +10.50 m.
- At +10.50 m (i.e. above the DBF tsunami), water will start to flood the site platform. Ingress of water is expected to start in some buildings and saturate the surface sewer system.
- At +11.15 m (i.e. +3.71 m above the DBF tsunami, or +0.65m above the site platform), water would reach the bottom of entrances and openings of buildings that host safety systems. There is no significant sloping on the site. Doors and penetrations to these buildings are designed to be watertight. Nevertheless, if water were to penetrate these buildings, it is assumed that all equipment located in rooms at and below this elevation would become inoperable due to flooding, and that this equipment could not be restored within 72 hours. This would lead to a loss of heat removal from the reactor and from the spent fuel pool due to water ingress into the reactor building, adjoining structures and into the UKA building. The UKA building at - 5.40 m below platform level, houses the rooms of KAA-secured cooling water system, JNA cooling system, and other safety systems. Battery rooms of safety systems and normal operation systems are located in adjoining structures. Rooms for DC boards of safety systems are located -2.10 m below platform level. The SDGS building houses service basements at -5.00 m below platform level, with auxiliary equipment of the diesel generator and ventilation rooms.
  It is assumed that flooding to 11.15 m would lead to station blackout. In this mode, the passive heat removal system, HA-2.3 and the alternative diesel generator fulfil the function of residual heat removal from the reactor and spent fuel pool.

  This scenario is described in detail in topic 2 (loss of safety functions), and the design has provisions to ensure residual heat removal for 72 hours without fuel damage.

The Akkuyu NPP has a certain amount of mobile or semi-mobile equipment to be used in DEC situations, including mobile pumps, mobile heat exchangers and mobile diesel generators located and stored around site platform level. The plant has also several underground emergency shelters for plant staff (building entrances around +10.85 m) and several buildings used to manage emergencies[49]. The

[49] In addition to the available emergency response tools and equipment located directly onsite, there is a joint civil defence storage facility with a site for the Mobile Crisis Centre and special emergency response equipment, which is located offsite in a specially protected area.

elevation of the above-mentioned means does not provide additional protection against external flooding compared to SSCs used in lower levels of defence in depth (DiD).

Due to the location and elevation of the above-mentioned means and facilities, external flooding may challenge multiple levels of DiD at the same time. Items important to safety should be located taking into consideration common cause failure mechanisms generated by hazards[50]. Facilities, tools and equipment to be used in an emergency, and in the accident management programme, are unlikely to be affected by or made unavailable by accidents[51]. Emergency facilities should be usable under all emergency conditions[52], including flooding by tsunami.

5..2.3 Strong safety features and areas for safety improvement identified in the process

The PRT sees the following as commendable practices:

- the regional paleo-tsunami study to increase confidence in tsunami catalogues;
- the detailed local bathymetry survey around the Akkuyu site to reduce the uncertainties over tsunami modelling;
- the timely consideration of some of the PRT preliminary recommendations expressed during the 2022 PRT mission in Ankara, using DTHA 2023.

5..2.4 Possible measures to increase robustness

As mentioned in earlier chapters, to further increase the robustness of the plant against external flooding, the PRT recommends the following:

- Defining the acceptance criteria for the necessary margins to cliff edge effects beyond the DB tsunami. These acceptance criteria should be in line with international practices for external hazards in general, and for coastal flooding in particular.
- Increasing the robustness against external flooding of the means necessary during emergencies and accident management, including mobile means and emergency facilities.
- To check that plant measures against water ingress into safety-related buildings and underground galleries are robustly designed and implemented.

5..2.5 Measures (including further studies) already decided or implemented by operators and/or required by regulators

The following organisational and engineering arrangements are identified in the NR as potential further safety improvements:

- To improve stability of the plant against secondary effects of an earthquake (namely the integrity of bank protection structures), it is recommended to develop a procedure for regular inspections of bank protection structures, breakwater dikes, water intake facilities and tunnels for essential service pipelines.

5..3 Peer review conclusions and recommendations specific to this area

The plant has applied a sound and formalised methodology to screen and characterise external flooding hazards. Among other steps, the PRT sees the regional paleo-tsunami study and the detailed local bathymetry survey of the Akkuyu site as commendable practices to reduce uncertainties. The PRT also appreciates the timely consideration of some of the PRT preliminary recommendations expressed during the 2022 PRT mission in Ankara, using DTHA 2023.

[50] IAEA SSR-2/1 Rev.1: 'Safety of Nuclear Power Plants: Design', 2016.

[51] IAEA SSR-2/2 Rev.1: 'Safety of Nuclear Power Plants: Commissioning and Operation', 2016.

[52] IAEA GS-G-2.1: 'Arrangements for Preparedness for a Nuclear or Radiological Emergency', 2007.

To better assess and further increase the robustness of the plant against external flooding, the PRT makes the following recommendations in particular.

As regards the regulatory basis for safety assessment, the PRT recommends:

- Defining the acceptance criteria for the margins beyond the tsunami DB (i.e. to prevent cliff edge effects in the design). These acceptance criteria should be in line with international practices for external hazards in general, and for coastal flooding in particular.

As regards design basis floods, the PRT recommends:

- Reconsidering the adequacy of the statistical combination of the results of the four DTHA studies and confirming that the most conservative, but physically reasonable, DTHA results are used as tsunami demand in the DB. The PRT recommends removing the unrealistic WP values from the tsunami DB assessment because of their significant impact on the key DB tsunami demand metrics, i.e. maximum and minimum sea water levels, and maximum flow velocities.
- Thoroughly reviewing the differences between the rupture parameters used in the DTHA 2017 and DTHA 2023 studies and correcting them as necessary. Any modification to the rupture parameters should be covered by a careful and sound justification.
- Revising the methodology used for defining the DB tsunami to ensure it follows established international practice.
- Clarifying the inconsistency between saturation of the hazard curve at +8 m and the existence of numerous modelling results above that value, and then developing a tsunami hazard curve that better covers exceedance probabilities lower than 10^{-3}/year. Reducing uncertainties over the tsunami hazard curve for low exceedance probabilities by adding modelling points for lower exceedance frequencies. Deriving updated tsunami hazard curves once the tsunami hazard assessment is considered up to date and will integrate the other PRT recommendations.

As regards safety margins, the PRT recommends:

- Increasing the robustness against external flooding (for example, relocating to a higher elevation as appropriate) using the means necessary during emergencies and accident management, including mobile means and emergency facilities.
- Checking that plant measures against water ingress into safety-related buildings and underground galleries are designed and implemented in a robust manner.
- Once the DB tsunami has been updated in line with the PRT recommendations and once the acceptance criteria for the necessary margins for cliff edge effects beyond the tsunami DB have been defined, assess whether the existing margins are sufficient and take actions as appropriate.

- ***Description of current situation of plants in the country with respect to extreme weather***

5..1 Design basis extreme weather

5..1.1 Regulatory basis for safety assessment and regulatory oversight (national requirements, international standards, licensing basis already used by another country…)

Akkuyu NPP extreme weather studies have been conducted on the basis of Turkish regulations, Russian Federation regulations and IAEA Safety Standards, in particular:

- Turkish Regulation, Decree on Licensing of Nuclear Installations
- Turkish Regulation on Nuclear Power Plant Sites
- Russian Regulation NP-064-05, Consideration of external natural and man-made impacts on projects of nuclear power application

- Russian Regulation NP-031-01, Standards for design of seismic resistant nuclear power plant
- Russian Regulation RB-022-01, Recommendations on Evaluation of Tornado Characteristics for Nuclear Facilities, Moscow, 2001
- STO 1.1.1.03.002.0912-2012, Regulations on design of constructions sealed enclosures zone accident localization of NPP with dual containment
- SP 20.13330.2011, Set of rules. loads and effects. Revised edition. SNiP 2.01.07-85*
- Safety of buildings and structures, Technical regulations, No.384-FZ of 30.12.2009
- PiN AE-5.6 1986, Standards of NPP construction designing for different types of reactors
- IAEA SSR-2/1 Rev1., Safety of Nuclear Power Plant Design
- IAEA NS-R-3, Site Evaluation for Nuclear Installations [superseded by IAEA SSR-1]
- IAEA SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations
- IAEA Series No. 50-SGS11A, Extreme Meteorological Events in Nuclear Power Plant Siting, Excluding Tropical Cyclones, Vienna, 1983 [superseeded]
- AKU-PSAR040307-BQB0001, Preliminary Safety Analysis Report

5..1.2 Derivation of the design bases for extreme weather loads

Extreme precipitation

The 24-hour probable maximum precipitation was calculated based on daily rainfall data from nearby meteorological stations. Probabilistic depth-duration curves were established. The resulting design basis for daily maximum precipitation with recurrence once in 10 000 years is equal to 302.7 mm/day.

The DBF precipitation has been defined as 688.5 mm per 24 consecutive hours, which offers a comfortable margin beyond the precipitation corresponding to a probability of 10^{-4}/year.

Extreme winds and tornadoes

Meteorological data representative of the Akkuyu NPP area and adopted as the Akkuyu NPP design basis are defined in Section 4.7 of the SPR. In order to define design bases for strong winds and tornadoes, three types of data by source of origin were used. The first type of meteorological data is taken from ENVY Akkuyu NPP site engineering survey reports. ENVY studied the meteorological data of long-term observations obtained from Anamur and Silifke meteorological stations. These data were used to derive average and extreme meteorological parameters. 'The Summary of Average and Extreme Weather Conditions' released by the Turkish State Meteorological Service was also used. Chapter 4 of the SPR states that the Anamur meteorological station is more representative of the Akkuyu site than the Silifke meteorological station due to the topographic similarity of the Anamur meteorological station and the Akkuyu NPP. The second type of data is based on a series of atmospheric measurements of the Akkuyu NPP site. The third type of data is taken from Russian regulations (SNiP and GOST). Permanent and temporary (long-term, short-term and special) loads are taken into account for designing Akkuyu NPP buildings and structures. Loads and effects applied to buildings and structures are accepted in compliance with the Russian and Turkish requirements.

5..1.3 Technical background for requirements, safety assessment and regulatory oversight

Extreme winds and tornadoes

The Preliminary Safety Analysis Report (PSAR) recommends using design values for wind loads corresponding to the maximum wind speed recorded at the representative meteorological station (MS) – Anamur MS. The design value for high wind (5 s gusts) is 76.1 m/s for the 10^-4 annual exceedance probability. The 10-min averaged wind speed is 42 m/s for the annual exceedance probability (AEP) of 10^-4, based on the meteorological data recorded at Anamur MS between 1967 and 2015. Safety Category I buildings and structures (classification according to PiN AE-5.6) are designed to withstand extreme wind loads of 76.1 m/s, corresponding to 5 s gusts and annual exceedance probability (AEP) of 10^{-4} per year.

The intensity class of the design basis tornado (DBT) was determined in accordance with the recommendations in Russian Regulation RB-022-01 ‘Recommendations on evaluation of the tornado characteristics for the nuclear facilities’ and Specific Safety Guide No. SSG-18 ‘Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations, Vienna, IAEA, 2011’. According to the database of probable tornadoes and the PSAR, 13 tornadoes were recorded within 100 000 km^2 of the Akkuyu NPP between 1981 and 2015. Records include data from the Turkish State Meteorological Service, Directorate for Disaster Cleanup Operations, the European Severe Weather Database and Karaman & Markovsky (2014). The DBT was classified as F2 on the Fujita scale. The parameters of the DBT were determined from the catalogue by Karaman & Markovsky (2014): maximum horizontal rotational speed of 60 m/s, maximum horizontal translational speed of 15 m/s, and maximum air pressure drop of 44 hPa. Missiles are not considered in the design as F2 tornadoes only lift up light objects that are said to be not capable of damaging the NPP building. The annual probability of a DBT passage through the district where the Akkuyu NPP is located is $7.85*10^{-7}$. Category I buildings and structures (classification according to PiN AE-5.6) are designed to withstand a DBT of F2 intensity on the Fujita scale.

The organisation of prompt warnings of tornadoes is included in the ‘Action plan on protection of personnel in the event of accident at Akkuyu NPP EPR-III-BS-CT-03-01-2021’ dated 18/06/2022.

Extreme precipitation

In 2019, comprehensive meteorological, surface and marine surveys were conducted around the Akkuyu site. These surveys allowed the necessary data to be collected in order to assess the design bases for extreme precipitation in particular, and to establish probabilistic rain depth duration curves for different probabilities of exceedance.

Conclusions on adequacy of design basis

Extreme winds and tornadoes

Design values for extreme winds and tornadoes seem robust and are derived from comprehensive input data. They seem adequate with regard to relevant Turkish and Russian regulations, and take into account IAEA requirements.

Extreme precipitation

Design bases for extreme precipitation seem to be robust.

5..1.4 Compliance of plant(s) with current requirements for design basis

Plant provisions against extreme precipitation

An underground rainwater sewage system is designed to protect the site against rainwater. Water enters the system through gutter inlets located along the roadway. Rainwater from the area surrounding the Akkuyu NPP site is collected in the drainage channels 01-02-03UZN located around the fence of the site. All water collected through these channels is evacuated gravitationally to the sea. The geological characteristics around the site exclude the possibility of significant mudflow.

Plant provisions against extreme winds and tornadoes

The effects of the DBT were considered in the design in accordance with Russian Regulation RB-022-01. According to Section 3.7.2 ‘Design Load Combinations Applied to Civil Structures of PSAR’, permanent and temporary (long-term, short-term and special) loads are taken into account in the design of the Akkuyu NPP buildings and structures. Loads and effects applied to buildings and structures are accepted in compliance with the Russian, Turkish and international requirements. Safety factors are considered in the design according to the requirements of Federal Law (No.384-FZ of 30.12.2009) for all rated forces. Calculations of permanent and temporary (long-term, short-term, special) loads are performed in accordance with SP 20.13330.2011, depending on load time. Wind loads were calculated as follows: typical wind speed at the level of 10 m above ground is determined by 10-minute intervals of averaging, and exceeded once in 50 years on average for design is 32.2 m/s. Buildings and structures referred to as safety category I (according to PiN AE-5.6 1986) and seismic stability category II (according to NP-031-01) are also analysed for extreme natural and man-made impacts. The following wind and tornado impacts are considered as special loads for building construction structures: extreme wind velocity (possible once in 10 000 years) – 76.1 m/s; loads due to tornado: class of intensity with probability of travelling over the NPP site once in 10 000 years is 2.0 on the Fujita scale. Tornado parameters: maximum wall swirl velocity (maximum wind speed) – 60 m/s; travelling speed of a tornado movement – 15 m/s; pressure difference between tornado funnel periphery and its centre – 44 GPa.

5..2 Assessment of robustness of plants beyond the design basis

5..2.1 Approach used for safety margins assessment

The NPP buildings and structures are rated for resisting the following initiating events: extreme temperatures, extreme wind loading, tornadoes, extreme rains, snow loads and icing, dust storms, lightning strikes.

Extreme precipitation

Very significant margins exist above the DB. A a detailed approach was therefore not needed.

5..2.2 Main results on safety margins and cliff edge effects

The following results have been compiled based on the Level 1 PSA methodology for external initiating events: development of selection criteria and compilation of a complete generalised list of external initiating events (EIE), screening analysis based on 'qualitative' selection criteria, boundary and detailed analyses of selected EIEs, analysis of the severe core damage frequency during equipment failures caused by EIEs or a combination of them. EIEs to be considered when assessing severe core damage have been selected on the basis of extreme values (maximums) of the intensity and frequency of EIEs, as well as on the basis of qualitative selection criteria and taking into account the design basis of the NPP unit. Sources of external events specific to the Akkuyu NPP units have been identified based on a detailed analysis of information about the NPP unit site, design features of the unit, the location of unit structures and systems, and facilities located in the NPP area. No sources out of the list recommended by RB-021-01 were found among the identified ones.

Extreme air temperatures

In accordance with SPR, the extreme air temperatures possible once in 10 000 years are assumed as follows: minimum of − 12.7 C, maximum of + 50.4 C.

The Akkuyu NPP is designed so that the process equipment of safety systems and normal operation systems remains functional at the maximum indoor temperature of + 40°C. The control and monitoring of system hardware is functional at a room temperature not higher than 40°C. An extreme maximum outdoor air temperature of 50.4°C is used for the confirmatory analysis of ventilation systems in safety system rooms. The analysis confirmed that Akkuyu NPP ventilation systems maintain a temperature of not more than 40°C in safety system rooms at this extreme maximum outdoor temperature (50.4°C). As for low temperatures, the Akkuyu NPP is designed so that the equipment of safety systems and normal operation systems remains functional at the minimum indoor temperature of + 5°C. An extreme minimum outdoor air temperature of − 12.7°C is used for the confirmatory analysis of heating and ventilation systems in safety system rooms. The analysis confirmed that heating and ventilation systems maintain a temperature of not less than + 5°C in safety system rooms at this extreme minimum outdoor temperature (− 12.7°C). Due to long extremely low temperatures of outdoor air, the air temperature in normal operation system rooms may drop below + 5°C. Operator intervention is necessary to prevent this by partially disabling ventilation systems. Besides the ventilation and air conditioning system, extreme outdoor temperatures can affect the passive heat removal system. It is designed for the following extreme outdoor air temperatures: typical minimum temperature − 14.1°C, typical maximum temperature + 50.4°C. Due to long extremely high outdoor air temperatures, the air temperature may rise in normal operation system rooms. This can lead to electrical and process equipment overheating and shutting down when appropriate trips and interlocks are triggered. However, regular operation of the safety system will protect the power unit. Based on these considerations, the NR concludes that extremely high/extremely low outdoor air temperatures (possible once every 10 000 years) do not cause a cliff edge effect leading to a significant deterioration in safety.

Extreme precipitation

The capacity of the drainage system at the Akkuyu NPP is 688.5 mm per 24 hours. The daily maximum precipitation at the nearby Anamur meteorological station, corresponding to a probability of 10^{-4}/year, is estimated to be 302.7 mm per 24 hours. If well maintained and if available in all circumstances (including loss of off-site power and station blackout), the capacity of the drainage system could cope with more than twice the design basis flood precipitation. Such a high amount of daily precipitation has a probability of exceedance lower than 10^{-6}/year.

The NR also mentions that, assuming a 20-minute duration of extreme rainfall and an absolutely flat NPP site, water will accumulate in a layer of up to 9.5 mm without drainage. With drainage to the sea, only a partial flooding of the site is possible in the area of auxiliary structures that do not house safety-related equipment.

All safety-related building entrances are + 0.65 m above the plant grade elevation in order to further protect the safety-related facilities from intense precipitation at the Akkuyu NPP site.

Extreme winds and tornadoes

The extreme wind speed that can occur once every 10 000 years is assumed equal to 76.1 m/s. The tornado intensity class with the probability of this weather event within 100 000 km^2 of the Akkuyu NPP once every 10 000 years is assumed to be F2 on the Fujita scale. The intensity class of the design basis tornado was determined in accordance with the recommendations in Russian Regulation RB-022-01, Recommendations on evaluation of the characteristics of whirlwind for the nuclear facilities and according to the IEAE safety standard ‘Specific Safety Guide No. SSG-18. Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations. Vienna, IAEA, 2011’, based on the database of probable tornadoes (1981-2015) and according to the extended database of probable tornadoes (1981-2022).

The NR states that extreme wind loads do not result in the violation of safe operation limits because the buildings and structures are designed for such loads. Within the F2 tornado class, light missiles can be generated. The NR states that this would not impact Akkuyu NPP buildings or structures, according to PSAR and Russian regulations. According to the NR, wind loads also cannot affect the availability of the passive heat removal system as the system is designed for heat removal under wind loads of up to 90 m/s, which is significantly higher than the maximum wind likely to occur once in 10 000 years. Natural circulation in the passive heat removal system could be disrupted by tornado impact. However, the tornado effect is of a transient nature because the tornado moves along with the tornado generating cloud at a speed of up to tens of kilometres per hour. Natural circulation in the passive heat removal system is restored to normal as soon as the tornado leaves. However, extreme wind may cause irreversible deterioration of the passive heat removal system, according to the Table 7 resulting from external initiating events analysis made on the basis of SPR and PSAR. The NR states that the Akkuyu NPP is designed to withstand extreme wind with a sufficient safety margin, and that extreme wind and tornadoes would not compromise the reliability of heat removal or cause a cliff edge effect.

5..2.3 Strong safety features and areas for safety improvement identified in the process

The PRT sees the following as commendable practices:

- conducting comprehensive meteorological, surface and marine hydrological surveys.

5..3 Peer review conclusions and recommendations specific to this area

The analysis of hazards caused by extreme weather conditions at the Akkuyu site has been conducted on the basis of Turkish regulations, Russian Federation regulations and IAEA Safety Standards. The design values for extreme precipitation (external flooding caused by rain), extreme winds, tornadoes and extreme (high/low) temperatures were defined according to events with occurrence probabilities of 10^{-4} per year. This is in line with WENRA requirements. The intensity class of the design basis tornado was determined in accordance with IAEA guidance. The design values for extreme precipitation, extreme winds, tornadoes and extreme temperatures seem robust and are derived from comprehensive input data.

Protection of the plant against flooding by rain is provided by an underground rainwater sewage system that drains water by gravitation to the sea. The geological characteristics around the site exclude the possibility of significant mudflow. The effects of a design basis tornado of intensity class 2 on the Fujita scale were considered in the design. Protection against extremely high temperatures is provided by ventilation systems that maintain a temperature of not more than 40°C in safety system rooms. This ensures that safety systems and normal operation systems remain functional. As for low temperatures, the Akkuyu NPP is designed to ensure that safety systems and normal operation systems function at minimum indoor temperatures of 5°C. This temperature can be maintained by heating and ventilation systems. For long-lasting periods of cold, operator intervention is necessary to prevent the temperature from falling below 5°C.

Safety margins for flooding by extreme precipitation are more than twice the design basis flood precipitation. The design of the passive heat removal system for wind loads of up to 90 m/s provides a significant safety margin. A comprehensive analysis of safety margins for extreme wind appears to be unavailable. However, the PRT regards this as a minor issue as civil structures important to safety are designed to withstand earthquakes, which have similar load requirements to wind.

The PRT has no recommendations for hazards related to extreme weather conditions.

6 ASSESSING THE CAPACITY OF PLANTS TO DEAL WITH LOSS OF ELECTRICAL POWER AND LOSS OF ULTIMATE HEAT SINK

Description of the current state of plants in Türkiye

6..1 Regulatory basis for safety assessment and regulatory oversight (national requirements, international standards, licensing basis already used by another country)

The design of the Akkuyu NPP, of type AES 2006 VVER-1200, model V-509 is the result of an evolutionary development process of the Russian VVER (Vodo- Vodyanoi Energetichesky Reaktor) - type pressurised water reactor (PWR) family. According to the commonly-applied international categorisation, the concept is considered to be in the family of Generation III+ plant designs. Operating experience with the VVER-type units amounts to about 1 500 reactor years in several countries.

Version V-509 was developed in 2013, based on version 392M constructed at Novovoronezh II nuclear power plant, and adapted for the Akkuyu NPP project. However, it differs significantly from the 392M design, as the plant has been adapted to include elements of the VVER TOI standard. The pilot project of VVER TOI design is currently under construction in Russia at Kursk II nuclear power plant. Among the adaptations of Version V-509, the reactor core is equipped with 163 fuel assemblies and 94 control rods, in comparison with 121 control rods in V-392M. The fuel elements are of TWS-TOI design, as opposed to the TWS-2006 fuel elements used in the AES-2006 V392M reactor. Also, the layout of the main equipment of the primary circuit has been adapted to the VVER TOI design.

The designer of the Akkuyu NPP project is JSC 'Atomenergoproekt' (Moscow).

According to the national report (NR), the legal framework in Türkiye is composed of laws, international treaties, decree laws, regulations, guides and standards organised in the typical pyramid structure commonly applied in many countries. Based on the Directive on Determination of Licensing Basis Regulations, Guides and Standards and Reference Plant for Nuclear Power Plants issued in 2012[53], and in accordance with the Regulation on Authorisations for Nuclear Facilities issued in 2023, issues that are not sufficiently addressed by existing Turkish regulations on nuclear safety will need to comply with IAEA Safety Fundamentals and Safety Requirements. Although the Directive was revoked by the relevant articles of the Regulation on Authorisations for Nuclear Facilities, the same statements are valid in the Regulation. Regulations of the vendor or designer country must also be applied for issues that are not sufficiently addressed by the Turkish regulations. For remaining issues, third party country laws, regulations, standards, or the IAEA Safety Guides may be referenced. The revoked Directive also contained provisions for selecting and approving a reference plant that represents the NPP units to be installed[54]. In March 2023 a new Regulation on Authorisations for Nuclear Facilities was published[55], replacing the Decree on Licensing of Nuclear Installations in force since 1983.

Basic requirements for the design of an NPP are laid down in the Regulation on Design Principles for Safety of Nuclear Power Plants and the Regulation on Specific Principles for Safety of Nuclear Power Plants. Both Regulations are dated 2008, thus predating IAEA SSR-2/1 Rev. 1 of 2016. The Regulation

[53]See Türkiye’s National Report for the Joint 8th and 9th Review Meeting of the Convention on Nuclear Safety.

[54]See Türkiye’s National Report for the Joint 8th and 9th Review Meeting of the Convention on Nuclear Safety.

[55] https://www.mevzuat.gov.tr/mevzuat?MevzuatNo=40114&MevzuatTur=7&MevzuatTertip=5

on Nuclear Power Plant Sites of 2009 sets out the nuclear safety requirements for siting of nuclear power plants[56]. It incorporates IAEA Safety Requirement SSR-1.

No information is provided in the NR on whether or how the requirements of IAEA SSR 2/1, Rev. 1 and SSR 2/2, Rev. 1 are implemented in Turkish regulations and standards for the design, commissioning and operation of the Akkuyu NPP. According to the NR, for some specific topics related to protection against flooding and seismic hazards, as well as emergency management provisions, the following IAEA standards are cited:

- IAEA SSR-2/1 Rev. 1., Safety of Nuclear Power Plant Design, 2016;
- IAEA NS-R-3 Site Evaluation for Nuclear Installations[57], 2003;
- IAEA SSG-18 Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations, 2011;
- IAEA SSG-9 Seismic Hazards in Site Evaluation for Nuclear Installations[58], 2010;
- IAEA TECDOC-955 Generic assessment procedures for determining protective actions during a reactor accident, 1997.

In addition, regulations of the Russian Federation, as the vendor country, have been taken into account for the Probabilistic Safety Analyses (PSA) at the design stage, for seismic resistance and other external natural hazards and hazards caused by human activity as follows:

- NP-001-97 (OPB-88/97) General regulations on ensuring safety of nuclear power plants[59], 1997;
- NP-031-01 Standards for Design of Seismic Resistant Nuclear Power Plant, 2002;
- NP-006-098 Requirements to Contents of Safety Analysis Report of Nuclear Power Plant with VVER Reactors[60], 2003;
- NP-064-05 Consideration of external natural and man-induced impacts on nuclear facilities[61], 2005.

Türkiye participated in the IAEA missions aimed at further developing the national nuclear regulatory framework and infrastructure. The Integrated Nuclear Infrastructure Review (INIR) Mission took place on 4-14 November 2013 in Türkiye, with the participation of related organisations, including the former national regulator: the Turkish Atomic Energy Agency (TAEK). The results of the mission have not been published. However, according to the current Nuclear Regulatory Authority (NDK), a mission report with recommendations, has been prepared and submitted to the Turkish Ministry of Energy and

[56]See Türkiye’s National Report for the Joint 8th and 9th Review Meeting of the Convention on Nuclear Safety.

[57] IAEA NS-R-3 Safety Standards has been superseded by SSR-1 Site Evaluation for Nuclear Installations, 2019.

[58] IAEA SSG-9 Safety Standards has been superseded by SSG-9 (Rev. 1) Seismic Hazards in Site Evaluation for Nuclear Installations,2022.

[59] Russian Regulation NP-001-97 has been superseded by NP-001-15 General Provisions for Nuclear Power Plant Safety Assurance, 2015.

[60] Russian Regulation NP-006-098 has been superseded by NP-006-18 Requirements for the content of Safety Analysis Reports for nuclear power plant units with VVER reactors, 2017.

[61] Russian Regulation NP-064-05 has been superseded by NP-064-17 Record of external natural and human-induced impacts on nuclear facilities, 2018.

Natural Resources[62]. Another mission by the Integrated Regulatory Review Service (IRRS) took place on 04-16 May 2022 in Ankara, as requested by Türkiye, hosted by the Nuclear Regulatory Authority (NDK) in order to review the legal and regulatory framework for nuclear and radiation safety. The mission report identified areas for improvement in the national regulatory infrastructure, some of which were taken up in the national action plan for improving national nuclear regulation[63].

6..2 Main requirements for this specific area

Nuclear power plant (NPP) safety is achieved by fully applying the defence in depth (DiD) principle based on the application of a system of measures composed by off-site and on-site power systems, physical barriers and different levels of ultimate heat sink.

The NR sets out basic criteria for the composition and function of the respective safety systems. They are structured in accordance with recognised international practices and comprise single failure criterion, redundancy, diversity, independence and physical separation.

However, the NR does not include detailed requirements such as codes and standards, norms and practices of other countries, or specific requirements of the systems designer or licence holder.

6..3 Technical background for requirements, safety assessment and regulatory oversight (Deterministic approach, PSA, Operational Experience Feedback)

According to the NR, Probabilistic Safety Analysis (PSA) is applied in and is integral to the safety assessment conducted as part of the licensing procedure for the Akkuyu NPP. PSA levels 1 and 2 have been developed at the design stage to assess Akkuyu NPP safety. The analyses were developed in accordance with Russian regulations, taking into account IAEA recommendations. The total fuel damage frequency (FDF)[64] in the core for all groups of internal initiating events for all operating modes of the power unit and the 18-month fuel cycle is reported in the NR to be $2.79 \cdot 10^{-7}$ per year. The total FDF calculated in the PSA Level 1, for all internal and external events, is reported as 2.58×10^{-6} per year for the reactor core and 9.76×10^{-7} per year for the spent fuel pool. Moreover, no other specific values of the FDF for the current state of PSA Level 1 and the results of PSA Level 2 are presented in the NR. The two latter values of FDF mentioned above were provided during discussions between the PRT and Russian experts during the on-site mission to the Akkuyu NPP in May 2024.

In IAEA safety standard SSG 3, reference is made to the theoretical basis of achievable core damage frequency (CDF) for new reactor concepts considering the longstanding experience with the design and operation of nuclear power plants. This document also refers to IAEA document INSAG 12, where as an objective to be achieved for new plants, a maximum CDF of 1×10^{-5} per reactor year has been proposed for a full scope PSA Level 1 (all operational modes, all potential initiating events and potential hazards). The new Russian regulation NP-001-15 establishes as targets that the total severe accident probability for each NPP power unit (as well as for fuel storages outside the power unit), should not exceed 10^{-5} per year, and the total probability of a large-scale emergency release for each NPP power unit should not exceed 10^{-7} per year.

Regarding Deterministic Safety Analysis (DSA), the NR does not include a description of the general application of DSA. However, the NR does include the results of the DSAs of the NPP for the stress tests

[62] https://www.ndk.gov.tr/uaea-misyonlar

[63] https://www.iaea.org/newscenter/pressreleases/iaea-mission-finds-commitment-to-safety-in-turkiye-as-country-builds-first-nuclear-power-plant

[64] In the NR, the term ‘fuel damage frequency’ is used.

requirements. Since DSA is an internationally-applied standard procedure for analysing the safety of nuclear facilities, the application of DSA and inclusion of the results in the respective chapters of the Safety Analysis Report, is assumed by the PRT.

6..4 Compliance of the plant with current requirements

The technical concept of the AES 2006 V-392M design, on which the design of the Akkuyu NPP units is based, has already been assessed and reviewed in detail during the licensing procedure for the two units at the Novovoronezh II site in Russia (regulatory reviews conducted, construction and operation licence issued). Based on the statements of the designer, the technical design concept was checked against international and national Russian requirements for nuclear safety. However, as noted in Section 6.1.1 there are significant differences between the Novovoronezh II units and the Akkuyu units.

As stated in the NR, the main requirements for the safety of the NPP are laid down in the documents referred to in Section 6.1.1, some of which have been replaced by newer versions. The design of V-509 was developed in 2013 based on the provisions of NP-001-97, which was valid in Russia at that time. In February 2016, the new regulation NP-001-15 (General Provisions for Nuclear Power Plant Safety Assurance) entered into force, which is considered to comply with IAEA standards SF-1 'Fundamental Safety Principles', SSR-2/1 'Safety of Nuclear Power Plants: Design' and SSR-2/2 'Safety of Nuclear Power Plants: Commissioning and Operation', including operational experience and lessons learned from the Fukushima accident. However, according to information provided to the PRT during the mission to the Akkuyu NPP in summer 2024, NP-001-97 remains the valid regulation in the licensing base for the Akkuyu NPP. The NR does not contain information on the technical requirements requested and applied in Türkiye for the design of the Akkuyu NPP.

- ***Assessment of robustness of plants***

6..1 Approach used for assessing safety margins

The assessment of the robustness of the plant to cover the issues of station blackout (SBO) and loss of ultimate heat sinks (UHS) was conducted according to the EU stress test specifications of May 2011. For both cases, the levels of action required in the specifications were considered and assessed. Actions and countermeasures were described and safety margins were defined in general, as well as cliff edge effects and related time periods up to their occurrence.

The approach involves an assessment of technical measures and some administrative measures that enable the plant to cope with the consequences of a total failure of power supply (SBO and/or LUHS). An SBO includes loss of power from the national grid, followed by the reactor scram and an immediate or subsequent failure of the alternative power supplies. For both types of event, an assessment of the sufficiency of residual heat removal from the reactor to avoid a core damage event was carried out.

6..2 Main results on safety margins and cliff edge effects

The review of the Turkish NR concluded that the ENSREG specification developed for the EU stress tests was followed in the process to assess loss of electrical power supply and loss of ultimate heat sink.

Based on the text of the NR, supplemented by the discussions of the PRT with NDK, as well as with Russian experts from the project organisations responsible for designing and constructing the NPP, it can be considered that the design is generally appropriate to meet the stress test requirements.

Since the NPP is currently under construction, no comparison could be made between the statements made in the NR and during discussions with the project counterparts, and the physical realisation of the plant systems and components. Therefore, the demonstration that the combined active and passive safety systems, large water reserves stored inside the containment and other design features provide sufficient robustness and time margins for all relevant accidents considered in the EU stress tests specification, was only conducted on the basis of the presented design information. The information in the NR was based on the Preliminary Safety Analysis Report (PSAR). In addition, during the discussions with NDK, information was provided on the changes that had been made to the safety relevant features of the design of the plant presented in the NR. These changes have been taken into account in the preparation of this report.

Loss of off-site power (LOOP)

Off-site power supply features

The Akkuyu NPP is planned to be connected to the Turkish power grid on a voltage level of 380 kV via six long distance (>70 km) 380 kV transmission lines and two short (<6 km) transmission lines. The NR states that one of the six transmission lines 'Komet' is to be cancelled (it had already been cancelled at the time of preparation of this report). The two short transmission lines connect the NPP to the local 154 kV distribution network through 380/154 kV autotransformers. The local 154 kV network is used to supply power to the site during construction. The NR states that it is not clear how the 154 kV grid will be utilised in future. Based on the latest information on design modifications, construction of an additional transformer, which will connect the Akkuyu NPP with the Gezende HEPP via one or both of the 154 kV transmission lines is under way.

During normal operation, power supply to the unit is taken from the 24 kV unit generator output via two 24/10.5 kV main (primary) auxiliary transformers (AT). When the main ATs are disconnected, power supply to the unit is provided from the 380 kV switchgear via two 380/10.5 kV standby auxiliary transformers.

The electrical power of the NPP units may be reduced to house load in the case of LOOP. Operation in this mode is limited to 2 hours.

On-site power supply

A back up Emergency Power Supply System (EPSS) is installed at each unit that feeds the safety relevant consumers which are needed for ensuring the operability of the safety functions, such as core residual heat removal and maintenance of the plant's integrity.

The emergency power supply of each unit is provided by two 10.5 kV standby diesel generator sets (SDGS) providing power to two separate safety systems. Each SDGS has a power capacity of 6.3 MW which is sufficient for providing power supply to all loads in one of the two redundant safety trains comprising all safety-related systems and components needed to bring and maintain the unit in a safe mode in the event of loss of power from the main auxiliary transformers and standby auxiliary transformers.

The two SDGS, along with the 10 kV and 0.4 kV switchgear of the respective safety trains of the EPSS are accommodated in separate buildings of seismic category 1 designed for the maximum design seismic impact of SSE-level earthquake. The two buildings are located at opposite sides of the reactor building to prevent their simultaneous failure in the event of an aircraft crash.

With the power supply from the SDGS, the unit is guaranteed to operate for more than 72 hours. An operational time of 8 hours for each SDGS is ensured by its respective feeder tank, and an intermediate tank has adequate fuel inventory for 72 hours. The intermediate diesel fuel storages of each of two SDGS (and the MDGS, see below) are located in close proximity to the consumer. Fuel is pumped via a pipeline from the intermediate diesel fuel storage to the feeder tanks. According to the NR, in normal operating conditions the intermediate tanks are refilled through the line from the diesel fuel storage pump station, which contains enough fuel for the SDGSs to operate for further 4 days[65]. The total volume of the diesel fuel in the centralised diesel fuel storage is 1 000 m^3 per unit (four separate tanks). In the event of blackout, the intermediate fuel tanks can be refilled by on-site tanker trucks that will deliver the required diesel fuel from the centralised diesel fuel storage or the nearest petroleum depot. According to the NR, the oil system is designed to provide such oil inventory in the feeder tank to ensure independent operation for at least 15 days.

A third diesel generator, called the 'unit DG' or Main Diesel Generator Set (MDGS) is installed in a separate building designated to serve as an internal power source for the reliable power supply to normal operation auxiliary consumers important for safety and integrity of the main equipment. The building housing the MDGS is seismic category 2, designed for the seismic impact of an OBE-level earthquake. An uninterruptible power supply fed by batteries with 2-hour capacity is provided to back up the reliable power supply to normal operation consumers. SDGS and MDGS may operate simultaneously. To ensure the independence of the power supply of consumers of different safety classes, the NPP design excludes the possibility of connecting the output of the MDGS to the loads of the safety system of the EPSS.

The PRT was informed of a design modification during meetings and written questions and answers with the regulator/licensee: A new 7.5 MW water to air-cooled Additional Diesel Power Plant (ADPP) is to be used to replace any of the three DGs (MDGS or SDGS) when they are taken out of operation for scheduled maintenance/repair. The ADPP is located near the switchyard at +19.5m level and can be manually connected to any of the four units. Organisational and technical measures ensure that the ADPP can be connected to only one of four power units. The ADPP is also intended to provide back up to the EDG in one safety train should the other train be out of service due to online maintenance.

One mobile Alternative Air-cooled Diesel Generator Set (ADGS) per unit with the power of 2 MW is provided to allow connection to the 0.4 kV sections of both trains of the EPSS if SDGSs fail. The ADGS is intended to power a limited number of consumers: emergency systems and special engineered safety features required to manage accidents that are beyond the design basis accident (BDBA) according to the Akkuyu project terminology. A list of the equipment powered by the ADGS is provided in the NR. This list has been corrected and complemented by written replies to questions and information provided during meetings with the regulator/licensee.

The ADGS will also recharge the batteries (with discharge times of 2 and 72 hours) of the uninterruptible power supply (UPS) of the EPSS. For each unit, there is a connection point, which is designed to withstand design basis external hazards and is located on the +10,5 m platform. The ADGS is containerised, mounted on a trailer and stored at its place of use (UKD), but it is not permanently connected to avoid damage during external events. The storage place is also designed to ensure protection against design basis external hazards. It is seismic category 1, designed for the maximum design seismic impact of an SSE-level earthquake. Moving and connecting the ADGS to the EPSS takes 6-8 hours. According to the NR, the ADGS is not required during the first 72 hours following a beyond design basis accident. The internal fuel tank volume of ADGS is designed for 8 hours of operation, with further operation possible after refuelling. Alarms for levels in the supply tank are displayed on the control panel.

Station blackout

The EU-STs specifications define station blackout (SBO) as a loss of all permanently installed AC power sources on and off site. For the Akkuyu NPP, this constitutes the loss of the external national grid, loss of supply from the main generator as well as the loss of standby AC power sources (both SDGS per unit). Since the unit DG (MDGS) is not connected to the EPSS, it cannot serve as a backup power source to provide power to the safety trains should the two SDGS be lost. The SBO event generates a loss of

[65] Four days according to the NR, 5 days according to later information received verbally from the designer.

function of all active safety components (except power supply from the batteries for 2 and 72 hours). Heat removal from the core can no longer be provided by the cooling systems used for normal operation conditions, AOO conditions or DBA conditions. The active systems for spent fuel cooling in the spent fuel pool are also lost.

According to the NR, an SBO is considered to be a beyond design basis condition/accident (BDBA) and so the respective technical features enter into operation and the organisational sequences enter into force. Therefore, in the event of SDGS failure, the alternative ADGS must be connected as it powers some equipment required to manage beyond-design basis accidents. Energy needed for the category 1 consumers will be supported by the uninterruptable power supply (UPS) fed by emergency batteries of the EPSS. These consumers include special devices such as valves and general instrumentation and control (I&C) systems.

The EPSS DC system consists of two independent trains. Each EPSS DC train has two banks of batteries, which are designed for 2 and 72 hours of operation, respectively, without battery recharge. During the first 2 hours, batteries provide power for functions required in the early phase of the accident such as secondary pressure limitation (BRU-A), primary and secondary overpressure protection (PRZ PORV, SG PORV), isolation of failed SG (MSIV), adjustment of full power operation of the passive heat removal system (PHRS), closing containment isolating valves, and safety injection (control of the valves in ECCS hydro accumulator lines) in the event of loss of coolant.

Depending on how the accident develops, some equipment may also be required after the first 2 hours to perform functions such as PHRS regulators control, primary overpressure protection (PRZ PORV), fuel pool drain valve to maintain sump level, valves from HA-2 and HA-3 hydro accumulators for makeup of reactor and SFP and SG PORVs. The systems required after 2 hours are powered by the 72 hour batteries. Connection of the equipment to the 2 or 72 hour batteries is determined on the basis of the accident analyses. After battery discharge, power supply to components will be stopped which leads to the loss of their function and the loss of safety-related I&C systems, valves, switches and measuring devices. After 72 hours, it is assumed that ADGS will be connected and will provide power to the safety systems and special engineered safety features required to further mitigate the beyond design basis accidents, namely makeup of the primary circuit and SFP, PHRS, and containment pressure reduction through spray system. The batteries will also be recharged by the ADGS.

The operability (lighting, ventilation) of the main control room (MCR) and emergency control rooms (ECR) is powered by batteries and ADGS. The emergency and post-emergency monitoring system and the Engineered Safety Feature Actuation System (ESFAS) receive power from the 72-hour storage batteries.

Heat removal in the event of an SBO

In the event of an SBO, the heat removal from the reactor core and - based on this - the retention of the barrier system functionality, is maintained only by the steam generator passive heat removal system (SG PHRS – JNB50-system). The system consists of four independent channels (4 x 33%) – one for each steam generator - which operates based on natural convection circuits. Each circuit includes two air heat exchangers, with two air dampers – upper and lower - to open and close the air flow, as well as pressure regulating devices (one in each channel – two channels per train). Each pressure regulating device is equipped with two drives: one for passive and one for active operation. The heat removal is conducted through the Chain Reactor – Steam Generator – SG PHRS – Atmospheric Air (heat sink). Heat is removed to the atmosphere via the air heat exchangers located in channels integrated in the containment dome structure. Three (out of four) trains of the SG PHRS ensure the design operability of this safety feature and the adequate heat removal from the fuel elements for 72 hours[66].

Following an SBO and the closing of the turbo generator isolation valves, the pressure in the secondary circuit increases and actuates the BRU-A valves on the SGs, which are fed by the uninterruptible power supply.

During normal operation of the unit, the SG PHRS is in standby mode. This means the air dampers are closed and the pressure regulating devices (regulators) are open. The SG PHRS operation is initiated passively in the event of an SBO due to the loss of electrical power supply to the solenoid valves that hold the air dampers in the closed position. There is a 30-second delay between the loss of power and the opening of the dampers. As soon as the dampers are open, the regulators are adjusted according to the pressure on the secondary loop. The pressure-regulating device controls the air flow through the air heat exchanger by opening/closing under direct action of the pressure of the steam-condensate of the secondary circuit, and therefore regulates the pressure in the steam-condensate circuit. The device becomes fully open in standby mode (when the normal design pressure in the steam-condensate circuit is 7 MPa) and only begins to close when that pressure falls below 6.5 MPa. The full closure of the pressure regulating device occurs on reaching a SG pressure of 5.8 MPa (the pressure maintaining mode). If it becomes necessary to switch to the cooldown mode, the regulators can be completely opened and powered by the 72-hour batteries until the ADGS is connected.

The PHRS reaches its full design capacity after 120 seconds. Operation of the SG PHRS channels decreases pressure in the steam generators in accordance with the SG PHRS performance parameters, so the BRU-A on all the steam lines of the SGs are closed, loss of boiler feed water in the steam generators is stopped and the level in the SG is stabilised. The PHRS continues to operate in autonomous mode and does not require power supply from the batteries while operating in pressure regulating mode[67].

Spent fuel pool cooling under SBO conditions will be carried out by boiling water in the spent fuel pool and the evaporation of water above the level of the fuel assemblies. The fuel pool's excess volume above the normal level required for the spent fuel storage is at least 650 m^3. Under SBO conditions, this excess water is only used for the SFP.

In this way, under normal SFP loading conditions, cooling is assured for more than 72 hours without the uncovering of the fuel elements. Under the most severe loading conditions, i.e. with full emergency unloading of the reactor core into the spent fuel pool, the grace time before the uncovering of the fuel elements is only 35.6 hours. In this case, water can be provided from HA2 and HA3 via DN32[68] drain lines to compensate for the water lost due to evaporation. The required flowrate of 31 m^3/h (worst case) can be delivered, and this can extend the grace time before fuel uncovering beyond 72 hours. This requires manipulation of several valves, all of which are powered by 72-hour batteries. The combined available volume of water in HA2 and HA3 can provide a little over 72 hours of cooling for

[66] The deterministic analysis of the SBO event presented in the NR considers the operation of all four trains of the PHRS. However, information obtained from the designer during the desktop review and preliminary mission indicates that three PHRS trains are sufficient to meet the acceptance criteria in the event of an SBO (without primary circuit break).

[67] Nevertheless, the passive pressure-regulating function of the regulator can be overridden by an electrical drive if, during long-term loss of power supplies, it is decided not to maintain the reactor in hot standby. Therefore, the regulator is powered by the 72-hour batteries. Activation of the drive fully opens the pressure-regulating device and puts the PHRS into cooldown mode.

[68] Originally the drain lines were DN25. Following modification of the HA-2 pipeline for spent fuel pool cooling in the event of full fuel unloading, the drain lines were increased to DN32.

the SFP. Further cooling may be provided using the additional fuel pool cooling system (JMN61 pump + JMN60 heat exchanger).

To use this system, the 2 MW mobile alternative DG (ADGS) must be connected. When removing heat using the additional fuel pool cooling system, the heat removal to the ultimate heat sink proceeds via the alternative component cooling system as intermediate circuit. This requires the connection of a mobile heat exchanger (KAA25) and the use of the mobile pumping unit (11-41PEC10AP003). The steps to connect the mobile equipment take several hours. The specialised steps, procedural rules and timeframe will be laid down in special technical regulations. In the unlikely event that an SBO occurs during refuelling with the reactor vessel open, the heat removal will be interrupted, boiling in the reactor pressure vessel will occur rapidly and the fuel assemblies will be uncovered if countermeasures are not implemented quickly. The grace time before countermeasures are required is about 2.8 hours in the most severe loading condition, which is prior to refuelling. After refuelling, the grace time increases to 7.8 hours. To cope with this situation, it is possible to use the hydro accumulators HA-2 and HA-3 to flood the reactor pit with boric acid solution. In the initial design described in the NR, this was to be done remotely by the operators within 2.8 hours by opening the valve(s) at the drain of the HA-2 (and later HA-3), which are powered by 72-hour batteries.

Following design modifications made during the course of the PRT review, the actuation of HA-2 will be done automatically on the basis of level measurements in the reactor. Level monitoring sensors are installed temporarily during shutdown modes prior to reactor depressurisation and removed again before return to operation. Other than the actuation of HA-2 and HA-3, no further actions would be required within 72 hours.

If the normal AC supplies have not been restored within the first 72 hours, the required heat removal may be achieved using the first train of the spray system (JMN11 pump powered by the ADGS and JMN11 heat exchanger) or using the additional fuel pool cooling system (JMN61 pump + JMN60 heat exchanger). Both require the connection of the 2 MW mobile alternative DG (ADGS). The heat removal to the ultimate heat sink proceeds via the alternative component cooling system as intermediate circuit, as described above for the SFP cooling.

<u>Time margins until 'cliff edge effects'</u>

<u>Loss of off-site power (LOOP) and station blackout (SBO)</u>

As stated in the NR, the loss of off-site power supply (LOOP) is seen as a DBA, to be handled by the alternative on-site power supply (EDGSs). For various LOOP scenarios, the availability of specific systems to prevent a core damage accident demonstrates that sufficient core cooling as well as cooling of the spent fuel pool can be maintained until the normal power supply is restored. Automatic protection actions of the safety systems transfer the reactor plant to a safe state that can be maintained for an unlimited time subject to power supply from the EDGS.

According to the NR, the EDGs can remain operational for more than 72 hours without external support or aid. Sufficient fuel for four more days of operation is stored at the site and can be delivered to the EDGS by on-site trucks.

Therefore, significant for the assessment of possible “margins to cliff edge” as stated in the EU STs specifications including the respective time margins, is the SBO, which is a BDBA condition in the Akkuyu project terminology.

For the reactor core, cooling is provided by the PHRS. Operation of the PHRS is without time limits. In the case of the SFP with a fully unloaded core, cooling is assured by HA-2 and HA-3 for a little over 72 hours. Beyond this time, the alternative component cooling (KAA25AP001) and mobile pump unit (11-41PEC10AP003) are required, as well as connection of the ADGS to provide power for the pump of the first train of the spray system (JMN) or the additional fuel pool cooling system (JMN61 pump).

If an SBO would occur during refuelling with the reactor vessel open, a very short time period was originally envisaged for operator action before the cliff edge effects would occur. However, design changes have already been implemented to provide automatic actuation of HA-2 on the basis of the monitored water level in the reactor vessel.

Loss of ultimate heat sink (LUHS)

Heat sink for normal operation

At the Akkuyu NPP, the heat sink in normal operation mode is the Mediterranean Sea. Heat is removed from the condenser directly to the ultimate heat sink via the main cooling water system (PA system). The conventional cooling water (PC) system provides heat removal from the equipment in the turbine building (UMA), chiller building (UQR), normal operation standby diesel generator station (UBN) and compressor building (UTF). Normal operation residual heat removal from the SFP and primary circuit is done using the safety systems JMN (containment spray and spent fuel pool cooling system) and JNA (emergency and planned primary circuit cooldown and spent fuel pool cooling system). Heat is transferred via the chain JMN/JNA - essential service component cooling system (KAA10-20) - secured cooling water system (PE)- Mediterranean Sea. The PE system comprises two independent channels (2 x 100%) with heat exchangers, cooling water pumps and pipelines connected to two essential service pump houses (11/12UQC), which are interconnected to the single main pump house (10UQA). The main pump house also incorporates the non-essential service pump houses. As presented in the NR, expansion joints separate the 11/12UQC from the 10UQA within the common building.

Primary ultimate heat sink

The ultimate heat sink for removing residual heat from the fuel elements in the reactor core is the Mediterranean Sea via the essential service component cooling system (KAA10-20) and the secured cooling water system (PE).

Residual heat may be transferred to the essential service component cooling system via the route primary circuit – steam generator – steam generator emergency cooldown system (JNB10), or directly from the primary circuit via JNA system (emergency and planned primary circuit cooldown and spent fuel pool cooling system) depending on the initiating event.

The heat removal from the spent fuel pool in normal operation as well as under AOO and DBA conditions is carried out by the JNA (emergency and planned primary circuit cooldown and spent fuel pool cooling system) and /or the JMN (containment spray and spent fuel pool cooling system) depending on the situation. These systems also interface with the KAA10-20 system (essential service component cooling system). All three systems comprise two independent channels (2 x 100%).

Loss of primary ultimate heat sink

In accordance with IAEA TECDOC-1791, loss of ultimate heat sink (LUHS) is to be understood not only as the loss of the ultimate heat sink itself, but also as the failure of the heat transfer chain to the UHS. In the event of loss of capability to transfer heat to the UHS, which is considered as a BDBA condition according to the Akkuyu project terminology, the heat removal from the reactor core and, based on this, the retention of the barrier system functionality, is maintained in the same way as for an SBO as presented above, i.e. by the steam generator passive heat removal system (SG PHRS – JNB50-system).

Upon loss of the capability to transfer heat to the UHS, both the reactor trip and startup of the SG PHRS are triggered automatically on the basis of monitored process parameters. During PHRS startup and achieving stable operation, the residual heat is removed due to the automatic opening of the BRU-A valves on the SG. The operation of the BRU-A and SG PHRS proceeds in a similar way to that presented above in the event of an SBO. The alternative ultimate heat sink is atmospheric air.

Spent fuel pool cooling under LUHS conditions will be carried out by means of boiling water in the spent fuel pool and evaporation of water above the level of the fuel assemblies.

In this way, under normal SFP loading conditions, cooling is assured for more than 72 hours without the uncovering of the fuel elements. Under the most severe loading conditions, i.e. with full emergency unloading of the reactor core into the spent fuel pool, the grace time before the uncovering of the fuel elements is only 35.6 hours. In this case additional cooling may be provided using exactly the same systems as presented above in the event of an SBO.

In the unlikely event of an LUHS occurring during refuelling with the reactor vessel open, the heat removal is interrupted, boiling in the reactor pressure vessel will occur rapidly and the fuel assemblies will be uncovered if countermeasures are not implemented quickly. The grace time before countermeasures are required is about 2.8 hours in the most severe loading condition, which is prior to refuelling. After refuelling, the grace time increases to 7.8 hours. To cope with this situation, it is possible to use hydro accumulators HA-2 and HA-3 to flood the reactor pit with boric acid solution. In the original design described in the NR, this was envisaged to be done remotely by the operators within 2.8 hours by opening the valve(s) at the drain of the HA-2 (and later HA-3), which are powered by 72-hour batteries. Following the design modifications made during the course of the PRT review, the actuation of HA-2 will be done automatically on the basis of level measurements in the reactor. Level monitoring sensors are to be installed temporarily during shutdown modes prior to reactor depressurisation and removed again before return to operation. Other than the actuation of HA-2 and HA-3, no further actions would be required within 72 hours.

If the normal connection to the UHS is not restored within the first 72 hours, the required heat removal may be achieved using the first train of the spray system (JMN11 pump powered by the ADGS and JMN11 heat exchanger) or using the additional fuel pool cooling system ACSFP (JMN61 pump + JMN60 heat exchanger). Both require connection of the 2 MW mobile alternative DG (ADGS). The heat removal to the ultimate heat sink proceeds via the alternative component cooling system as intermediate circuit, as described above for the SFP cooling.

As the AC power supplies are available, a further chain of heat removal from the reactor is available through the normal operation safety-critical fuel pool cleaning and water supply system (FAL10), with heat exchangers and pumps – intermediate circuit of the normal operation consumers of the UJA building (Component cooling water system for normal operation loads - PJA) – additional normal operation service water pumps. It is designed for Level 2 of defence in depth and is safety class III and seismic class II. Power is supplied from normal operation diesel generator.

<u>Loss of ultimate heat sink and alternative ultimate heat sink</u>

The alternative heat sink for cooling the reactor core is the SG PHRS, which is a passive safety feature able to operate after actuation without power supply and active control for an unlimited period of time. Therefore, a failure of the SG PHRS and as a consequence a loss of alternative ultimate heat sink, in combination with loss of the primary UHS, is extremely unlikely. In this special case, severe accident measures according to the actions covered in chapter 7 would have to be launched.

For the SFP or for the case of the open reactor vessel during refuelling, using the water reserves in the hydro accumulators can provide more than 72 hours of grace. If the normal connection to the UHS has not been restored within this timeframe, and the cooling chain via the alternative component cooling system is also not possible, severe accident management measures may also need to be launched.

<u>Time margins until 'cliff edge effects'</u>

According to IAEA SSR-2/1 Rev. 1, the design of the NPP should provide assurance that a small deviation in a plant parameter will not lead to a cliff edge effect.

For the reactor, cooling is provided by the PHRS, which can operate without time limits.

In the event of an SFP with a fully unloaded core, cooling is assured by HA-2 and HA-3 for a little over 72 hours. After this, the alternative component cooling system (KAA25AP001) and the mobile pump unit (11-41PEC10AP003) are required, as is connecting the ADGS to provide power for the pump of the first train of the spray system (JMN) or the additional fuel pool cooling system (JMN61 pump).

If a LUHS were to occur during refuelling with the reactor vessel open, under the original design the operator would have a very short time to act before cliff edge effects would occur. However, following design changes, the water level in the reactor vessel is monitored and HA-2 is automatically triggered should the need arise.

The operational steps described above could be suitable to avoid time-based cliff edge effects.

Loss of UHS with an SBO

Regarding the cooling of the reactor core, the combination of an SBO and an LUHS (SBO+LUHS) is handled in the same way as if these events were to occur individually, since the SG PHRS removes heat to an alternative ultimate heat sink and does not require AC power supplies. Similarly, for the SFP, and in the event of an SBO+LUHS during refuelling with an open reactor vessel, the hydroaccumulators are available to provide the required cooling for more than 72 hours without AC supplies. The alternative component cooling system, powered by mobile means, would also be available to provide cooling after 72 hours.

6..3 Strong safety features and areas for improvement identified in the process

The design concept of VVER-1200 Version V-509 incorporates features of Version 392M constructed at Novovoronezh II nuclear power plant and a new VVER TOI design currently being developed in Russia. Version V-509 combines active 2 x 100% and passive 4 x 33% safety systems as well as special engineered safety features to ensure the residual heat removal from the core, and to avoid core damage or mitigate the consequences of a BDBA. Passive safety systems diversify the safety functions of active safety systems and serve as a compensatory measure if an active safety system fails. In addition, as identified in the lessons learned exercise following the accident at Fukushima Daiichi, the concept includes a requirement for autonomous operation for 72 hours in the event of a BDBA.

The alternative heat sink for cooling the reactor core is the SG PHRS, which is a passive safety feature that has already been installed in the Kudankulam NPP in India and the Novovoronezh II NPP in Russia. Experience gained during regular testing at these sites can be used by the Akkuyu NPP. Nevertheless, as the PHRS is extremely important for the Akkuyu NPP design in terms of safety justification, the PRT recommends a thorough assessment of all analyses conducted. Such analyses need to include not just the PHRS steam-water cooling loop, but also the PHRS air-cooling loop.

The large inventory of cooling water reserves in HA-1, HA-2 and HA-3 and the excess SFP coolant provide an additional safety margin for the residual heat removal in the DBA and BDBA events.

A key lesson learned from the accident at Fukushima Daichi NPP was the need to improve the effectiveness of the independence between different levels of defence in depth (DiD). This is reflected in the WENRA Safety Objective O4 for new NPPs and in IAEA SSR-2/1 Rev. 1. During its assessment, the PRT noted that some safety systems have multiple uses spanning several different levels of DiD, including normal operation, which is not in line with the intent of the above requirements. For example, within the limited scope of the stress tests, the PRT identified that the permanently operational cooling system for the SFP also acts as the safety-related containment spray (CS) system - JMN. To be aligned into one or another function, the manipulations of different controls is necessary. Such a dual use, i.e. operational use of the (highly important) dedicated safety system (as the containment spray is) is rather unusual in the international practice.

In addition, the active high-pressure and low-pressure emergency cooling system fulfils both the operational function of residual heat removal during operational shutdown and the emergency cooling of the core in the event of coolant loss accidents and therefore has a dual function.

Two safety systems are also used for the operational cooling of the SFP: the emergency and planned cooldown system and the containment spray system. Another safety feature is that the fuel storage pool simultaneously contains the coolant for the irradiated fuel elements located in the pool as well as additional coolant inventory for active refilling of the primary circuit in the event of a loss of cooling accident (LOCA). Proof of the sufficient availability and functionality of the safety systems emergency and planned cooldown system, spray system, emergency boron injection system (additional boron system) is therefore extremely important for the safety assessment.

The PRT acknowledges the importance of a sophisticated system of defence in depth (DiD) levels for assuring nuclear safety in all operational conditions, based on the technical design to be developed in the Akkuyu NPP. It therefore recommends developing and implementing a comprehensive DiD concept describing the technical features and organisational measures, and assuring independence between the different levels (special finding T2-5 in the table of findings).

The PRT recommends that concrete measures be taken to improve the reliability of active safety systems, in particular when a system is out of service for maintenance or repair, which is a main contributor to the associated risk. Special attention should also be paid to assuring the operability of the other support systems that are needed for the safety functions to fulfil their mission (e.g. the I&C, cooling, HVAC etc.) (special finding T2-1 in the table of findings).

6..4 Possible measures to increase robustness

As stated in the NR, NDK has identified measures that should increase the robustness of the power plant in the event of various extreme scenarios.

In the event of power supplies being lost, the following measures are proposed:

- Accident management procedures should be developed that include:
  - actions during a blackout with complete de-energisation of all four NPP power units,
  - actions during complete de-energisation of one, two or three power units with their disconnection from the power grid with switchover to auxiliary power or/and isolation of the operating power unit to 154 kV load.

In the event of both a loss of power and a loss of ultimate heat sink, the following measure is envisaged:

- NPP management and qualified operating personnel should be prepared for and trained in managing accidents involving NPP blackout and/or loss of ultimate heat sink.

6..5 Measures (including further studies) already decided, implemented by operators and/or requiring follow-up by regulators

Regarding the information provided in the NR, NDK announced several changes to the safety relevant features in the design of the plant during the discussions that took place while the peer review report was being prepared. For example:

- An additional heat removal circuit (in 10FAL system) has been introduced for DiD Level 2 in the design for a long-term makeup and cooling of the core for shutdown states with reactor open head to decrease the probability of IE loss of heat removal due to multiple failures.
- The original HA-2 pipeline diameter drain line has been increased from DN25 to DN32 diameter to ensure adequate spent fool pool cooling in the event of full fuel unloading.
- In the event of loss of heat removal during refuelling with the reactor vessel open, HA-2 will be activated automatically on the basis of level measurements in the reactor.
- The additional fuel pool cooling system (ACSFP) has extended its original function from cooling just the SFP to also cooling the reactor. For this, a connection from ACSFP to the core is provided and can be used for cooling the core in the event of BDBAs that need special purpose equipment (alternative component cooling circuit KAA25, mobile diesel generator and mobile pump unit).

- The power capacity of the additional water to air-cooled diesel generator for the entire site has been increased from 3 MW to 7.5 MW, giving it the same power capacity as the emergency diesel generators. The generator can be connected to the emergency power supply sections of any train to enable the repair or online maintenance of one of the emergency diesel generators.
- Construction of the additional transformer is under way, which will connect the Akkuyu NPP with the Gezende HPP via one or both of the 154 kV transmission lines.

All these changes have already been considered by the PRT when preparing this report.

6..6 Peer review conclusions and recommendations specific to this area

For Topic 2 of the Turkish NR, the review concluded that the ENSREG specification developed for the EU stress tests was strictly followed in the assessment process for loss of electrical power supply and loss of ultimate heat sink. Robustness and time margins were theoretically demonstrated for all relevant accidents considered in the EU stress tests due to the mix of active and passive safety systems, big water reserves stored inside the containment and other features.

Nevertheless, the PRT concludes that some safety elements need to be clarified and strengthened.

The availability of only two EDGs per unit raised concerns about the reliability of the emergency power supplies and the resulting reliance on the passive systems and mobile means, particularly in situations in which one EDG system is out of service for maintenance or repair, which is a main contributor to the associated risk. This concern has now been addressed by the inclusion of an additional diesel generator. This additional generator can be used to replace the diesel generators taken out of service for online maintenance, thereby retaining the 100% redundancy level during maintenance. (See also special finding T2-1 in the table of findings.)

Given that there is only 2 x 100% redundancy on the safety systems, the PRT recommends that special attention be paid to monitoring the reliability and availability of the systems and components throughout operation. This applies not only to the reliability of electrical supply, but also to the assurance of operability of the other support systems needed for the safety functions to be able to fulfil their mission (e.g. the I&C, cooling, HVAC etc.).

In accordance with IAEA SSR-2/1 Rev. 1, Requirement 28, the PRT consider it indispensable that limitations on the timing and duration of preventive maintenance and online repair, and related operational restrictions in the event that safety systems are unavailable, be specified in the OLC on the basis of an appropriate risk analysis. This applies to all safety systems, not just the EDGs. (See also special finding T2-1 in the table of findings.)

Because of the relatively high estimated frequency of the loss of normal connection to the primary ultimate heat sink of 1.9×10^{-3}, the PRT recommends that the event be considered as a PIE in the design basis. It also recommends ensuring that the analyses provided in chapter 15 of the safety analysis report be conducted in line with the appropriate rules and requirements for analysing design basis events. (See also special finding T2-2 in the table of findings.)

The PRT understands that mobile means (alternative, air-cooled mobile diesel generator (ADGS), mobile diesel-driven pump unit (PEC10) and mobile heat exchanger (KAA25)) have a role to play in the management of accidents in which the goal is to prevent fuel damage, as well as in the management of severe accidents. They are therefore likely to be on the list of equipment ultimately required to prevent large or early releases. Unless it can be demonstrated that the mobile means are not ultimately required to prevent large or early releases, they should be protected against beyond design basis external hazards. Supporting the assessment in Topic 1 of possible external hazard impacts, and supplementing its recommendations based on the assessment results, the PRT also recommends increasing the robustness against flooding, of both the mobile means at their designated storage locations, and of their connecting points to the plant systems. If vulnerabilities are identified, action to ensure adequate protection must be taken. (See also special finding T2-3 in the table of findings.)

The PRT has concerns about the check valve that enables passive operation of the HA-2 hydroaccumulators. In particular, if the valve does not work reliably, it could undermine the 'passive' feature of this system. The PRT recommends that NDK request a robust demonstration of the reliability of the check valve or of the measures in place to ensure the reliable functioning of the system. (See also special finding T2-4 in the table of findings.)

The PRT noted that some safety systems have multiple uses spanning several different levels of DiD, including normal operation, which is not in line with the intent of IAEA SSR-2/1 Rev 1. It therefore recommends an intensive review of the DiD concept prepared by the licensee which should contain the technical features and organisational measures as well as the assurance of independence between the different DiD levels. (See also special finding T2-5 in the table of findings).

The PRT was informed that the functionality of the PHRS was modelled by using a specially developed computer code. The code was validated experimentally against results from a half scale, partial mock-up in one of the supplier's research facilities. The experiments were conducted at reduced operational conditions, and the model was used to extrapolate to full-scale PHRS geometry and operating conditions. Commissioning tests in a real scale PHRS at the Kudankulam plant were carried out, which provided additional data for validation of the computer code. However, full power operating conditions could not be achieved during those tests. Due to the extreme importance of the PHRS for the safety justification for the design of the Akkuyu NPP, the PRT recommends a thorough assessment of all the analyses undertaken. In view of the frequency with which the PHRS is to be called upon, a thorough review is recommended, followed by conservative DBA analyses (as opposed to the best estimate, which is acceptable for a DEC/BDBA). This should include envisaged maximum allowable tube plugging in the SG and in the PHRS itself. Such analyses need to include not only the PHRS steam-water cooling loop, but also the PHRS air-cooling loop. The air flow and the pressure drop analysis across the full spectrum of operational conditions needs to be corroborated and values independently confirmed for the credit to be taken for the PHRS operation in the event of an SBO. (See also special finding T2-6 in the table of findings.)

The JNA system is designed in such a way that one train's high-pressure injection is made into two of the cold legs and the other train's high-pressure injection is made directly into the RPV downcomer (joining a common injection pipeline used also by the hydro accumulators). The loss of water injected if one or the other train fails (of a total of two), might have very different consequences. In the event of a LOCA caused by a break of the connecting pipe to the RPV shared with the hydro accumulators, it is not just the JNA train that would be lost, but also the hydro accumulators that use the same flow path. The fact that the flow paths are not symmetrical may lead to a very different plant response to specific initiators and JNA failure combinations. The PRT recommends that a detailed analysis be undertaken (or reviewed if it already has) to assess the most critical conditions with loss of one or another train of the JNA system and its effects. It should be demonstrated that in any combination of an initiator with a failure of either train of the JNA, the function of the system is ensured and the design conditions to protect the core are fulfilled. (Special finding T2-7 in the table of findings.)

7 PLANT ASSESSMENT RELATIVE TO SEVERE ACCIDENT MANAGEMENT

7.1 Description of the current state of nuclear power plants in Turkey

7.1.1 Regulatory basis for safety assessment and regulatory oversight (national requirements, international standards, licensing basis already used by another country)

At the time the national report was prepared and due to the project being in its early stages, there was no on-site emergency plan and no severe accident management programme during the preliminary stages of the stress test. The 2019 National Radiation Emergency Plan and the 2022 Mersin Provincial Radiation Emergency Plan in 2022, set out the duties and responsibilities of national and local institutions and organisations. As stated in the national report, the plans and procedures for accident management are expected to reflect the IAEA recommendations and the world's experience after the Fukushima accident. The Severe Accident Management Programme was developed by the licensee and evaluated by the designer before it was submitted to the regulatory body. In line with the DiD principle, specific procedures and guidelines were developed for the different levels:

- *Level 3: Emergency operating procedures (EOPs);*
- *Level 4a: Beyond-Desing-Basis Accident Management Guide (BDBAMG);*
- *Level 4b: Severe Accident Management Guidelines (SAMGs);*
- *Level 5: Regional and national emergency plans.*

A full-scope simulator of the main control room is being used to train personnel to implement the procedures and guidelines. According to the information available, however, both the training and the exams will be held in Russian, which may limit regulatory oversight due to language barriers. To overcome the possible issues, in May 2022, in coordination with external services from the MCR, for example during severe accidents, the applied approach ensures that at least one of the operators has to be a Turkish citizen, and therefore fluent in Turkish. After the phase 1 visit to Ankara in May 2022 it was clarified that the Severe Accident Management Programme and the emergency plans will be developed and tested before loading the fuel into the reactors. During the plant visit, NDK carried out a regulatory review of EOPs, BDBAMGs and SAMGs, as part of the licensing of the operations. It was explained that the development of the EOPs, BDBAMGs and SAMGs included various stages, including analytical substantiation, correction, validation and verification, before the loading of fuel, final adjustment after commissioning, and piloting the commercial operation.

Since 2018, when the national report was issued, besides the National Radiation Emergency Plan and the Provincial Emergency Plan, the legal and regulatory framework for emergency preparedness and response was significantly extended and improved, which included the issuance of NP-001-97 (Emergency Planning for Protection of Personnel and Population in Case of Accidents, and Accident Management) and NP-005-16 (Emergency Response and SA situations). A full-scale exercise was carried out in December 2022 and March 2023 to simulate a general emergency, including evacuation and on-site and off-site decontamination activities, following a hypothetical radioactive release. NDK evaluated the overall response as sufficient.

Türkiye has ratified and adopted several international conventions, including the Convention on Nuclear Safety and the Convention on Early Notification of a Nuclear Accident, and has concluded several agreements with other countries for cooperation on nuclear safety. The concept of 'practical elimination' is not mentioned in the national report. As stated by NDK during discussions, the concept of practical elimination has not been included in the Turkish regulations and regulatory framework. A document on 'practical exclusion' has been prepared by the licensee to describe the overall concept applied in the design of the plant to practically eliminate sequences of events that may lead to large or early radioactive releases. It is not clear whether this concept or an equivalent safety objective is being used to assess the safety of the plant and whether this important IAEA and European safety requirement has been considered during regulatory review and licensing. The latest WENRA safety reference levels have not been considered as Türkiye is not a member of WENRA, however certain European standards (e.g. European Utility Requirements) are applied on a case-by-case basis.

7.1.2 Main requirements applied to this specific area

There are no detailed requirements regarding severe accident management in the Turkish regulations. IAEA standards, in particular IAEA SSR 2/1, NS-G-2.15 (Safety Guide ‘Severe Accident Management Programmes for Nuclear Power Plants’), is explicitly mentioned in the list that acts as the basis for plant licensing. NDK plans to replace NS-G-2.15 with SSG-54 that superseded the latter one. On emergency preparedness and response, IAEA standards, including GSR part 7, are used for aspects not covered by the national regulations.

A frequency lower than 1E-5 per reactor year is defined as the target for core damage frequency (CDF) and for a large release frequency (LRF), the target for which is less than 1E-7 per reactor year. In the Turkish regulations, the LRF is defined as a release that requires the evacuation of populations beyond the area covered by planning protective measures. As stipulated in the Regulations on Radiation Protection in Nuclear Emergencies and the National Radiation Emergency Plan the planning zones comprise the Precautionary Action Zone (PAZ), extending to 5 km, and the Urgent Protective Planning Zone (UPZ), extending to 20 km. The dose criteria for the UPZ are an effective dose and an equivalent dose to the fetus equal to 100 mSv in 7 days. One must therefore conclude that the frequency calculated in PSA Level 2 should apply to avoid evacuation of the public beyond a radius of 20 km. NDK clarified in its responses that the applicant must demonstrate that the calculated radius of the emergency zone is less than the above distances, in the case of an accident with the most severe consequences.

7.1.3 Technical background for requirement, safety assessment and regulatory oversight

As stated in the national report, a full-scope PSA (internal initiating events, external and internal hazards, in all operating states) is included in the documentation for a construction licence, which will be updated for the commissioning licence. PSA Levels 1 and 2 were developed during the design of the reactors, in line with Russian regulations and IAEA standards. Turkish regulations have been developed for PSA and have also been applied during the licensing, as clarified by NDK in their response to relevant questions from the peer review team before and during the phase 2 visit to the Akkuyu NPP. At these PSA levels, frequencies for core damage in the reactor and fuel damage in the spent fuel pool, transport casks etc. fulfilled the requirement to be below the target of 1.00E-05/year. PSA Level 2 at that stage was based on conservative assumptions due to the early phase of the project and the unavailability of the details of the final design. It was stated by the licensee that PSA Levels 1 and 2 are and will be used in the development of the Severe Accident Management Programme (EOPs, BDBAMGs, SAMGs). NDK requested an updated version of PSA Levels 1 and 2 and the associated validation of the EOPs, BDBAMGs and SAMGs. According to the available information these documents are currently under regulatory review.

7.1.4 Compliance of plants with current requirements (national requirements)

As provided in the national report and in the responses from NDK, significant parts of the Accident Management Programme, in particular the EOPs, BDBAMGs and SAMGs, are still being developed and under regulatory review, based on the Russian and recent Turkish regulations and the relevant IAEA standards. The regulatory review has not yet been completed, and formal approval has not been given. Approval of the Accident Management Programme will be given as part of the licensing for operation, prior to fuel loading. In particular, it is intended to complete the validation and adjustment of EOPs, BDBAMGs and SAMGs after the reactor has been commissioned.

As no detailed national requirements have been set out for severe accident management, the PRT has assessed the compliance of the plant design and the SAMGs against the relevant international requirements (particularly IAEA SSR-2/1 Rev. 1 and WENRA safety reference levels), based on the information that was available to the PRT during the stress test process. In general, the SSC that performs the safety functions necessary for severe accident management at DiD Level 4 should be as independent as possible from other levels of defence, as stipulated by Requirement 4.13A of IAEA SSR-2/1 (Rev. 1). The PRT has stressed the importance of this principle in their review since, prior to a postulated core melt accident, all the preceding DiD levels would have failed, and it is highly unlikely that safety systems common to several DiD levels would be available to both prevent and mitigate a severe accident.

7.2 Assessment of the robustness of plants

7.2.1 Adequacy of current organisations, and operational and design provisions

NDK, the licensee and the vendor all presented a high level of knowledge and understanding of the technology and of what happens during severe accidents. The organisations involved were deemed to be well staffed and the personnel had the proper qualifications to both develop and review the management of severe accidents in terms of design, strategies and guidelines. During the consultations with NDK, the licensee and the representatives of the Vendor, all parties behaved in a transparent and supportive manner giving knowledgeable answers to the PRT. However, the fact that certain documents were only at an early stage of development made it more difficult for the PRT to make an assessment.

The defence in depth principle has generally been taken into consideration and applied in the design, but some seemingly overlapping design solutions were identified where systems seem to perform similar functions at various levels of defence in depth. Since defence in depth is a fundamental safety principle of all modern NPP designs, the licensee should treat this principle as a high priority to ensure and demonstrate their compliance with the principle and so that the regulator can verify that the design complies with the DiD approach.

A reoccurring issue identified during the stress test was the lack of information and the limitations on NDK's access to information, which may pose an obstacle for the regulator to consider every relevant factor during its decision-making process, especially when the limited information presented to the regulator seems to be unclear or self-contradictory. The following are a few examples of such cases:

- The models and calculations of containment behaviour under severe accident conditions were missing from the submitted PSAR, therefore it was not possible to verify in the early stages of the peer review process the claims made by the licensee about the design loads of the containment structure . The issue was resolved as both the review process and the Akkuyu project progressed over the years and, based on the latest information available, it was explained that calculations of containment behaviour under severe accident conditions, as well as graphs of hydrogen concentration, pressure and temperature in the containment, are included and presented in the SAR.
- There were claims in the PSAR that the KLB22 system (annulus ventilation) performs functions at all DiD levels, and that it is able to create a negative pressure in the inner containment, which was in contradiction of the system's function as an annulus ventilation system. There was also contradictory information about the system functions during a severe accident, and the time frame for its operation (before/after 72 hours). During the Akkuyu site visit, however, it was clarified by the Vendor that the system is only required to perform its function in DBA scenarios, where the fulfilment of the acceptance criteria cannot be achieved without it. For

lower DiD levels (Level 3b and 4) it is only used when available as part of the ALARA approach (if it is operable under SAM conditions before or after the first 72 hours).

In particular, the containment spray function is applied in both DBA and BDBA accident management, as well as for SFP cooling, even though alternative means to provide power supply, pumps and a cooling circuit to support the spray system are planned in case of a core melt accident. In summary, the verification of the adequacy of the design is highly dependent on the quality of the information presented in the documentation available to the PRT and, if such seemingly inconsistent information is identified, then corrections and clarifications should be provided for the regulator as soon as reasonably possible. Additionally, a detailed description of the various mitigation strategies, SSC functions and operator actions should be presented either in the PSAR or in a separate document to help both the licensee and the regulator have an in-depth understanding of the design.

In terms of its overall design, the facility can be considered robust. It applies a novel design approach (compared to other similar facilities) to ensure that a severe accident is avoided primarily through passive means, which it is claimed will ensure that for almost all cases no core damage can occur for the first 72 hours after the occurrence of the initial event. Again due to the lack of approved and submitted analyses or lack of access by the regulator to the relevant documents, the design of these passive solutions, however, needs to be demonstrated under certain specific conditions, such as the maximum pressure, hydrogen deflagration and local burns during an accident,. For instance, hydrogen deflagration and detonation are currently being analysed by the licensee and vendor, therefore the analyses have not been submitted to NDK and consequently not subjected to review by the PRT.

During the Akkuyu site visit, NDK informed the PRT that they had received the overall environmental qualification programme, and that they were reviewing it. The programme includes calculations of the environmental conditions inside the containment during a severe accident. It also includes a plan on how the safety features used for severe accident management will be adapted to the environmental conditions. The representatives of the licensee informed the PRT that the component requirement specifications include environmental conditions, and that the qualification tests are performed after selecting the components. If no suitable component is found on market, additional justifications will be provided. The PRT saw one example of valve specification with the temperature values. There was no time, however, to check any additional examples related to severe accident management instrumentation which will be installed in harsh temperature, pressure and radiation conditions inside the containment.

The PRT concluded that, in general, there is an overall qualification programme under review by NDK and equipment qualification tests are performed after the selection. However, due to the importance of ensuring that the equipment functions under severe accident conditions, the PRT recommends to NDK to thoroughly review the environmental qualification programme and the approaches proposed by the licensee.

During the Akkuyu site visit the PRT identified contradictory information on the role of the on-site fire brigade during severe accidents. While certain information provided by the licensee claimed that the on-site fire brigade will have no role in managing a severe accident (beyond ‘normal’ firefighting tasks) other documents indicated that there are severe accident management strategies in place according to which the fire brigade is expected to deliver cooling water into the primary containment through systems dedicated to this purpose.

Organisation and arrangements of the licensee to manage accidents

At the time of the Turkish National Report there was no approved Emergency Operating Procedure (EOP), Beyond Design Basis Accident Management Guide (BDBMAG), or Severe Accident Management Guidelines (SAMG) available. The first version of the EOP, BDBMAG and the SAMG documents were finalised and submitted to NDK by the time of the PRT’s site visit to Akkuyu. The documents were presented to the PRT during the site visit and the PRT was given a brief explanation on the content and structure of the EOP, BDBMAG and the SAMG by the representatives of the licensee, who showed the process and instructions that operators will have to follow through selected examples. For these examples the PRT concluded that the overall process and the entry/exit points between the documents are clear.

The National Radiation Emergency Plan (URAP) sets out the overall planning philosophy, a description of the emergency response process, the requirements for the development and updating of emergency plans and manuals, and emergency training and drills. NDK expects that the plans and procedures to be developed will reflect IAEA recommendations and the lessons learned from the Fukushima accident. According to the information provided by NDK, the following legal bases, international recommendations and best practices have been taken into consideration during the preparation of the emergency response:

- Regulation on the Management of Radiation Emergencies (Turkish)
- National Radiation Emergency Plan (Turkish)
- Regulation on Radiation Protection in Nuclear Facilities (Turkish)
- IAEA GSR Part 7 (IAEA)
- NP-001-97 section 5.5, Emergency Planning on Protection of Personnel and Population in Case of Accidents and Accident Management
- NP-005-16 Emergency response and severe accident situations.

As explained by the experts of the licensee during the PRT site visit, the operators play a limited role and have limited scope to take decisions during severe accidents, and the exact strategy to be followed shall be identified by a select group of highly trained experts who are located in the shelter. This approach is reflected in the SAMG in the sense that the set of instructions that the operator has to follow is also rather limited.

Procedures and guidelines for accident management

Based on the preliminary information presented in the NR and the discussions during the visit to NDK headquarters in Ankara, the accident strategies for the Akkuyu NPP are described in three different sets of instructions that correspond to three different levels of defence in depth:

- Emergency operating procedures (EOP) – DBC 3-4
- Beyond Design Basis Management Guide (BDBAMG) – DEC-A
- Severe Accident Management Guide (SAMG) – DEC-B

The EOP is further divided into a symptom-based set of instructions and a set of event-based instructions, which depend on factors like the recognisability of the events taking place. While this approach is in line with IAEA SSR 2/2 and has advantages (e.g. it is considered highly efficient for simple and easily recognisable scenarios, they may decrease the response time of the operators and simplify the EOP structure), there are also risk factors which have to be addressed and minimised during the development of the EOP and the operator training programme. Event-based procedures are descriptive and instruct the operator to proceed in a single series of steps without providing any contingencies to deal with additional dependent or independent failures. Such procedures focus the operator's attention on those specific parameters and controls associated with the particular event being corrected or mitigated, and generally do not direct the operator to assess the overall status of the plant by reviewing various plant parameters beyond those associated with the particular assumed event. However, a commonly recognised issue with such procedures is that if the operator misdiagnoses the event or fails to recognise certain factors affecting the boundary conditions on which the procedures were based, then there is a risk that a false set of instructions will be followed and the outcome of the event will be more severe than would have been the case without the operator's intervention. In order to avoid such situations:

- The justification of event-based instructions should be robust and the experience from simulator training should be collected and used to validate the claim of easy and early recognition of these scenarios.
- The extent to which event-based instructions can be used should be limited to a set of events with easily recognisable characteristics and minimal chances of misdiagnoses.
- The operator training programme should pay special attention to the use of these instructions, including how to recognise that the event-based instructions are invalid or no longer valid, and should switch to the symptom-based instructions as early as possible.

According to the description provided by the licensee, management of severe accidents will be based primarily on the instructions given by the group of experts located in the shelter. While this approach is justifiable, it should be stated that several severe accident management strategies and methods that were mentioned by the experts of the licensee during the site visit were not reflected in any of reviewed procedures or guides (e.g. the use of fire engines to provide cooling water to the core catcher in the long term). This additional set of alternative strategies should be documented and provided to the decision makers to ensure that information is not lost and to increase the speed and efficiency of decision-making, and should also be presented in a comprehensive manner to NDK.

During the review of the SAMG, the PRT observed that instructions for depressurisation of the containment seemed to have entry points set above the design pressure of the containment. While it is understandable that with a robust design the containment is not expected to fail immediately after reaching the design limit, the PRT believes that 'delaying' depressurisation actions until the design pressure has been exceeded may be counterproductive, and as a general rule such instructions should be set at least to the design pressure of the containment, or preferably below this value. Alternatively, there must be solid justification for other set points.

While the Level 1 PSA models were submitted to and approved by NDK, the preparation of the Level 2 PSA was delayed, and its review and approval process is ongoing. Based on the information presented in the NR and the discussions with NDK, in line with the IAEA guides SSG-3 and SSG-4, it has been stated that the results of these analyses will be utilised in the development/amendment of the listed guidelines by the designing organisation.

IAEA SRS No. 25. recommends that the regulatory body and the utility agree on the intended (and potential future) uses of the Level 2 PSA, and confirm that the proposed scope of the analysis is consistent with these uses. It is also considered good practice for the regulatory body to specify the regulatory expectations of the PSA (in terms of resolution, the accepted level of conservative assumptions, etc.) that it would expect the utility to fulfil for specific PSA applications. This can then be compared with what the utility has proposed, and an agreement reached on what would be expected if the scope of the PSA falls short. Under this approach such findings need to be brought to the attention of the utility so that the scope of the analysis can be changed at an early stage. According to the consultations with NDK, this recommendation was not followed in the previous practice/approach of the licensee and the Level 2 PSA was submitted for regulatory review at a much later stage of the project. This limited the ability of the regulatory body to review and evaluate the models based on their applicability to SAMG development purposes.

It also worth mentioning that the regulatory body currently has limited access to the PSA models of the licensee, which may reduce the efficiency of a comprehensive review and, therefore, the possibility of egulatory approval of various PSA applications. Full access to the PSA models could also help the regulatory body to develop its own risk-informed applications, therefore it is recommended to provide PSA models to the regulator.

General design approach for accident management

In general, the Akkuyu design aims to fulfil the DiD principles based on the following approach:

- For DiD Level 3a, the fulfilment of acceptance criteria is ensured via a set of active systems that have a double redundancy (LPIS, HPIS, control rods, emergency cooldown system, annulus ventilation, etc.) in combination with the first stage hydroaccumulators, which is necessary to avoid early core damage under LOCA scenarios.
- For DiD Level 3b, these active systems are replaced primarily by a set of passive safety features (2nd and 3rd stage hydroaccumulators, PHRS), in combination with or in addition to certain active SSCs (e.g. the emergency boron injection system) for specific scenarios. This aims to ensure that core damage is avoided even in the event of a station blackout, with the exception of the accumulators that operate the various valves put the passive systems into operation. Through this approach the licensee claims that there is no realistic scenario that could lead to core damage in the first 72 hours (before the 3rd stage hydroaccumulators are depleted). The design of these systems also includes a certain level of redundancy and play a crucial role in ensuring that the facility can only enter a DiD Level 4 once the fast phenomena of the transient in the initial stages is over and its effects are mitigated.
- For DiD Level 4 there are dedicated safety features but minimal redundancy, which is supplemented by a set of additional (both safety and non-safety rated) systems as alternative means for operator actions. The basic premise of the design is that, due to the active and passive safety functions and features at previous DiD levels, there is no realistic scenario which can lead to core or fuel damage in the first 72 hours after the initiating event has occurred, and after 72 hours external support will be able to reach the site and/or actions implemented on site will ensure the long-term cooling of the core/fuel. This approach involes critical inherent assumptions, based on which these conclusions can be drawn, namely:
  - The passive parts of the sprinkler system will maintain their functionality throughout the severe accident scenario and will be able to condensate the water evaporating both from the SFP and the core catcher.
  - External help and/or on-site measures will be available within 72 hours.

  Since the design solutions are based on these assumptions, they have to be justified in a robust manner to avoid invalidating the developed scenarios in the long term.

Accident management for a scenario of loss of the core cooling function

The Akkuyu NPP includes several design provisions for the management of postulated loss-of-coolant accidents and loss of off-site and on-site electrical power supply. The passive systems in particular aim to increase the time available to restore the active core cooling systems and, if it is not possible to restore cooling, to delay the progress of the accident so that no core damage is expected in the first 72 hours after the initiating event. If the accident progresses to a severe accident, the containment spray system plays a key role in pressure reduction and maintaining containment leak tightness.

The main active safety systems are the emergency core cooling system (ECCS), the emergency boron injection system, the emergency cooldown system and the containment spray system (JMN). The active safety systems consist of two physically separated trains (2 x 100%). The JMN injects water into the containment atmosphere through spray nozzles in the upper dome of the containment volume, which limits the increase of containment pressure and washes out fission products from the containment atmosphere in loss-of-coolant accidents. It is claimed, however, that the sprinkler system is not considered in DSA for design basis accidents since the pressure will be below the design pressure of containment even without the actuation of the spray. The Mediterranean Sea serves as the ultimate heat sink for the active cooling systems. The water source for the active ECCS and the boron injection system is the spent fuel pool, which is designed to contain excess water for the designated safety functions.

The passive core cooling systems in the Akkuyu NPP design are hydroaccumulators connected to the primary circuit, and the passive heat removal system connected to the secondary side of the steam generators (SG PHRS). Hydroaccumulators consist of three stages that discharge coolant to the primary circuit at specific setpoints as the primary circuit pressure decreases. Together, these three stages can provide core cooling for 72 hours after the initiating event in case of failure of the active safety systems.

The SG PHRS is capable of removing decay heat through the secondary circuit, in principle without any time limitation, provided that the natural circulation that removes heat to the ultimate heat sink (atmospheric air) can be established. The air-cooled SG PHRS is activated automatically in case of failure of the power supply system for the active cooling systems.

Electrical power for the core and spent fuel pool cooling systems is supplied by the emergency power supply system (EPS), which includes two emergency diesel generators each connected to an independent power supply train. The Akkuyu site has a set of alternative diesel generators (ADGS) that are planned to be manually connected to the two EPS trains to supply the accident management measures more than 72 hours after the initiating event.

The EPS has batteries for DC power supply with discharge times of 2 hours and 72 hours. The national stress test report (revision 2) states that the two-hour batteries provide several safety functions such as primary and secondary overpressure protection and closing of containment isolation valves, and that the two-hour discharge time will be discussed in further stages of the licensing. The accident and post-accident monitoring systems, and the valves necessary for operating the passive safety systems, are powered by the batteries designed for a discharge time of 72 hours.

If the active and passive cooling systems fail to maintain the water level in the reactor and this leads to the melting of the core, the Akkuyu NPP design includes SAM systems to mitigate the consequences of a severe accident and protect containment integrity (see next section). However, the concept seems to rely on the assumption that no melting of the core is possible within 72 hours of the initiating event (such as LOCA). This is expected to provide adequate time to make the necessary manual and mobile connections for severe accident management, e.g. putting the alternative fuel pool cooling system into use for containment cooling.

Accident management to protect the integrity of the confinement function

According to the discussions during the site visit the design pressure of the primary containment is 0.56 MPa. The AES 2006 VVER-1200 design includes provision for cooling and combustible gas removal to limit the containment pressure to an acceptable level, without structural failure in DBA and BDBAs. The containment itself is a double containment; the inner containment pre-stretched structure provides a robust confinement even with high internal pressures, while the annulus between the inner and outer containment can provide the means for collecting and filtering the radioactive releases from the inner containment. The outer containment protects the inner containment from external hazards.

According to the information available to the PRT, the spray pipelines and nozzles of the JMN containment spray system can be connected to the additional fuel pool cooling system (JMN60/61) that is otherwise independent of the components of the JMN system designed for DBA. Using the alternative SFPCS can reduce the containment pressure in a severe accident if the containment spray system is not available. JMN60/61 was not addressed in the NR (Revision 2), in which the containment spray system was indicated as the means for containment pressure management in both DBA and BDBA. The PRT understands that JMN60/61 is a later addition to the design to increase the level of independence of severe accident management from DBA, and that the SAM-dedicated operation of JMN60/61 is mainly based on non-permanent connections and mobile equipment.

It can be concluded that containment pressure reduction in a severe accident largely relies on the capacity and availability of the containment spray because there are no passive means to limit containment pressure and remove heat from the containment atmosphere. To prevent reactor pressure vessel failure at high primary circuit pressure, which is generally recognised as a potential cause of early containment failure, the primary circuit can be depressurised by opening the PRZ PORVs together with the valves of the emergency gas removal system of the primary circuit (KTP). There are no separate depressurisation pipelines and valves dedicated to severe accident management.

Passive autocatalytic recombiners and a hydrogen monitoring system are installed in the containment rooms to manage the combustible gases formed during a severe accident. The recombiners reduce the hydrogen concentration below the detonation regime. However, combustion in the deflagration regime is not ruled out since the internal structures of the containment are expected to withstand deflagrations. On the simultaneous operation of the containment spray and the possibility of high hydrogen concentration in the containment, NDK explained that hydrogen concentration is checked before the spray is activated to make sure that the spray activation is safe (as the condensation of steam can make the atmosphere less inert), however combustion in the deflagration regime is not ruled out.

A main design feature dedicated to severe accident management is the core catcher (core melt localisation system). It is designed to retain and cool the corium in a controlled manner and to prevent any interaction between the load-bearing containment structure and the corium, and to prevent the generation of flammable compounds through molten core-concrete interactions (MCCI). The core catcher vessel contains sacrificial materials that melt and mix with the corium. The chemical and physical interactions between the corium and the sacrificial materials ensure subcriticality, reduce the thermal loading on the core catcher vessel, stop the generation of hydrogen from the core materials relocated into the core catcher, and ensure the formation of an isolating crust at the top of the molten core to limit releases from the corium.

The core catcher vessel is cooled externally by boiling the water, and the generated steam flow is directed into the upper volume of the containment.

The core catcher is internally flooded by passive valves after all the corium has been relocated from the RPV to the core catcher, so that the melt is covered with water. Restoring continuous cooling is necessary for long-term accident management because the decay heat generation continues in the core catcher, and the heat must be transferred from the containment into the ultimate heat sink to stop boiling in the core catcher and reach the safe state after a severe accident. Long-term cooling of the core catcher is arranged using the additional fuel pool cooling system (JMN60/61), the containment spray system (JMN) or a mobile pump (fire engine) depending on the availability of the systems in post-accident conditions.

The core catcher does not require a power supply, but a number of valve operations may be necessary to ensure the supply of water to the core catcher (for external cooling) and to the internal volume of the core catcher to ensure adequate cooling of the vessel.

The severe accident power supply relies on the DC batteries that provide electrical power to the active components, mainly valves and the severe accident monitoring system. The ADGS can be manually connected to supply EPS trains to power the pumps of the alternative SFPCS as well as other consumers and recharge the DC batteries but, by design, the connection is not expected to be necessary before 72 hours after the initiating event.

Accident management for mitigating the consequences of loss of the containment integrity

In line with most of the design solutions for VVER-1200 reactors, the safety design of the Akkuyu NPP includes hardware provisions and operating guidelines specifically developed for the purpose of retaining containment integrity in a core melt accident. Regardless of the design provisions, certain scenarios in which loss of containment leak tightness is seen as plausible are considered below.

The containment cooling function might be lost due a component failure in the alternative SFPCS, or due to a failure to connect the non-permanent equipment of the cooling circuit or the alternative diesel generators to supply power to this system, or the loss of ultimate heat sink. Loss of cooling might result in an uncontrolled pressure increase in the containment due to steam generated in the core catcher. However, it should be noted that for new-build NPPs, the design is expected to reliably prevent the loss of containment integrity. Emergency preparedness plans are discussed in Section 7.2.1

Since the cooling of core melt in the core catcher largely relies on the boiling of water on the external surface of the core catcher, a failure to provide water from water reserves in the containment might lead to failure of the core catcher vessel. Molten core interactions with the containment load-bearing structures are possible only if the core catcher fails, or if the structures above the core catcher fail to direct the flow of molten material from the RPV into the core catcher vessel, causing a bypass (which can be considered a very unlikely scenario).

Evaluation of factors that may impede accident management

Based on the review of the NR and the consultations with NDK it can be concluded that the design is robust and ensures an adequate level of nuclear safety for most accident sequences. There are, however, issues with certain statements and inconsistencies regarding the severe accident management strategies and for some questions the PRT has received answers that seemed contradictory. Examples of such issues identified during the review are:

- External hazards that are capable of simultaneously damaging multiple (and even inherently passive) pieces of equipment may pose a significant risk to accident management strategies if these strategies rely on transporting coolant, power, etc. via the same components (e.g. pipelines). This is especially the case if these routes are supposed to be used for multiple levels of DiD. For example, it is unclear how it is ensured that the ADGs are available for power supply to the safety-critical systems under all conditions.
- Some of the safety functions for severe accident management may require connections that are not permanently installed but carried out by field operations during the accident (e.g. the containment spray utilising the SFPCS). It is unclear how it is ensured that such operations can be performed under accident conditions.
- According to the NR and the answers received from NDK the main control room may become practically inaccessible under accident conditions and no operator shifts may take place. This can lead to a strategy defined for a certain scenario being jeopardised if it requires long-term intervention by the operators.
- While the method of practical elimination is not applied (at least not directly according to the answers of the licensee), it seems that certain scenarios (such as severe accidents with core catcher bypass) are screened out based on practical elimination methodologies. Practical elimination, however, is a strict and well-defined methodology and should only be applied if all the necessary steps are properly followed and robust justifications for using it are provided.
- The separation of means and measures for various DiD levels is not strictly followed in many strategies, because of which the plant relies on the same set of systems for different DiD levels, which may compromise the strategies developed for severe accident scenarios.

Accident management for events in the spent fuel pools

The design of the Akkuyu NPP relies on the excess water in the spent fuel pool as a source of coolant and on the boron solution to supply the active safety systems responsible for residual heat removal through the primary circuit, ensuring subcriticality through boron injection into the core and by reducing containment pressure through the spray system. The active ECCS and the containment spray system are used to replace the evaporating makeup water to the spent fuel pool through the sump if emergency power supply is available. The second stage hydroaccumulators supply boron solution to make up the spent fuel pool in case of a complete loss of all AC power supply or the loss of ultimate heat sink.

There are no separate means to manage an accident in which fuel in the spent fuel pool is uncovered, although it should be noted that a high level of excess water in the pool greatly increases the time available for fuel uncovering. It should be noted that the spent fuel pool is located inside the containment, making most of the means for mitigating a severe accident in the reactor also available for SFP accidents (excluding the core catcher).

Based on the assessment it can be concluded that, similarly to the reactor core, the spent fool pool safety functions are ensured via active and passive safety features (2nd and 3rd stage hydro accumulators) during BDBA conditions. The large quantity of water in the hydroaccumulators aims to ensure that the water makeup can be achieved even in the event of a total station blackout. As regards accident management in the spent fuel pool, the approach of the licensee was summarised as follows:

> 'In the spent fuel pool, SBO and/or loss of ultimate heat sink accidents are reviewed as BDBA (common initiating event for core and spent fuel pool). Excess water in the SFP is 650 m3. Because these accidents are reviewed during all power operation states including shutdown modes, the decay heat profile and dam position may change. Decay heat is 1.088 MW for the power operation state (dam closed, just before unloading), 5.5 MW for partial refuelling (dam open), 11.41 MW for full unloading (dam open) and 21.96 MW (for emergency unloading).'

During the review of the report and the discussions with NDK the following conclusions were made:

- The strategy of mitigating design extension conditions in the SFP seems to be in conflict with the principle of defence in depth, or the description of the strategy provided by the licensee remained unclear. Based on the answers received from NDK, the licensee intends to use the same systems for more than one level of DiD, although the active components involved in performing a certain function can change (e.g. a different set of pumps providing coolant for the same pipeline sections). Based on the answers, however, there are shared active components (e.g. check valves) which may jeopardise the safety functions on multiple levels of DiD.
- NDK reviewed the scenarios with pipeline breaks (even in different locations) and liner breaks in SFP during discussions on the single failure justification criteria . NDK, however, did not review the scenario of SFP pipeline/liner breaks together with SBO or loss of ultimate heat sink. Because the pressure in the SFP is almost atmospheric, the probability of an SFP line break with SBO was considered to be very low. Furthermore, the PRT found no indication that the initiating event of a spent fuel assembly becoming stuck in the refuelling machine during transport had been considered in the design. All of these initiating events may result in a different and more severe type of scenario than the loss of cooling type of scenarios taken into consideration in the design. In the case of pipeline and liner break to ensure that the fuel assemblies are continuously covered, pose a more significant challenge due to the loss of coolant through the leakage and the evaporation also present in the previous scenarios. In the case of a fuel assembly stuck in the refuelling machine the issue is that the water level above the stuck fuel assembly is much smaller than for the assemblies safely stored on the lower compartments of the SFP, therefore the fuel will be uncovered sooner, hence giving operators less time to intervene. It should be mentioned, however, that due to the 2nd stage hydroaccumulators the water level is not supposed to drop for the first 72 hours, hence fuel damage is not supposed to occur even in the case of the scenarios described. The reason the SFP can rely on water from the 2nd stage hydroaccumulators while not jeopardising the safety of the core is that the risk of LOCA-type accidents occurring simultaneously in both the core and the SFP is extremely low, therefore while the SFP relies on the HA it can be safely assumed that it is not needed for the core cooling.

7.2.2 Margins, cliff-edge effects and areas for improvements

Strong points, good practices

The SG PHRS, which removes heat from the primary circuit through the secondary circuit without the need for external power, reduces the probability of a core melt accident resulting from total loss of off-site and on-site power and total loss of ultimate heat sink. The hydroaccumulators provide passive core cooling up to 72 hours after the initiating event even in the case of large break LOCA scenarios, allowing time to restore power supply and the active means of core cooling by operations at the NPP site. The passive core cooling systems provide a significant level of robustness against core melt accidents.

Regardless of the autonomy achieved by the passive systems, the mitigation of severe accidents on the fourth level of DiD should be considered independently. The core catcher is generally considered structurally and functionally robust for the purpose of retaining and cooling corium, and thus preventing MCCI and melt-through of the containment structures.

Passive autocatalytic recombiners are used to prevent hydrogen combustion in environments that would threaten containment integrity. Hydrogen management using PARs is already well-known technology and the performance of the PAR system depends mainly on the adequate number and location of the PARs in the containment, which should be based on a comprehensive analysis of hydrogen behaviour in the containment.

Weak points, deficiencies (areas for improvements)

SA mitigation systems at DiD Level 4 are not fully independent from systems used at other DiD levels. Some components of the JMN containment spray system are utilised in both DBA and in all BDBA to reduce pressure in the containment. The additional fuel cooling system (JMN60/61) has a dedicated pump and a heat exchanger that can supply cooling water to the containment spray system, but this connection is not permanently installed.

The heat transfer chain through the intermediate cooling circuit can also be replaced with mobile equipment. While these solutions provide alternatives for maintaining containment cooling and heat removal to the ultimate heat sink, there are no permanently installed containment heat removal systems that would be dedicated to severe accident management at DiD Level 4.

The utilisation of non-permanent systems requires operator and field actions in accident conditions at the plant site (to prepare the connections for water injection, electricity or fuel), which complicates the accident management and may be subject to human or organisational errors, as well as to difficulties in performing the necessary operations in all possible accident conditions on-site (including seismic and extreme weather events).

It is worth mentioning that the technical solution for the containment heat removal of the Akkuyu units, similar to the VVER-1200 units in the Novovoronezh II design, differs significantly from the one implemented in the Leningrad II NPP VVER-1200 units, which have a water-cooled passive containment heat removal system. Even though this design is not directly comparable to the Akkuyu design (as they have other differences), the PRT considers that a passive containment heat removal system could be an advantage in comparison to relying only on water supply through the containment spray system and the capability of the containment structures to absorb and condense part of the thermal loading.

It also should be noted that while limitation of containment pressure depends on the operation of the spray system, in certain scenarios it might be necessary to prohibit its function because this would make the atmosphere less inert and increase the risk of hydrogen detonation. It is not completely clear whether not using the spray is contradictory to the goal of limiting containment pressure increase in any stages of a severe accident. Because the details of the analyses of containment pressure behaviour were not available to the PRT, it was not possible to assess whether the limiting scenario(s) have been extensively covered in PSAR/FSAR.

The primary circuit depressurisation system that is used to prevent reactor pressure vessel failure at high primary circuit pressure is not independent of other DiD levels since the PORV pressuriser is needed for this safety function. In addition, there is no permanently installed power supply system dedicated specifically to severe accident management. The active systems requiring electrical power are supplied by the mobile ADGs which, during normal operation, are stored near their expected operating locations, and which can be connected to the EPSS within the 72-hour time limit before the depletion of the HAs.

In general, when full independence of DiD Level 4 is not achievable, the safety justification should very carefully consider which systems used in different DiD levels could be suitable for severe accident management, as some of these systems may have been lost earlier in the accident scenario (or their loss is the cause of the accident progressing to core melt).

The severe accident management concept relies strongly on the statement that the reactor core cannot melt before 72 hours has passed from the initiating event and that the long grace period facilitates even complex connection of non-permanent equipment under all accident conditions. It is not clear whether adequate consideration has been given to fast-progressing scenarios, e.g. large break LOCA or potential failure modes of the passive hydroaccumulators. There was no information available to the PRT on whether the possibility of failure of the passive systems has been taken into account in developing the SAMGs (due to, e.g. valve failure or the pipe break in the pipelines connected to the HAs).

There is no emergency feedwater system in the design, and the emergency cooldown system (which is supposed to ensure emergency heat removal via the secondary circuit) cannot replace water exiting from the steam generators to the containment or the atmosphere. This limits the possible strategies and means available to the operator (e.g. long-term heat removal through the opening and closing of the AR valves) to remove residual heat in most scenarios and force the operator/plant response to sacrifice a barrier to ensure residual heat removal via feed and bleed.

According to NDK, the environmental qualification programme will be carried out after the installation of the equipment. The PRT considers this to be a rather late stage, as the commissioning of unit 1 was already ongoing during the ENSREG mission to the Akkuyu NPP.

7.2.3 Possible measures to increase robustness

The feasibility of the manual operations planned for SAM mitigation should be assessed under all accident conditions to ensure that there is adequate time and opportunity (e.g. safe accessibility) to reliably conduct such operations. The PRT recommends replacing, where possible, the temporary connections dedicated to DiD Level 4 with permanent equipment and connections.

Adequate provisions to manage a core melt accident occurring less than 72 hours from the initiating event should be implemented, considering the loss of the passive cooling systems. Even though the failure of the passive systems is unlikely, due consideration should be given to identifying and taking appropriate measures against such accident scenarios. Particularly in a LOCA scenario (presuming no active ECCS are available), the hydroaccumulators are the only means to cool the reactor. HA failure in this case can lead to a fast uncovery of the reactor core.

7.2.4 New initiatives from operators and others, and requirements or follow-up actions from regulatory authorities: modifications, further studies, decisions regarding the operation of plants

NDK should check that the operator training programme for DBA and BDBA scenarios are in line with the results of the Level 1 and Level 2 PSA, in order to ensure that the programme focuses on the most relevant and probable scenarios that the operators may encounter.

7.3 Peer review conclusions and recommendations specific to this area

Conclusions and recommendations

Recommendation 1

The licensee should investigate the possibility of alternative long-term cooling/depressurization strategies and means for containment under severe accident conditions.

Justification: both during the review of the NR and the visits to Ankara and the Akkuyu site it was concluded by the PRT that in the long term the use of the sprinkler system (JMN10) together with the additional fuel pool cooling system (JMN60) are the only available means to ensure long-term heat removal from the corium in the core catcher and the depressurisation of the containment. While according to the design this function may be fulfilled by various set of pumps dedicated to the different DiD levels, i this is all based on the assumption that the piping and sprinkler heads (which have a double redundancy) will be available/undamaged during severe accident scenarios (e.g. after a major earthquake). In order to reduce the SA strategy's dependence on this assumption additional strategies could be identified that ensure diverse means of depressurisation and heat removal (e.g. filtered venting strategies, long-term containment cooling solutions, passive containment cooling, etc.)

Recommendation 2

The licensee should develop procedures and provide training to the on-site fire brigade, and carry out relevant joint drills to make sure that the firefighters are well informed and trained in their roles during severe accidents.

Justification: during the site visit to Akkuyu and the visit to the headquarters of the on-site fire brigade it was concluded that the firefighters are unaware of their possible roles during severe accidents (e.g. to supply water from fire trucks to the core catcher through a set of pipes dedicated to this purpose). In order to ensure that such actions will be reliably carried out during severe accidents, procedures must be established and the firefighters must be informed and trained to carry out these actions.

Recommendation 3

The Licensee should investigate the need to install additional radiation detectors in the vicinity of the facility to ensure comprehensive coverage of the area.

Justification: during the visit site visit to Akkuyu and the short visit to the radiation centre at the site it was observed that the area to the north of the nuclear power plant seems to have no gamma detectors at the moment, and it remained unclear whether it is planned to deploy additional detectors in the near future, or whether the current state of the system should be considered the final version. It was also briefly mentioned that the high number of detectors in the eastern sector is due to the typical wind direction in the area, which the PRT considered a fair point but not a reason for not having detectors in the norther sectors. To ensure that the organisations responsible for emergency response have appropriate and comprehensive information about the radiation levels around the site, the possibility of installing further radiation detectors should be investigated.

Recommendation 4

The assumed external support/intervention after 72 hours should be robustly justified (analysed, implemented and training provided) both by the licensee and the emergency response organisations in Türkiye.

Justification: the design of the facility is based on the assumption that external help will reach the site within 72 hours which, while in line with international practice, requires a robust justification in the form of detailed plans and agreements on:

- What are the types of equipment that may be needed to be delivered to the site?
- Who is responsible for delivering this equipment and by what means?
- Is this equipment actually present and available in the region, either in governmental organisations or at industrial facilities?
- Are there agreements in place between these parties and the government or the licensee to provide this equipment in case of an emergency?

Since the 72 hours can be considered critical for the facility (e.g. this is the depletion time of the 3rd stage hydroaccumulators), these issues have to be resolved in advance and generic statements about external support may not be considered sufficient.

Recommendation 5

It is recommended that the assessment of the habitability of the ECR and accessibility of control points is performed and submitted to NDK for regulatory review, including a radiological assessment.

(Ref.: T3-8)

Justification: the basic premise on the use of ECR is that it can be safely approached even in severe accident conditions and its conditions allow long-term occupation of the room by the operator's personnel.

Recommendation 8

NDK should thoroughly review the general approach applied for the practical elimination of scenarios resulting in unacceptable consequences.

(Ref.: T3-12)

Justification: the principle of the practical elimination of scenarios resulting in unacceptable consequences is an internationally accepted approach that ensures that within the design envelope no unacceptable consequence (e.g. large-scale or early release) can occur. As a basic design principle in the nuclear industry, its application should be thoroughly reviewed by the regulator.

Recommendation 9

The issue of the primary circuit depressurisation by the means dedicated to DiD Level 4 must be carefully analysed, and, if relevant, the appropriate measures must be developed and implemented.

(Ref.: T3-18)

Justification: in the event that cooling of the reactor is lost, depressurisation is performed by the joint operation of PORV and the emergency gas removal system. As both systems are needed, no diversity is ensured for the depressurisation of the reactor in case of loss of cooling capability. In addition, these systems are not dedicated to this function (and this DiD level). This lack of diversity seems to weaken the practical elimination of a high-pressure ejection of the core.

Identified good practices

Good practice 1

The designer has included dedicated coolant injection routes that can be used during severe accidents as an alternative means to ensure residual heat removal by connecting fire engines to the junction located on the outside of the containment. This means that such aspects were proactively taken into consideration by the designer regardless of how small the possibility of a severe accident scenario was assumed.

Good practice 2

According to the licensee the simulator centre is capable of simulating severe accident scenarios, which was considered as a good practice by the PRT that ensures a greater reliability of operator actions and interventions by other actors during severe accident sequences.

8 MAIN CONCLUSIONS OF THE PEER REVIEW TEAM

Introduction

Türkiye confirmed its willingness to voluntarily undertake a comprehensive risk and safety assessment in accordance with the stress tests specifications agreed by the European Commission and the European Nuclear Safety Regulators Group (ENSREG) on 24 May 2011. Türkiye submitted its second version of the national stress tests report on the Akkuyu Nuclear Power Plant to the Directorate-General for Energy of the European Commission and ENSREG for peer review in 2019A two-stage approach was agreed between ENSREG and Türkiye for the conduct of the peer review. In the first stage, a visit to NDK in Ankara took place in May 2022. This stage was preceded by the PRT's review of the Turkish National Report. In the second stage, a visit to the Akkuyu NPP took place in May 2024, after which the peer review report was finalised. During the two visits, in-depth discussions were held between the PRT, NDK and representatives of the Akkuyu NPP to address the questions and findings of the PRT. During the Akkuyu site visit, there were a number of side visits to see locations, systems, structures and components relevant to the peer review.

The peer review team's general comments on the Turkish National Report

The stress test in Türkiye is the second stress test undertaken on NPPs under construction (the first one took place in Belarus). Stress tests conducted in the EU and non-EU countries have been on pre-existing reactor designs that were already operational. It is acknowledged that conducting stress tests on a NPP under construction sets limitations on the availability of information and the opportunities to examine physical systems, structures and components, compared to operational NPPs.

In the opinion of the PRT the Turkish National Report was drafted in accordance with the requirements of the EU stress tests. The Turkish counterparts answered all questions and provided answers during the peer review process to the extent possible. Due to information security reasons parts of the information were presented to the PRT only during the visit to Ankara in May 2022 and during the site visit to the Akkuyu NPP in May 2024.

Topic 1: ASSESSMENT RELATIVE TO EARTHQUAKES, FLOODING AND OTHER EXTREME WEATHER CONDITIONS

Earthquake

The PRT notes that the seismic risk at the Akkuyu NPP has been analysed through various analyses and assessments. According to the probabilistic safety assessment (PSA) results, the contribution of seismic risk to the fuel damage frequency (FDF) is between 70 and 80% of the total FDF, which demonstrates the significance of seismic safety. The Akkuyu NPP is designed to withstand a design basis earthquake (DBE, but also termed SSE) with a return period of 10 000 years and a peak horizontal ground acceleration of $PGA_{SSE\text{-}h}$ = 0.388 g. The DBE is derived from a PSHA which corresponds to the current state of the science on the subject. Both, the definition of the DBE and the seismic hazard assessment are in line with national and WENRA requirements. The PRT concludes that the licensee and the regulator have diligently addressed seismic hazards and made other related assessments, and the PRT specifically assesses the performed PSHA as good practice.

The PRT recommended addressing several areas to ensure and enhance the seismic safety of the Akkuyu NPP. A complete list of recommendations is provided in Chapter 5.1.4 and in the Annex. The PRT would like to highlight the following areas of concerns as being of special importance:

- The concept of 'practical elimination' has not been applied in developing a position on whether a 40% seismic margin is sufficient to demonstrate the practical elimination of accidents leading to large or early releases.
- The high hazard posed by seismogenic sources at distances of up to 50 km from the site have not been carefully assessed using the data from the seismic monitoring.
- The determination of seismic design parameters and seismic margins of the systems, structures and components is inconsistent with EU and international practice and has not been made conservatively.

Some of the systems, components and structures do not fully meet the design basis and seismic margins criteria.

Flooding and extreme weather

The Akkuyu NPP is situated on a small bay which is open on its south-west side to the Mediterranean Sea. A tsunami is considered to be the most significant potential flooding risk at the Akkuyu NPP site. The assessment of tsunami hazards at the site has been made according to several steps. The PRT concludes that a sound and formalised methodology has been applied to screen and characterise external flooding hazards.

The PRT has identified several areas which it recommended should be addressed to ensure and improve flooding safety at the Akkuyu NPP site. A complete list of recommendations is provided in Chapter 5.1.4 and in the Annex. The PRT would like to highlight the following areas of concerns as being especially important:

- The methods used for defining the design basis tsunami and the calculated design basis flood height are not in line with international practice and may contain unrealistic parameter values, leading to unconservative design basis flood height.
- The margins and measures in place seem to be insufficient to cope with external flooding above the site platform level (+10.50 m) and to prevent cliff edge effects.

The PRT did not identify any major concerns regarding extreme weather conditions.

Topic 2: ASSESSMENT RELATIVE TO LOSS OF ELECTRICAL POWER AND LOSS OF ULTIMATE HEAT SINK

The PRT notes that the Akkuyu NPP safety systems for cooling of the core and spent fuel pool consist of both active and passive systems. Both in the case of a loss of electrical power (station blackout, or SBO) and loss of ultimate heat sink (LUHS), the heat removal from the reactor core is maintained by passive means by the steam generator passive heat removal system (SG PHRS). The SG PHRS ensures adequate heat removal from the core for 72 hours. For spent fuel pool cooling, under SBO and LUHS conditions, of the fuel assemblies is maintained by boiling the water in the spent fuel pool with additional water reservoirs provided by the hydro accumulators.

The PRT identified areas which it recommends should be addressed to ensure and enhance safety at the Akkuyu NPP site with regard to SBO and LUHS. A complete list of recommendations is provided in Chapter 6.1.4 and in the Annex. The PRT would like to highlight the following areas of concern:

- The availability and operability of the active safety trains during operation should be ensured by regulations, requirements set out in the operating limits and by conditions supported by risk analyses.
- The reliability of the check valve that provides for passive operation of the HA-2 hydro accumulators should be ensured.

- Adequacy of the confirmatory demonstration and analysis of the performance of the passive heat removal system and the active core cooling system in different accident scenarios should be ensured.

Topic 3: ASSESSMENT RELATIVE TO SEVERE ACCIDENT MANAGEMENT

The PRT notes that the design of the Akkuyu NPP utilises a variety of passive measures and features (e.g. large volumes of water n hydroaccumulators and passive residual heat removal system) to ensure that there is no realistic scenario within the first 72 hours that could lead to severe accidents. In case of severe accident, safety features (e.g. core catcher) and guidelines are in place to ensure protection of the containment. Due to the timing of the peer review and the site visit to the Akkuyu NPP, the PRT was not able to fully review the relevant emergency operating procedures, beyond design basis accident management guidelines and severe accident management guidelines (documentation on these was submitted to NDK).

The PRT identified areas which are recommended to be addressed to ensure and enhance severe accident management at the Akkuyu NPP. A complete list of recommendations is provided in Chapter 7.3 and in the Annex. The PRT would like to highlight the following areas of concern:

- Cooling and depressurisation of the containment in case of a severe accident relies only on the containment spray system.
- The lack of regulatory criteria for the practical elimination of accidents leading to large or early releases, and the lack of a review of the application of the practical elimination concept in the design.
- The use of safety systems/safety features (e.g. containment spray, primary system depressurisation) at more than one level of the defence in depth.
- External support needed at the Akkuyu NPP after 72 hours from the accident has not been fully planned and agreed between the Akkuyu NPP and relevant external organisations that would provide the support.

Future outlook

The Akkuyu NPP is still under construction. Therefore, the PRT considers that the results of the stress test provide a timely opportunity to enhance plant safety by improving certain systems, components and structures to increase the plant's safety and robustness, thereby aiming to eliminate the possibility of large or early releases. The PRT recommends that NDK identify the necessary safety improvements in response to the recommendations made by the PRT in this report and to the recommendations made by NDK itself, and incorporate them into a national action plan (NAcP) containing all the relevant safety improvement measures and associated implementation schedules. It should also include, as appropriate, recommendations and suggestions from the review of the European stress tests. The NAcP should ensure timely implementation of the safety improvement measures in accordance with their safety significance. In consideration of the practice adopted by the EU Member States, the PRT further recommends that the NAcP be subject to a future review. The approach to a meaningful review should be determined by NDK.

LIST OF ACRONYMS

AC	Alternating current
ADGS	Alternative diesel generator set
AOO	Anticipated operational occurence
APC	Akkuyu Project Company
ASCE	American Society of Civil Engineers
BCC	Backup crisis centre
BCP	Backup control panel
BDB	Beyond design basis
BDBA	Beyond design basis accident
BDBAMG	Beyond design basis accident management guideline
BZOV	Demineralised water tank
CD	Core damage
CDF	Core damage frequency
CDFM	Conservative deterministic failure margin
CSNO	Coolant system of normal operation
CSS	Containment spray system
DAR	Additional emergency cooling system
DB	Design basis
DBA	Design basis accident
DBE	Design basis earthquake
DBF	Design basis floods
DC	Direct current
DEC	Design extension conditions
DG	Diesel generator
DG ENERGY	European Commission Directorate-General for Energy
DGS	Diesel generator station
DiD	Defence in depth
DOE	Department of Energy
DSHA	Deterministic seismic hazard analysis
DTHA	Deterministic tsunami hazard assessment
EC	European Commission
ECCS	Emergency core cooling system
ECR	Emergency control room
EDG	Emergency diesel generator
ENSREG	European Nuclear Safety Regulators Group
EOP	Emergency operating procedure
EPSS	Emergency power supply system
ERG	Emergency response guidelines
ERT	Emergency response team
ESFAS	Engineered safety feature actuation system
ESWS	Essential service water system
EU	European Union
FSA	Fault sequence analysis
g	standard value of the gravitational acceleration (9,81 m/s2)
HCLPF	High confidence low probability of failure
HPP	Hydroelectric power plant
I&C	Instrumentation and control
IAEA	International Atomic Energy Agency
IEP	Internal emergency plan
INSAG	International Nuclear Safety Advisory Group
INSC	Instrument for Nuclear Safety Cooperation

IPE	Institute of the Physics of the Earth, Moscow
ISFSI	Independent spent fuel storage installation
JNA	Emergency and planned primary circuit cool-down and spent fuel pool cooling system
JMN	Containment spray and spent fuel pool cooling system
JSC	Joint stock company
LOCA	Loss-of-coolant accident
LOOP	Loss of off-site power
LTE	Lifetime Extension
LTO	Long-term operation
MCC	Main Crisis centre
MCP	Main circulation pump
MCR	Main control room
MRZ	Russian abbreviation for the maximum design earthquake (~ Safe Shutdown)
MSK	Medvedev–Sponheuer–Karnik
NAcP	National action plan
NPP	Nuclear power plant
NR	(Stress test) national report
OLC	Operational limits and conditions
PAMS	Post accident monitoring system
PD	Plant damage
PGA	Peak round acceleration
PHRS	Passive heat removal system
PMSS	Probable maximum storm surge
PNAE	Russian nuclear standard
PMP	Probable maximum precipitation
PR	Peer review
PRT	Peer review team
PSA	Probabilistic safety assessment (also known as PRA)
PSAR	Preliminary safety analysis report
PSHA	Probabilistic seismic hazard analysis
PSR	Periodic safety review
PTHA	Probabilistic tsunami hazard assessment
PWR	Pressurised water reactor
PZ	Russian abbreviation for design earthquake
RCC	Regional crisis centre
RCPS	Reactor coolant pump set
RLE	Review level earthquake
RPV	Reactor pressure vessel
SAM	Severe accident management
SAMG	Severe accident management guidelines
SAR	Safety analysis report
SBO	Station blackout
SEC	Second emergency cooling
SEL	Selected equipment list
SF	Safety Fundamentals
SFP	Spent fuel pool
SG	Steam generator
SHA	Seismic hazard assessment
SMA	Seismic margin assessment
SNiP	Russian civil code
SoV	Separation of variables
SPR	Site parameter report

SPSA	Seismic probabilistic risk assessment
SSC	Structures, systems and components
SSE	Safe shutdown earthquake
SSEL	Safe shutdown equipment list
SSR	Specific safety requirements
ST	Stress test
TAEK	Turkish Atomic Energy Authority
TH	Turbine hall
TSO	Technical support organisation
UHS	Ultimate heat sink
US	United States
US NRC	United States Nuclear Regulatory Commission
VSN	Temporary Russian civil code
VVER	Water-water reactor
WANO	World Association of Nuclear Operators
WANO MC	World Association of Nuclear Operators Moscow Centre
WENRA	Western European Nuclear Regulators Association
WOG ERG	Westinghouse Owners Group Emergency Response Guidelines

APPENDIX: TABLE OF FINDINGS

Topic 1: Natural hazards	
Finding nr	**Peer review team finding**
T1-00	**Introductory remarks** Different from the past stress test peer reviews in the EU, Taiwan and Armenia, the plant under review is a new NPP. This fact has important implications for the review process, as the safety expectations for new NPPs are generally higher than those for existing plants. The latest ENSREG stress tests for the new Belarusian NPP in Astravets accounted for this fact by considering Europe and WENRA's safety expectations for new reactors. The same benchmarks are being applied in the stress tests on the new Turkish Akkuyu NPP. In line with the EU's Nuclear Safety Directive (Directive 2014/87/Euratom), WENRA (2013[69]) stipulates that for new NPPs '*accidents with core melt which would lead to early or large releases have to be practically eliminated*'. The Vienna Declaration on Nuclear Safety[70] formulates the same objective for new NPPs, although it does not refer to the notion of practical elimination (Principle 1 of the Declaration[71]). This Declaration was adopted by Türkiye and Russia, among the other contracting parties. In this context, the possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if they can be considered with a high degree of confidence to be extremely unlikely to arise (WENRA, 2019[72]).

[69] WENRA, 2013. Report on safety of new NPP designs. http://www.wenra.org/publications/.

[70] IAEA Vienna Declaration on Nuclear Safety, INFCIRC/872, 18 February 2015.

[71] *'New nuclear power plants are to be designed, sited, and constructed, consistent with the objective of preventing accidents in the commissioning and operation and, should an accident occur, mitigating possible releases of radionuclides causing long-term off site contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions.'*

[72] WENRA, 2019. Report Practical Elimination Applied to New NPP Designs - Key Elements and Expectations. http://www.wenra.org/publications/.

	IAEA provides literally the same definition. No consensual definition of the term '*extremely unlikely*' exists so far. However, European convergence can be observed towards a probabilistic target value in the order of 10^{-6} (or lower[73,74]) for the early or large release frequency by '*extreme unlikeliness*'. As regards the practical elimination of early or large releases, WENRA further specifies: '*For that reason, rare and severe external hazards, which may be additional to the general design basis, unless screened out (…), need to be taken into account in the overall safety analysis.*' They also say '*Rare and severe external hazards are additional to the general design basis, and represent more challenging or less frequent events. This is a similar situation to that between Design Basis Conditions (DBC) and Design Extension Conditions (DEC); they need to be considered in the design but the analysis could be realistic rather than conservative.*' These safety expectations require a broader and more extensive consideration of external hazards in plant design and the consideration of events with occurrence probabilities well below 10^{-4} per year in the safety demonstration. For the stress tests, this means that natural events with occurrence probabilities well below the design basis value (10^{-4} per year) need to be considered, to evaluate if sufficient protection is in place to practically eliminate early or large releases. Taking a probabilistic target value of 10^{-6} per year for the practical elimination of early or large releases requires us to demonstrate that, for different hazard types, events with occurrence probabilities in the range of 10^{-7} per year (e.g. earthquake, tsunami) do not lead to early or large releases. This is because the total early or large release frequency is the sum of all hazard contributions (e.g. earthquake, tsunami, internal hazards etc.)[75]. Taking the WENRA's stipulation on the practical elimination of early or large releases caused by external events as a benchmark for the Akkuyu stress tests therefore requires us to • review hazard values down to extremely low occurrence probabilities

[73] E.g. Sweden: '*Extremely unlikely'* has been interpreted to indicate a limit between 10^{-6} and 10^{-7} per year.

[74] *'[One] objective for … future plants is the practical elimination of accident sequences that could lead to large early radioactive releases.'* The objective for large off-site releases requiring a short-term off-site response is 10^{-5} per reactor-year for *existing* plants, indicating that lower probabilities are expected for new NPPs (IAEA INSAG-12, 1999).

[75] IAEA DS508 (Draft), 2021, p. 28: ''*When it is claimed that a particular accident condition has been practically eliminated on the basis of probabilistic arguments, it needs to be taken into account that the cumulative contribution of all the different cases must not exceed the target for large or early release frequency where such a target has been established by the regulatory body.*'

	• identify SSCs necessary to prevent early or large releases (i.e. SSCs ensuring the confinement function) • assess the robustness and margins of the identified SSCs • conclude whether SSCs are sufficiently robust to prevent early or large releases in the event of external events with severities that correspond to occurrence probabilities in the range of 10^{-6} to 10^{-7} per year. Relevant background is found in the documents cited below. As regards the described considerations and cited documents the PRT will apply the following benchmarks for its review of the Akkuyu NPP Project: **(a) Core damage frequency (CDF): 10^{-5} per reactor year** **(b) Large or early release frequency (LERF): 10^{-7} per reactor year** The PRT regards its approach as being in line with current practice of the VVER Working Group states. 'Comparative summary of main findings/Severe Accident Acceptance Criteria' of the OECD report TR-VVERWG-01 (2017) of the VVER Working Group states: *'In all member countries criteria for cumulative CDF is equal 10^{-5}/year. Large and early release should be practically eliminated.'* *'In India, Russian Federation, and Türkiye the frequency of large radioactive release should not exceed 10^{-7}/year.'* In Appendix D of the OECD document 'Questionnaire on Severe Accident Managements and Practices – Regulatory approaches used in Türkiye', section 3. 'What acceptance criteria are applicable for severe accident' the Turkish National Regulator (NDK, Nükleer Düzenleme Kurumu) replies by quoting of the Russian Standards: *'RF NP-082-07 Nuclear Safety Rules for Reactor Installations of Nuclear Power Plants* *2.1.9. During RI design process one shall pursue that the cumulative frequency of severe beyond design basis accidents estimated on the basis of the probabilistic safety assessment would not exceed 10^{-5} per reactor year.* *RF OPB-88/97, NP-001-97 General Regulations on Ensuring Safety of Nuclear Power Plants* *1.2.17. To avoid the necessity of evacuating population beyond the area covered by planning protective measures established according to regulatory requirements to NPP siting efforts should be made to ensure that, the estimated probability rate of limiting emergency release did not exceed 10^{-7} per reactor year.'* **References:**

	WENRA, 2013. Report – Safety of new NPP designs. http://www.wenra.org/publications/ WENRA, 2019. Report – Practical Elimination Applied to New NPP Designs - Key Elements and Expectations. http://www.wenra.org/publications/ IAEA, 1999. Basic Safety Principles for Nuclear Power Plants, 75-INSAG-3 Rev. 1, INSAG-12 IAEA, 2016. Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants. IAEA-TECDOC-1791. IAEA, 2019. Hierarchical Structure of Safety Goals for Nuclear Installations. IAEA-TECDOC 1874. Multinational Design Evaluation Programme (MDEP), Technical Report TR-VVERWG-01 (Public Use), VVER Working Group, Regulatory approaches and criteria used in severe accident analyses and severe accident management, Version 1.1, 20 November 2017, Gosatomnadzor of Russia, 1997. OPB-88/97, NP-001-97 'General Regulations on Ensuring Safety of Nuclear Power Plants', 14 November 1997 Gosatomnadzor of Russia, 2007. NP-082-07 'Nuclear Safety Rules for Reactor Facilities of Nuclear Power Plants', 10 December 2007
T1-01	**Seismic hazard: regulatory basis** ***Regulatory requirements for the design basis earthquake (DBE or SL-2):*** Cited in subchapters 2.1.1 and 2.1.3 of the NR, the seismic design of Akkuyu NPP buildings and structures is developed in accordance with the requirements of the Russian standards NP-031-01[76], MP 1.5.2.05.999.0027-2011[77] and MR 1.5.2.05.999.0025-2011[78]. These documents were included in the licensing basis for the Akkuyu NPP design[79]. An analysis of the licensing basis for compliance with IAEA and Turkish regulations on NPP sites was done as a part of the licensing process. The NR also states that the analysis demonstrated that the application of Russian regulations ensures acceptable compliance with the national requirements and IAEA Safety Standards, including seismic safety standards.

[76] NP-031-01: Standards for Design of Seismic Resistant Nuclear Power Plant. Russian Federation, 2002.

[77] MP 1.5.2.05.999.0027-2011: Seismic Design Standards for Nuclear Power Plants. Guidelines. Russian Federation, 2011.

[78] MR 1.5.2.05.999.0025-2011: Seismic Analysis and Design of Nuclear Power Plants. Guidelines. Russian Federation, 2011.

[79] List of Licensing Basis for Akkuyu Nuclear Power Plant. TAEK Atomic Energy Commission, 2014.

	The Russian standard NP-031-01 has been developed to take into account the recommendations of the superseded IAEA guidelines 50-SG-D15[80] and 50-SG-S1 (Rev. 1)[81]. NP-031-01 defines two levels of earthquake: 1. the **safe shutdown earthquake** (SSE), an event with a 10 000-year return period (0.5 % exceedance probability in 50 years) and 2. the **operating basis earthquake** (OBE), an event with a 1 000-year return period (5 % exceedance probability in 50 years). NP-031-01 also requires the minimum value of SSE at least 0.10 g and the minimum value of OBE at least 0.05 g, regardless of the site seismicity. The SSE and OBE defined in NP-031-01 comply with international practice and IAEA Safety Standards. According to IAEA SSG-6782 (which supersedes IAEA NS-G-1.683), two levels of seismic vibratory ground motion hazard – SL-1 and SL-2 – should be defined as the design basis earthquake for each nuclear installation.[84] If a probabilistic approach is used for the seismic hazard assessment, SL-2 should correspond to a level with an annual frequency of exceedance in the range of 10^{-3} to 10^{-5} (mean values) and SL-1 should correspond to a level with an annual frequency of exceedance in the range of 10^{-2} to 10^{-3} (mean values). The minimum level for SL-2 should correspond to a peak ground acceleration of 0.1 g at the free field or foundation level and should be not less than the values established by the national seismic codes for conventional installations. The Turkish regulation[85] cited in the NR defines two levels of design basis: S1 for Operational Basis Earthquake (OBE) and S2 for safe shutdown earthquakes (SSE). The NR states that the Turkish regulation requires S1 to be determined as a minimum of half of S2. The minimum acceptable level for S2 is 0.15 g. The higher minimum value for S2 should confirm the strictness of the Turkish national requirements for seismic safety.

[80] IAEA 50-SG-D15: Seismic Design and Qualification for Nuclear Power Plants. IAEA, 1992 (outdated).

[81] IAEA 50-SG-S1 (Rev. 1): Earthquakes and Associated Topics in Relation to Nuclear Power Plant Siting. IAEA, 1991 (outdated).

[82] IAEA SSG-67: Seismic Design for Nuclear Installations. Specific Safety Guide. IAEA, 2021.

[83] IAEA NS-G-1.6: Seismic Design and Qualification for Nuclear Power Plants. IAEA, 2003 (outdated).

[84] IAEA SSG-67 notes that SL-2 (SSE) should be designed to perform the safety function of structures, systems and components (SSCs) during and/or after the occurrence of a seismic event of such intensity. SL-1 (OBE) corresponds to a less severe earthquake than SL-2 and is related to the operational requirements of SSCs.

[85] In the NR cited as: Regulation on Nuclear Power Plant Sites, 2009.

	The minimum value of S2 is higher than the minimum value of SSE in the Russian standard NP-031-01 and the minimum value of SL-2 in IAEA SSG-67. However, the definitions of S1 and S2 levels cited in NR (pages 50-51) do not include the levels of annual frequency of exceedance. Regarding seismic site response analysis, the NR refers to independent Probabilistic Seismic Hazard Analysis (PSHA) and Deterministic Seismic Hazard Analysis (DSHA) studies. Generalized three-component accelerograms were produced for SSE and OBE levels compatible with the initial response spectra that meet the requirements of Russian regulations NP-031-01, NP-006-98[86]. The NR states that these accelerograms do not contradict the requirements of the Turkish national regulation and IAEA recommendations in SSG-9 (Rev. 1)[87]. Besides this conclusion, the NR has not cited many particular national or international regulatory documents that were followed during seismic investigations in the Akkuyu NPP region, area and site, or which were applied in the PSHA and DSHA site-specific studies. The NR states that the Vs_{30} value of 1138 m/s is assumed as the data for the entire site, to which the initial response spectra are referenced. No final site response analysis or regulatory document on this issue is further mentioned. Because Vs_{30} value at the Akkuyu NPP site is larger than 1100 m/s, according to IAEA SSG-67 this site is considered to be a rock site and a site response analysis is not necessary if it can be demonstrated that modifying the control point of seismic motion has a negligible effect. The seismic design of the seismic category I buildings and structures of the Akkuyu NPP follows documents ASCE 4-98[88] and ASCE 43-05[89]. ***Safety expectations for fuel damage frequency and large release frequency:*** During the Phase 1 country visit NDK informed about numerical values accepted by the regulator for fuel damage frequency (FDF, considering fuel damage in the reactor core, the spent fuel pool and the fresh fuel storage) and the large release frequency (LRF). These values were stated with FDF < 10^{-5} and LRF < 10^{-7} per year. The value FDF < 10^{-5} per year is an acceptance criterion stipulated in the licensing basis and license conditions. LRF < 10^{-7} per year was to be understood as a target value specified in the license conditions, similar to Russian regulations. NDK further explained that Russian regulations do not identify *'early'* releases.

[86] NP-006-098: Requirements to Contents of Safety Analysis Report of Nuclear Power Plant with VVER Reactors. Russian Federation, 2003.

[87] IAEA SSG-9 (Rev. 1): Seismic Hazards in Site Evaluation for Nuclear Installations. Specific Safety Guide. IAEA, 2022.

[88] ASCE 4-98, Seismic Analysis of Safety-Related Nuclear Structures and Commentary, American Society of Civil Engineers, 1998.

[89] ASCE 43-05, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities American Society of Civil Engineers, 2005.

	Assessment Regulatory requirements applied to the definition of the SL-2 seismic vibratory ground motion hazard are in line with WENRA requirements (WENRA, 2020[90]) and IAEA guidance[14] for defining the design basis earthquake (DBE). The acceptance criterion for fuel damage frequency (FDF) < 10^{-5} per year is in line with the safety expectations for new NPPs in European countries (nine WENRA countries require 10^{-5}, three countries require values between 10^{-5} and 10^{-6}.). Restricting the safety expectations for radiological releases to large releases (LRF) is different to regulations in European countries, the EU Nuclear Safety Directive (Directive 2014/87/Euratom) and the Vienna Declaration on Nuclear Safety that refer to large <u>or</u> early releases (LERF). Values accepted for LERF for new NPPs in WENRA countries range from 10^{-6} (five countries) to 10^{-7} per year (five countries including countries with significant seismic hazard). During the desktop review NDK confirmed FDF < 10^{-5} per year as an <u>acceptance criterion</u> but noted that it considers LERF/LRF to be a <u>safety target</u> '*in case of the designer can justify that its design has all possible and practical design improvement/solutions against these external hazards.*' Turkish regulations stipulate that '*Design measures are taken to prevent severe accidents which can lead to early loss of containment as much as possible. It must be shown that probability of severe accidents that may lead to early loss of the containment is very unlikely*'. PRT welcomes the fact that NDK has discussed the large and early releases definitions with the Akkuyu NPP, and that the licensee updated the PSA in accordance with this concept, consistently with criteria in Turkish regulations. **Recommendations** **T1-01-1. High priority.** The PRT recommends NDK formulating acceptance criteria (rather than *'targets'*) for the occurrence probability of large releases (LRF[91]) and also include safety expectations for early releases (LERF[92]) following the safety objectives for new NPPs in the WENRA countries (see T1-00, Introductory remarks).

[90] WENRA (2021), Report WENRA Safety Reference Levels for Existing Reactors 2020.

[91] Large release: situations that would require protective measures for the public that could not be limited in area or time (WENRA, 2013. Report – Safety of new NPP designs); A release of radioactive material for which off-site protective actions that are limited in terms of times and areas of application are insufficient for protecting people and the environment (IAEA Safety Glossary 2021).

[92] Early release: situations that would require off-site emergency measures but with insufficient time to implement them (WENRA, 2013. Report – Safety of new NPP designs); A release of radioactive material for which off-site protective actions are necessary but are unlikely to be fully effective in due time (IAEA Safety Glossary 2021).

T1-02	**Seismic hazard: design basis earthquake (DBE)** **Description:** Design parameters for the design basis earthquake (DBE; termed SSE = safe shutdown earthquake in the NR) were determined by probabilistic and deterministic approaches. SSCs of the Akkuyu NPP are designed based on the following peak ground accelerations for the SSE (10 000-year return period; NR, p. 48): SSE peak horizontal ground acceleration $\mathbf{PGA_{SSE\text{-}h} = 0.388\ g}$ SSE peak vertical ground acceleration $\mathbf{PGA_{SSE\text{-}v} = 0.295\ g}$ Seismic category 1 equipment is additionally tested for earthquakes with ground motion parameters 40% higher than the SSE. The 1.4×SSE seismic analysis is made using realistic approaches. Ground motion of the operating basis earthquake (OBE) was fixed at 50% of the SSE, which is equal to $PGA_{OBE\text{-}h}$ = 0.194 g and $PGA_{OBE\text{-}v}$ = 0.147 g. During the peer review, NDK explained that the DBE is based on the Seismic Hazard Assessment Report by ENVY/ KOERI (2013)[93]. The report combines PSHA and DSHA studies by the Kandilli Observatory and Earthquake Research Institute KOERI (under contract to ENVY Inc.), Paul C. Rizzo Associates, Inc. (PCR) and WorleyParsons Nuclear Services JSC (WP). PSA is said to follow IAEA SSG-9 using a logic tree approach, with four source zone models and eight GMPEs for active shallow regions and subduction zones (ENVY/ KOERI 2013; WP 2012[94], PCR 2012[95]). ENVY/ KOERI 2013 was reviewed by the Institute of Physics of the Earth. Hazard calculation used EZ-FRISK™ software. An additional SHA by study WorleyParsons (2014)[96], based on revised seismic source characterisations and SHA methodology was not considered for reasons and justifications provided in AKU-BDD0132, Rev. B04, 20.01.2017. According to NDK, the DBE spectra of ENVY/ KOERI (2013) and WorleyParsons (2014) are almost identical, with the ENVY/ KOERI (2013) spectra being slightly on the conservative side.

[93] ENVY/ KOERI (2013), Seismic Hazard Assessment of the Akkuyu NPP Site (Rev.03), March 2013.

[94] WP (2012), Independent Review of the Seismic Hazard Assessment of Akkuyu NPP Site. Interim PSHA Report, WorleyParsons Nuclear Services JSC, 23 April 2012.

[95] PCR (2012), Interim Probabilistic Seismic Hazard Analysis Report: Akkuyu Nuclear Power Plant - Draft, Prepared by Paul C. Rizzo Associates, Inc., 7 May 2012.

[96] WP (2014), Standalone Seismic Hazard Assessment Report. TNPP-00-SV-REP-EN-0122-R1. WorleyParsons, 29 April 2014.

	In 2023 NDK stated that the PSHA by ENVY/KOERI (2013) was used by the designer as the basis for their preliminary design and the resulting values for the SSE were adopted as the DBE. **Assessment** Defining the DBE parameters for the occurrence probability of 10^{-4} per year is in line with WENRA (2021) requirements and international practice. The deterministic and probabilistic seismic hazard assessments that led to the determination of the DBE was explained during the desktop review and the country visit. Calculating PSHA on input data (seismic zones, source parameters, GMPEs) and models prepared by four different expert groups is regarded good practice. The hazard results for the DBE (**$PGA_{SSE\text{-}h}$ = 0.388 g**) differ from the PSHA 2017 results (Site Parameters Report). The latter states a hazard value of **$PGA_{10\text{-}4\text{-}h}$ = 0.359 g** (10^{-4} per year, mean value, 5% damping) (see T1-03). **Questions and requested documents:** Please provide more details on the stated difference of 10% between the results of DSHA and PSHA and a plausible explanation for this small difference. The differences between the PSHAs 2013 and 2017 and the causes for performing a new PSHA after defining the DBE needs further explanation (compare T1-03). The issue is particularly important for evaluating ground motion hazards for occurrence probabilities <10^{-4} per year and assessing the adequacy of seismic margins.
T1-03	**Seismic Hazard: PSHA 2017 (Site Parameters Report)** **Description:** The Site Parameters Report (AKU-BDD0132 Rev. B04 20.01.2017, p. 6.6-1 to 6.6-176) includes a detailed description of a PSHA (here referred to as PSHA 2017) completed after the definition of the DBE, which is based on the PSHA by ENVY/KOERI (2013; see T1-02). The Site Parameters Report includes detailed information on source zone models, faults considered, associated Mmax and GR parameter, GMPEs, logic tree weights etc. PSHA considers: (a) five source zone models KOERI 1 (Mmax=6.5-6.9 for source zone including NPP), KOERI 1 = METU/EERC (Mmax = 6.0 - 6.4; Mmax assumed lower because the Ecemiş fault with Mmax = 6.6 - 8.0 is confined in its own source zone), Rizzo 1 (Mmax = 6.2), Rizzo 2 (Mmax = 6.2 plus Ecemit fault source zone with Mmax = 7.5), WorleyParsons (Mmax = 6.5); (b) 3 different GMPEs for subduction zone ground motion models (e.g. Cyprus) and 5 GMPEs for modelling shallow crustal seismicity; (c) 3 models for site conditions described by different values of V_{s30} (ca. 960 – 1300 m/s). The minimum magnitude considered in the PSHA is M_{min} = 3.5.

The NPP site is included in a background seismicity zone. PSHA assumes that no seismicity can originate from the area within a radius of 5 km around the site. It was explained that this assumption was justified by site investigations that included the trenching of faults at the site and the analysis of offshore seismic data (see finding T1-06 below).

Documentation made available by NDK includes the design of the logic trees, weights of logic tree branches and a description of the software used for hazard calculation. The set-up of the logic tree, weighting of the different models and hazard calculations were made by WorlyParsions. The logic tree accounts for different completeness periods assumed for the earthquake catalogue. Uncertainties over historical earthquake locations and intensity/magnitude conversions are apparently not considered. In all source models the highest hazard contribution derives from the background zone encompassing the site. Hazard deaggregation highlights seismic sources at distances 0 - 25 km with magnitudes M < 6.5 as the most important hazard contributors (p. 6.6-72 to 6.6-73).

PSHA revealed the following ground motion parameters (p. 6-6.62):

Hazard value (10^{-4} per year, mean value, 5% damping): **$PGA_{10\text{-}4\text{-}h}$ = 0.359 g**

Hazard value (10^{-5} per year, mean value; 5% damping): **$PGA_{10\text{-}5\text{-}h}$ = 0.662 g**

With respect to the DBE, NDK stated that the DBE spectra of ENVY/KOERI (2013) and PSHA 2017 are almost identical, with the (ENVY/KOERI, 2013) spectra being slightly on the conservative side. NDK further explained that main differences between the SHA of ENVY/KOERI (2013) and PSHA 2017 are minor modifications to seismic source zones, different weights assigned to the five source models in the logic tree, and the use of different software in PSHA. These differences were regarded inconsequential as indicated by the comparison of the DBE spectra.

Assessment:

The consolidated PSHA study based on input data and models prepared by four different groups (seismic zones, source parameters, GMPEs, soil conditions etc.) can be classified as good practice in seismic hazard analysis reports. The family of hazard curves calculated from the logic tree is based on 822 960 calculations. The logic tree appears robust and the hazard curves incorporate epistemic uncertainty in the input data. The seismic hazard obtained by this robust logic tree is a strong safety feature of the Akkuyu NPP.

The high hazard contribution from near-site sources shown by hazard deaggregation calls for a very careful assessment of potential seismogenic sources at distances up to 25 km or even 50 km from the site. The most important currently known potential active faults within this distance are the SW segment of the Ecemiş (Namrun) fault and the offshore Kozan fault.

The hazard contribution of the Ecemiş (Namrun) fault is treated differently in the source models but appears to be considered by all except the Rizzo 1 model (which has a relatively high total weight of about 0.3 in the source zone logic tree). It should be clarified how close this fault (or its SW termination) is located to the site. It appears that no detailed geomorphological, geophysical and paleoseismological analyses were

	performed onshore to assess the fault. The Kozan fault is not distinguished as a separate structure. It is contained in the background zone in all source zone models. During the country visit NDK explained that PSHA 2017 was approved by NDK and underwent independent peer reviews by TÜV Süd and an IAEA Seed Mission. The PRT, however, notes that the report by IAEA (2021) does not equate to a review of PSHA. **Recommendations:** The PRT recommends the following action to increase the reliability of the PSHA results: **T1-03-1. High priority.** Investigate the location and possible activity of the Kozan fault (see **T1-04**) **T1-03-2.** Evaluate the quality and completeness of the earthquake catalogues used in the PSHA (see **T1-05**) **This issue is solved.** **T1-03-3.** Extend geomorphological, geophysical, paleoseismological (etc.) investigations to faults at distances up to 25 km or even 50 km from the site to exclude their activity or assess their seismogenic potential (see **T1-06**) **T1-03-4.** Perform sensitivity analyses on the choice of the minimum magnitude (see **T1-07**) **This issue is solved**.
T1-04	**Seismic hazard: Ecemiş (Namrun) and Kozan faults** **Description** The total offset of the Ecemiş fault zone is between 60 and 90 km since the Oligocene – Miocene time (Jaffey and Robertson [6-93]; Satır Erdağ [6-136]), long-term slip rates are estimated to be in the order of 3 mm/a. Quaternary slip rates based on the amount of displacement of active stream courses are also in the order of 3 mm/a (Kocygit and Beyhan [6-100]; Jaffey and Robertson [6-93]; Bayer & Altin [6-25] and [6-26]). Other geomorphic features are shutter ridges, pressure ridges and warped terraces. This geological evidence is consistent and allows the estimate of Mmax = 7.5 (all information from AKU-BDD0132, Rev. B04, 20.1.2017 p. 6.6.-34). AKU-BDD0132 Rev. B04 20.1.2017 (p. 6.5-27; p. 6.5-33ff) includes the results of seismic monitoring for the near region and region around the site including earthquake locations, fault plane solutions and maps of geomorphological lineaments identified by WorleyParson (2012). Lineaments include a series of SW-NE-trending lines parallel to the Ecemiş fault N and E of the site, seismicity patterns and fault plane solutions compatible with sinistral strike-slip on SW-NE-striking strike-slip faults (i.e. faults paralleling the Ecemiş fault). Document AKU-BDD0132 Rev. B04 20.1.2017(p. 6.7-17 ff) includes the interpretation of offshore seismic reflection (Fugro, Anatolian Geophysical, 2012; industry seismic and data acquired for site investigations). Based on 3D offshore seismic and the continuity of the Mid

	Pliocene seismic horizons WorleyParsons conclude *'that seismic sections unambiguously demonstrate the lack of any offset in the top sediment deposits'*. During the desktop study, the PRT recommended investigating the active Ecemiş (Namrun) fault and the Kozan fault, both possibly extending into the near region of the site and reviewing existing literature with respect to these faults. NDK consequently re-assessed the named faults based on recent literature, unpublished filed studies and a regional study based on GPS data. With respect to the Ecemiş fault field studies confirm that the fault terminates at a distance of more than 150 km from the NPP site (Altunel, 2022[97]). For the Namrun fault it was concluded that the fault does not extend across the Mut basin (ca. 50 km N of Akkuyu). Strain rate fields calclucated from GPS data are regarded to support the interpretation by showing that strain in the near region of Akkuyu is not in line with strike-slip faulting. The review also addresses the possible offshore extension of the Ecemiş and Kozan faults concluding that seismic data claiming the existence of the offshore Kozan fault is weak, although it is conceded that data is not clear enough to finalise the present debates. **Assessment:** An onshore continuation of the Ecemiş fault, referred to as Namrun fault segment with a number of branch faults, has been introduced by Kocygit & Beyhan (1998[98], Fig. 8; 1999[99]) who inferred a continuation of the fault into the near region NW of the ANPP site. Higgins et al. (2015[100]) adopt the model of Kocygit & Beyhan estimating a minimum sinistral slip rate of 1.1 ± 0.3 mm/a for the Ecemiş fault. However, according to the re-assessment initiated by NDK in 2023 and the data provided to the PRT in addition to the NR it appears reasonable to assume that the Ecetmiş (Namrun) fault does not extend into the near region of the NPP. While the offshore seismic sections are interpreted by the licensee to indicate no Quaternary or active faulting in the offshore South of the site this is not so clear to the PRT. Seismic lines (e.g. AKU-BDD0132 Rev. B04 20.1.2017, line ANPP105, Figure 6.7-16, between letters E, A and T) may indicate Pliocene to Quaternary growth strata associated with faults oriented parallel to the Kozan fault.

[97] Altunel (2022), Paleoseismological study within the scope of preparing the geological geotechnical survey report based on the master zoning plan, Mersin Province, Tarsus District, Gülek District. Prepared for Akya Project, to be submitted to the Ministry of Environment, Urbanization and Climate Change.

[98] Kocygit & Beyhan (1998), Tectonophysics 284: 317-336.

[99] Kocygit & Beyhan (1999), Tectonophysics 314: 481–496.

[100] Higgins et al. (2015), Tectonics, 10.1002/2015TC003864.

	Extension of the Kozan fault into the area S of Akkuyu was inferred from a detailed study of 2D industry seismic survey (Aksu et al. 2014[101], Fig. 24). The patterns shown in AKU-BDD0132 Rev. B04 20.1.2017 (see above) approximately fit the location and orientation of the faults inferred by Aksu et al. (2014). Aksu et al. (2014, Figs. 2, 23) determined a conservative rate of 0.6 – 1.0 cm/a of left-lateral motion to the Kozan Fault in the Pliocene–Quaternary from the offset deltalobes of the Gösku river ca. 50 km NE of Akkuyu. The regional tectonic study of the Cilica Basin by Aksu et al. (2021) identifies two major structures in the offshore near region of the NPP: the E-W-to ENE-WSW-striking sinistral Anamur-Silifike fault zone (‘Kozan fault’) offsetting Pliocene–Quaternary sediments (Aksu et al., 2021, Fig. 15, 17[102]) and deep-seated S-direted thrusts (Fig. 17). Fault plane solutions (FPS) indicating sinistral slip on E-W to SW-NE-striking faults and thrust-type FPS offshore Akkuyu (AKU-BDD0132 Rev. B04 20.1.2017, Figure 6.5-30) fit the interpretation by Aksu et al. (2021). FPS, however, are not in line with the strain rate fields calculated from GPS data. Data therefore indicate the presence, rather than the absence, of seismic fault sources in the offshore south of the site. Notably, indications of thrust faulting are in line with the observed uplift of the coastline (AKU-BDD0132 Rev. B04 20.1.2017, p. 6.2-5 ff; Ögretmen et al., 2015[103]). Active faulting in the near offshore of Akkuyu is also supported by the data recorded by the local seismic observation network. Within an observation period of only 3 months a remarkable number of small earhquakes (M up to ca. 2) were recorded at distances of few tens of kilometres from the plant[104]. This seismicity should be recorded and anlysed in order to obtain information on the related seismogenic structures.The PRT stresses that hazard deaggregation performed in the PSHA (2017) highlights the contribution of near-site sources. A correct assessment of these sources is therefore of high importance for the correct hazard assessment and corresponding nuclear safety of the plant. **The issue is resolved with respect to the Ecemiş (Namrun) fault.** **Recommendation:** **T1-04-1. High priority.** The PRT recommends investigating the potentially active Anamur-Silifike (‘Kozan’) fault zone, possibly extending into the near region of the site in detail to increase the reliability of the PSHA result.

[101] Aksu et al. (2014), Tectonophysics 622: 22–43.

[102] Aksu et al. (2021), Tectonophysics 814: 228952.

[103] Ögretmen et al. (2015), Natural Hazards, 79: 1569-1589.

[104] Report ‘Akkuyu Nuclear Power Plant Seismic Monitoring of Site Parameters. Quarterly Report 2023 (1) Stage 1.10 Monitoring Period 01.01.-31.03.2023’.

	T1-04-2. To increase the sensitivity and accuracy of the local seismic monitoring network and to support action related to recommendation T1-04-03, the PRT recommends installing two additional broadband stations on the Mediterranean shore W and E of Akkuyu, respectively. **T1-04-3.** The PRT recommends that the seismicity in the near region offshore of Akkuyu be systematically observed and analysed. Analyses should include efforts to locate hypocentres as accurately as possible and calculate fault plane solutions for stronger events. **T1-04-4.** The PRT suggests reviewing the data produced during the site investigations such as geomorphological interpretations and the interpretation of offshore seismic reflection in the light of the results of seismic monitoring to ensure that no active faults extend into the near region of the site.
T1-05	**Seismic hazard: input data and earthquake catalogue** **Description**: The Site Parameters Report (AKU-BDD0132 Rev. B04 20.1.2017) states that the maximum magnitude has been set based on the local catalogue compiled for the ANNP site. In another section, the SPR states that recurrence parameters have been recomputed using the regional catalogue compiled for the Akkuyu NNP. Furthermore, the report states that the source characteristics of the Koeri Model 2 have been updated using the results of recent seismicity (including the new earthquake catalogue developed). It is also referred to the 'instrumental catalogue of KOERI' or instrumental catalogue for Türkiye. And it is also stated that the earthquake catalogue includes no earthquakes within source zone 15 (i.e. the site vicinity within 5 km of the site). Moreover, the terms 'IRIS catalogue' and 'recently updated consolidated project catalogue' are also used in the SPR. Appropriate homogenisation and declustering applications and completeness analysis were applied to the instrumental catalogue. The Site Parameters Report states that the seismological data was gathered by Rizzo and Associates and KOERI independently. Rizzo and Associates included internationally collected data to a greater extent, and that 'some checks and analyses were performed for the historical data', and that it was concluded that there were two points that needed further study. The first was related to historical earthquake that was reported to occur at the boundary of the Akkuyu site source zone (in the KOERI and Rizzo models). This earthquake was investigated through a dedicated study by E. Guidoboni. It was firmly established that the actual epicentre was in Western Türkiye and not the one reported by the published catalogue. The second point was related to earthquakes that occurred in classical times and with extremely scarce information. An archeoseismological study was conducted to the East of Akkuyu and possible classical earthquakes were investigated through archeological evidence. The maximum magnitudes of all these events were determined to be smaller than what was already postulated in the seismic source models. **Assessment**:

	Kadirioglu et al. (2018) published an improved earthquake catalogue (M>4.0) for Türkiye (1900-2012). The citation from Kadirioglu et al. (2018) reads: 'Many catalogues, agency reports and research articles have been published on seismicity of Türkiye and its surrounding since 1950s. Given existing magnitude heterogeneity, erroneous information on epicentral location, event date and time, this past published data however is far from fulfilling the required standards.' The study also comments on previously compiled earthquake catalogues of Türkiye and describes their shortcomings. Another study by Kalafat et al. (2021) states: 'The previously available catalogues for Türkiye had some deficiencies such as magnitude heterogeneity, some uncertainties and errors on epicentral location, date and time that may lead to unreliable results in seismic hazard and risk studies. These ambiguities reinforced the need for an improved and updated catalogue to support the earthquake hazard and risk assessments.' During the desktop review NDK provided detailed information on the earthquake catalogue (ENVY/KOERI 2013). The PRT also received a detailed comparison of this catalogue with the one by Kadirioglu et al. (2018) with respect to regional coverage and magnitude homogenisation, and an electronic copy of the ENVY/KOERI (2013) catalogue. NDK further confirmed that the earthquake catalogue was compiled in compliance with IAEA standards (IAEA SSG-9 (Rev.1), paragraphs 3.44 - 3.48) and if the most recent input/research data were used. **Resolution:** NDK stated that a specific earthquake catalogue was developed and used for the purpose of the consolidated PSHA study showing that it was compiled considering recommendations of the IAEA Specific Safety Guide SSG-9 (Rev.1, 2022, paragraphs 3.44 - 3.48) and most recent data/research papers. **The issue is solved.**
T1-06	**Seismic hazard: use of paleoseismological data** **Description:** During the desktop review the PRT asked NDK to clarify whether paleogeological and paleoseismological data were used in the PSHA study and to what extent. To determine the seismic hazard to an AEP of 10^{-5} and lower (needed in the PSA) with reasonable accuracy, as much paleogeological and paleoseismological data as possible should be implemented into the PSHA. NDK answered that paleoseismological (trenching) data are shown in WorlyParson (TNPP-00-SV-REP-EN-0053-R2, p. 6-298ff). Further, comprehensive paleogeological study was performed by Prof. Erdoğan Demirtaşlı (Geopet) in 2013 (AKU-BDD0132 Rev. B04 20.01.2017, p.

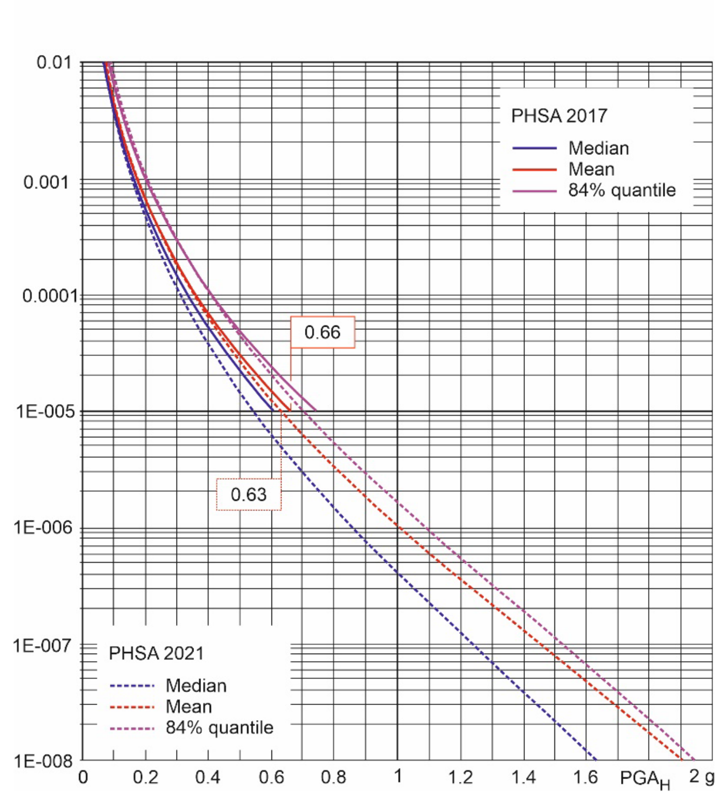
	6.2-4ff). Trenching addressed adressed faults at the site and the Akkuyu and Aksaz fault in site vicinity (ca. 0.5 to 1 km from the NPP). Results confirm that none of the trenched faults is active. With respect to the near region of the site (< 25 km distance), NDK stated that neither the Ecemiş Fault nor the Namrun Fault extend into this area and that paleoseismological studies are not warrented. **Assessment** PRT reviewed the provided data concluding that the two documents cited above neither include results of paleoseismological investigations nor the involvement of paleoseismological data in PSHA. Data, however, indicate that faults identified at the site are not active, although some of the documentations in TNPP-00-SV-REP-EN-0053-R2, (p. 6-298ff) do not suffice to exclude Quaternary faulting, due to the absence of Quaternary cover on top of the faults (e.g. trench T1, T2). No age dating of post-tectonic sediments seems to be available from the trenches, which is contrary to international practice. The high hazard contribution from near-site sources shown by the hazard deaggregation of PSHA results calls for a very careful assessment of faults at distances up to 25 km or even 50 km from the site (T1-03). According to current information, no geomorphological, geophysical, paleoseismological (etc.) efforts were made in the indicated area to analyse known faults, exclude their activity, or, where fault activity is confirmed, assess their seismogenic potential. **Recommendation:** **T1-06-1.** The PRT recommends extending systematic geomorphological, geophysical, paleoseismological (etc.) investigations to faults at distances up to 25 km or even 50 km from the site to exclude their activity; and if fault activity is confirmed, assess their seismogenic potential by constraining the timing, magnitudes and recurrence intervals of strong (prehistoric) earthquakes. The recommendation follows the requirements by WENRA (2021[105], Issue TU) to extend the time coverage of the seismological database beyond historical records and WENRA guidance[106].
T1-07	**Seismic hazard: minimum magnitude** **Description:**

[105] WENRA (2021), Report WENRA Safety Reference Levels for Existing Reactors 2020, Issue TU.

[106] WENRA (2020), Guidance Document Issue TU: External Hazards Head Document; WENRA (2020), Guidance Document Issue TU: External Hazards Guidance on Seismic Events.

	As reported in the Site Parameters Report, the minimum magnitude (M_{min}) considered in PSHA 2017 is M_{min} = 3.5. **Assessment:** Even small differences in M_{min} (~0.1) may influence the calculated peak accelerations and hazard response spectra, especially at lower returning periods and higher frequencies. The degree to which M_{min} affects the calculated seismic hazard depends on the seismicity level, zonation type, maximum magnitude choice, period (in the case of response spectra) and the attenuation relationship. The choice of M_{min} should not be determined by the completeness levels of the earthquake catalogue as it was indicated to the PRT by NDK experts, nor by the range of applicability of the selected GMPEs. The choice of the M_{min} value should be justified in the documentation on the hazard study. **Resolution:** The PRT suggested performing a sensitivity analysis on the choice of the lower bound magnitude (M_{min}) used for the PSHA calculation to justify the choice M_{min} = 3.5, or, alternatively to make a more conservative choice of the M_{min} value. NDK replied that M_{min} = 3.5 was used by KOERI originally, and this represented a conservative approach. M_{min} = 3.5 was also considered for the consolidated SHA because M_{min} should not vary from one model to the other as it does not represent modelling uncertainty. The resulting DBE may be somewhat conservative regarding the damage potential of an Mw = 3.5 earthquake. NDK further provided a sensitivity analysis comparing the hazard results for M_{min} = 3.5 and M_{min} = 5.0 for the seismic source zone containing the NPP. This analysis showed that the choice of M_{min} = 3.5 leads to conservative results, in line with the findingy by Bender and Campbell (1989). **The issue is solved.**
T1-08	**Seismic Hazard: PSHA used for PSA (PSHA 2021)** **Description:** The Akkuyu Site Parameters Report (AKU-BDD0132 Rev. B04 20.1.2017, p. 6-6-1 to 6.6-176) includes hazard curves and uniform hazard response spectra (UHRS) for horizontal and vertical ground motion down to occurrence probabilities of 10^{-5} per year (T1-03). Hazard curves extending to lower occurrence probabilities, which are required for PSA, are included neither in the cited document nor in in the NR. NDK's written replies to the PRT desktop question Topic 1 explained that hazard curves were calculated for probabilities 10^{-3}, 10^{-4} and 10^{-5} following IAEA SSG-9 and according to a SSHAC[107] guidance. NDK further stated that design incorporates a beyond design basis earthquake

[107] U.S. NRC, 1997. Recommendation for Probabilistic Seismic Hazard Analysis: 'Guidance on Uncertainty and the Use of Experts', NUREG/CR-6372.

	based on the European Utility Requirements (EUR) criteria, i.e. a factor of 40% over the design basis earthquake, to exclude cliff edge effects associated with the seismic design of the plant. The reply also included the graph of a family of hazard curves extending to occurrence probabilities down to 10^{-8} per year (here referred to as PSA 2021). During the 2022 country visit, NDK stated that these hazard curves differ from the curves in PSHA 2017 (Site Parameters Report):  Left: comparision of hazard curves PSHA 2017 and PSHA 2021 PSHA 2017 (Site Parameters Report): Hazard curves considering all source models and using the reported σ GMPE (AKU-BDD0132 Rev. B04 20.01.2017, p. 6.6-61. PSHA 2021 (WorleyParson): Hazard curves for PGA (truncation at 3σ) used for PSA. NDK explained that the PSA 2021 hazard curves ('single station, total sigma') were used in a first version of the seismic PSA for power unit 1. These hazard curves seem less demanding than the PSHA 2017 results (see below). However, for the current PSA of unit 1 and the other units NDK required the use of seismic hazard curves which fully conform to PSHA 2017 (Site Parameters Report).

	The PSHA 2017 curves were used for annual occurrence probabilities between 10^{-2} and 10^{-5}. This range includes the occurrence probability of events causing ground motion equal to 1.4 SSE. For the exceedance interval from 10^{-5} to 10^{-7} the curve from the WorleyParsons report TNPP-00-SV-REP-EN-0128-R1 (2.9.2014) with truncation at **3σ** (PSA 2021) was used (elsewhere it has been stated that all curves for annual exceedance probabilities from 10^{-2} to 10^{-7} were calculated on the **'full sigma'** approach). The PSA 2021 curves extending to the lower occurrence probabilities were said to be calculated on a *'less conservative approach'* (*'single station, total sigma'*) than the calculation of PSHA 2017 curves, which were characterised as *'conservative'.* The PRT understood that calculations of hazard curves in the WorleyParsons report used the same input data as the ones described in the Seismic Parameters Report (PSHA 2017). **Assessment:** Developing seismic hazard values down to 10^{-5} per year (as in PSHA 2017) is not sufficient for the assessment of the seismic risk in the PSA. The PSA and the corresponding probabilistic demonstration of practical elimination of large or early releases needs calculations of hazard values to extremely low occurrence probabilities. The PSHA 2021 hazard curves meet this requirement by extending to the occurrence probability of10^{-8} per year. The hazard values from the hazard curves PSHA 2017 and PSA 2021 are not identical. Graphical comparison shows that both PHSA 2017 and PSA 2021 reveal virtually identical hazard values, down to the occurrence probability of 10^{-4} per year, while between 10^{-4} and 10^{-5} the PSA 2021 curves show lower PGA values than the PSHA 2017 curves. The difference increases progressively towards lower occurrence frequencies (0.66 g vs 0.63 g for 10^{-5}; see figure above). Based on the currently available information, the PRT concludes that the PSHA 2017 hazard results were replaced by the less demanding hazard values of the PSHA 2021. The PRT was not provided with a document indicating how the hazard re-calculation also changed the UHRS and, correspondingly, the design spectra that underlie the design and the fragility assessment. The Russian and NDK experts verbally confirmed in the meetings that the design spectra enveloped the updated UHRS. **Recommendations** **T1-08-1. High priority.** The PRT recommends NDK analysing and clarifying the differences between the hazard curves included in PSHA 2017 (Site Parameters Report) and the hazard curve PSA 2021 (used for PSA). Typically, a mean hazard curve (considering adequately the uncertainties) is used in the PSA. NDK should request a detailed justification for the approach used and the reason for deriving a *'less conservative'* hazard curve for the PSA.

T1-09	**Fragility and seismic margin assessment (general)** **Description:** Seismic categories I and II equipment is assessed for seismic margin (i.e. prevent cliff edge effects) at a seismic demand level 40% higher than the SSE, according to the recommendation of the European Utility Requirements (EUR). In the event of such seismic impact, the NPP must be transferred into a safe state, preventing the release of radioactive substances into the environment. According to verbal information from the Russian and NDK experts received during the site visit, seismic analysis was performed at the 1.4×SSE level. The HCLPF values were obtained from the fragility assessment by the Separation-of-Variables (SoV)[108, 109, 110] method, performed for a reference earthquake RE = 1.4×SSE. According to the verbal information from the Russian and NDK experts received during the site visit, the SSE design floor-response spectra enveloped not only the structural nodes at the corresponding floor elevation but also all the nodes over the building height, i.e. they are the envelope of all floor-response spectra. Furthermore, in the evaluation of the floor-response spectra, the effective out-of-plane stiffness reduction of the reinforced concrete slabs and walls due to cracking was ignored, both in the SSE design and the fragility assessment. In evaluating the floor-response spectra used as seismic demand in the design and fragility assessment of the components, the ground motion incoherence was not considered. In the meetings with the Russian and NDK experts in 2022 and during the site visit, the PRT learned that Russian practice/experience was used to define the energy absorption factor F_{μ} in the design and the fragility assessment. In the meetings with the Russian and NDK experts in 2022, the PRT learned that the fragility of all safety-relevant component classes was assessed. No components were screened out and thus no surrogate element was used in the Seismic PSA. **Assessment:**

[108] Electric Power Research Institute, EPRI-TR-103959, 'Methodology for Developing Seismic Fragilities', Palo Alto, 1994.

[109] Electric Power Research Institute, EPRI-1002988, 'Seismic Fragility Application Guide', Palo Alto, 2002.

[110] Electric Power Research Institute, EPRI-1019200, 'Seismic Fragility Applications Guide: Update', Palo Alto, 2009.

	The general seismic margin of 40% is less than margins used in other countries for new NPPs, e.g. the regulator in France requires a factor of 1.5, in USA 1.67[111]. The selected 40% margin may or may not be sufficient to render early or large releases extremely unlikely with a high degree of confidence. The actual margin needed to reach this objective is driven: (1) by the general safety expectation (working hypothesis: no early or large releases for events with occurrence probabilities of 10^{-7} per year); (2) the hazard progression beyond the design basis requirements, as depicted by hazard curves that extend to sufficiently low occurrence probabilities (e.g. 10^{-7} per year). **Compare T1-02.** The PRT notes that the safety assessments were performed for a single-unit site. However, there are four units at the Akkuyu site. Therefore, the PRT recommends that NDK require the licensee to perform a multi-unit risk assessment in accordance with the latest IAEA Safety Reports SRS No 96 (2019) and SRS No 110 (2023). The SoV fragility evaluation should be performed with the median spectral acceleration seismic demand, elastic capacity using median-centred material strength and strength equations and inelastic energy absorption based on median limits for nonlinear distortion. Under the assumption of lognormal distribution, it is necessary to calculate three parameters to define the fragility curves for SSC failure modes: • median ground acceleration capacity (Am), • aleatory variability (βR), • epistemic variability (βU). A high confidence of low probability of failure (HCLPF) capacity is defined as the ground motion level at which there is a high (95%) confidence of a low (at most 5%) probability of failure. In fragility terms, it is defined as HCLPF = A_m*exp[-1.656($\beta_R + \beta_U$)]. The HCLPF seismic capacity values are compared to the **1.4×PGA$_{SSE-h}$ = 0.543 g**. **It is expected to see all safety-relevant SSCs having a seismic margin above the 1.4× PGA$_{SSE-h}$ = 0.543 g level, to avoid cliff edge effects.**

[111] Gürpinar et al. (2015), Considerations for Beyond Design Basis External Hazards in NPP Safety Analysis, Transactions, SMiRT-23, Division IV, Paper ID 424.

	According to the verbal information received from the licensee and NDK experts during the 2022 meetings, in the evaluation of the fragilities the new EPRI guide (2018)[112] (incorporating lessons learned from the post-Fukushima SMAs and PSAs performed in the US) was not considered. The evaluation of the energy absorption factor F_{μ} is decisive for the fragility evaluation. This factor should be calculated in accordance with the chosen SoV fragility assessment method based on median limits for nonlinear distortion. The Russian practice/experience was used to define energy absorption factors Fμ in the design and the fragility assessment, in which the coefficient K_e is used, and it is the reciprocal of the energy absorption factor F_{μ}. In this regard, NDK stated: 'The seismic design analyses of buildings and structures of the Akkuyu NPP to SSE-SL2 (1.4SSE) and DBE (SL1) level were performed in accordance with NP-031-01, taking into account the reduction factors of 0.625 for buildings of the seismic category I and 0.5 – for the buildings of seismic category II, taking into account the development of inelastic deformations. Thus, this safety factor is assumed to be equal to one and is not considered further in the analysis of seismic damage to buildings and structures.' From the statement above, it is clear that in the SSE design, the licensee used a K_e value of 0.625 (Fμ = 1.6) for buildings in seismic resistance category I, and K_e = 0.50 (i.e. F_{μ} = 2.0) for buildings in seismic resistance category II. In the seismic fragility assessment, the licensee then used a value of K_e = 1.00 for buildings in both seismic resistance categories. The PRT notes that the Russian practice for evaluating and applying the energy absorption factor F_{μ} in the design basis domain is inconsistent with EU and international practice. IAEA SSG-67, ASCE 43-05 and ASCE 43-19 require that structures in seismic category 1 should be designed to exhibit linear behaviour. The obsolete version of this IAEA guide (NS-G-1.6) states that structures in seismic category 1 and 3 may be designed to exhibit nonlinear behaviour, provided that their acceptance criteria are met with a safety margin consistent with the seismic categorisation. The PRT found that the Russian approach to the SSE design of the Akkuyu NPP SSCs, considering K_e coefficients other than 1.0, contradicts EU and international practice. As a consequence, the PRT recommends that NDK require the licensee to demonstrate the required conservatism, according to the acceptance criteria for the design basis. As regards floor-response spectra, the application of the energy absorption factor F_{μ} is only applied in designing the structures but not in developing the in-structure and floor-response spectra. The floor-response spectra used in the design conservatively envelops the structural nodes at all floor elevations throughout the building height.

[112] Electric Power Research Institute, 2018. Seismic Fragility and Seismic Margin Guidance for Seismic Probabilistic Risk Assessments, TR-3002012994.

	The design floor-response spectra were determined without consideration of the ground motion incoherence. The PRT considers the application of Abrahamson's model (1993) to be an inadequate assumption that will yield floor-response spectra which are not on the safe side. The basis for Abrahamson's spatial coherence model (1993) is soft soil conditions. This assumption contradicts the hard foundation medium conditions (i.e. hard-rock) at the Akkuyu site, where the first 30 m soil strata exhibit an average shear wave velocity V_{s30} = 1138 m/s (assumed according to the NR as the datum for the entire site). It should be noted that according to Section 3.7.2 of the US NRC SRP[113] and DC/COLISG-01[114], the US regulator accepts currently only the use of the horizontal and vertical coherency functions developed for hard-rock conditions based on the Pinyon Flat Array Data ($V_{s30} \approx 1030$ m/s), as detailed in the EPRI (2007)[115] report. Applicants for a licence in the US are expected to present comparisons between calculated coherent and incoherent seismic demands. Based on the US NRC staff's current experience, the following maximum reductions in the amplitude of spectral accelerations are acceptable for the ISRS: • 0 to 10 Hz – 0% reduction; • 30 Hz and above – 30% reduction; • 10 to 30 Hz – reduction based on linear variation between 0% at 10 Hz and 30% at 30 Hz. The design floor-response spectra in the vertical direction do not take into account the effective out-of-plane stiffness reduction of the reinforced concrete slabs due to cracking. According to NDK, the Russian standard NP-031-01 does not include provisions for this effect. Similarly, the effective out-of-plane stiffness reduction of the reinforced concrete walls due to cracking should be considered, if applicable, in the horizontal design response spectra for component design and their anchorages.

[113] U.S. NRC, 2013. Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition — Design of Structures, Components, Equipment, and Systems, Chapter 3, Section 3.7.3 Seismic Subsystem Analysis, NUREG-0800, Rev. 4.

[114] U.S. NRC, 2008. Interim Staff Guidance (ISG) DC/COL-ISG-01, Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications, US NRC ADAMS Accession No. ML081400277.

[115] Electric Power Research Institute, 2007. Hard-Rock Coherency Functions Based on the Pinyon Flat Array Data, US NRC ADAMS Accession No. ML071980104.

	Neglecting the effective out-of-plane stiffness reduction of the reinforced concrete slabs and walls due to cracking in both the SSE design and fragility assessment may lead to the unsafe design of components and/or their anchorage. The PRT recommends that NDK require the licensee to demonstrate that ignoring these out-of-plane effects in both the DBE (SSE) design and fragility assessments will not result in an unsafe design or lack of seismic margins of safety-relevant SSCs and/or their anchorage. In the meetings with the Russian and NDK experts in 2022, the PRT pointed out that frequently the governing failure mode of a component is its anchorage. In this regard the rigid baseplate assumption as outlined in EN 1992-4:2018[116] is not conservative for slender baseplate geometries and may lead to unsafe design. Hence, the flexibility of the baseplate has to be taken into account. However, the finite-element based design approach (FE), taking into account the baseplate flexibility, is sensitive to some key analysis assumptions (e.g. fastener stiffness). The FE approach implemented currently in the design software of some manufacturers (e.g. HILTI) under-predicts the anchorage capacity and can produce over-conservative design. Basic design and fragility assessment should reflect this issue. In the meetings with the Russian and NDK experts, the PRT learned that the fragility of all safety relevant components classes were assessed and no components were screened out. The PRT doubts the information that seismically rugged SSCs were not screened out. Perhaps there has been a misunderstanding. However, if Tables 2-3 and 2-4 from EPRI NP-6041-SLR1 (1991)[117] were used to screen out seismically rugged SSCs in the Seismic PSA evaluation, Section 4.2 in EPRI (2018)[112] recommends all screened out components be replaced by a surrogate element in the seismic PSA, with the surrogate element having the following two fragility parameters (lognormal fragility function): $\mathbf{C_{gm}} = 2 \times Sa_{SL} \times \exp(-b_{PV,R})$ and $\mathbf{b_C} = 0.3$; where: C_{gm} is the median peak ground spectral acceleration capacity of the surrogate element, Sa_{SL} is the peak 5% damped spectral acceleration of the ground motion, representing the EPRI NP-6041-SLR1[117] HCLPF level that the SSC satisfies, $b_{PV,R}$ is the response spectral peak and valley variability included in the fragility evaluation,

[116] CEN, TC 250 - Structural Eurocodes, Eurocode 2, 'Design of concrete structures - Part 4: Design of fastenings for use in concrete', EN 1992-4: 2018.

[117] Electric Power Research Institute, 1991. A Methodology for Assessment of Nuclear Power Plant Seismic Margin, Report NP-6041-SL Revision 1.

	b_C is the composite (mean) variability assigned to the surrogate element. **Recommendations:** **T1-09-1. High priority:** The PRT recommends that NDK/the licensee verify that the chosen 40% margin is sufficient to eliminate cliff edge effects and make early or large releases extremely unlikely with a high degree of confidence (i.e. practically eliminate early or large releases). (see introductory remark T1-00). **T1-09-3.** NDK/the licensee should verify that a realistic out-of-plane stiffness reduction of the reinforced concrete slabs and walls due to cracking would not adversely affect the floor-response spectra in the vertical and horizontal directions used in the design and fragility assessment and thus prevent unsafe design of components and/or their anchorage[118]. **T1-09-4.** NDK should ask the licensee to justify the accuracy of the value of the energy absorption factors they derived because of the decisive contribution of the absorption factors to the seismic fragility of safety relevant SSCs. **T1-09-5.** NDK/the licensee should verify that the design and fragility assessment of anchorages with slender baseplate geometries are not affected by an an unrealistic rigid baseplate assumption, leading to unsafe design. **T1-09-6.** NDK should ensure that the PSAR includes a reference to the standard or procedure for seismic qualification by test that is going to be used.
T1-10	**Seismic margins** **Description: and Assessment:** The following assessment compares the HCLPF values of SSCs that are important for safety with the margin $1.4 \times PGA_{SSE\text{-}h} = 0.543$ g, for which, according to NDK, it must be ensured that the NPP can be transferred into a safe state and the release of radioactive substances into the environment prevented.

[118] The PRT takes note that the response spectra were calculated based on NP-031-01, wherein no provision exists for the task of lowered stiffness properties in reinforced concrete structures, following their cracking recommendation.

	A document provided by NDK during the desktop review[119] provides a comprehensive list of the HCLPF values of all types of SSCs. Hazard value (10^{-4} per year, mean value) and DBE: **$PGA_{SSE\text{-}h}$ = 0.388 g**, $PGA_{SSE\text{-}v}$ = 0.295 g Hazard value (10^{-4}) × 1.4 = 1.4×**$PGA_{SSE\text{-}h}$ = 0.543 g** Hazard value (10^{-5} per year, mean value): **$PGA_{10\text{-}5\text{-}h}$ = 0.662 g** or **$PGA_{10\text{-}5\text{-}h}$ ≈ 0.63 g** (see T1-08) The lowest HCLPF values are listed for the following SSCs (values lower or close to the hazard value 1.4×$PGA_{SSE\text{-}h}$ = 0.543 g are highlighted in bold; list is not comprehensive): 11UQZ and 12UQZ (emergency power supply systems): 0.38 g 12BMF (switchgear): 0.50 g 11QKB (cold supply): 0.49 g (pump) 11SAC (10UBA normal operation power supply building): 0.46 g (ventilation unit, fan) 11SAE (10UAZ 400 kV power output system bus duct tunnel): 0.51 g (ventilation unit) 11SAD (ventilation system 11UBN-13UBN EDG building): 0.46 g (fans) 11XKA10 (generator set): 0.49 g SSCs of the safety systems' power supply (see below) **Limiting margins for safety systems required in DiD levels 3 and 4**:

[119] Probabilistic Safety Assessment Volume 1 Level 2 Probabilistic Safety Assessment Book 16 Fuel damage frequency assessment for seismic events: AKU-VAB0101-BAA0016.

	Containment and containment integrity: 10UJA (inner containment): **0.55 g** (loss of robustness of certain bearing reinforced concrete structures indicated in the NR; however, according to Russian experts it is evaluated for cracks in the foundation). The value was said to be not equal to ground motion causing collapse (collapse would induce core damage). NDK stated that this value is rather low for containment buildings. Russian experts announced they would re-calculate the fragilities using less conservative assumptions, to arrive at higher margin numbers. 10UJA (inner containment): **0.68 g** (loss of containment tightness due to cracking)[120] Outer containment: not analysed; the designer expected much higher stability than for inner containment as the outer containment is designed for e.g. airplane crash. JMT and JMU (PARS: passive hydrogen recombiners), hydrogen management system in the containment: fragility defined by strength of their support = **0.75 g**. JMN system: Sprinkler system for protecting inner containment from overpressure = 1.25 g (piping) and 1.31 g (pump). Emergency shutdown: Control rods in full insertion position: 2.5 g. **<u>Assssment</u>**: The PRT considers this value implausibly high. During the country visit, the designer explained that the value of 2.5 g applies to the reactor internals. As regards the control rods, shutdown is ensured within the required time (4 seconds) after a trigger value is detected at OBE level (0.194 g).

[120] As regards the practical elimination of early or large releases, note IAEA DS508, p.27: 'Meeting a probabilistic target alone is not a justification to exclude the analysis and possible implementation of additional reasonable design or operational measures to reduce the risk. Thus, a low probability of occurrence of an accident with core damage is not a reason for not protecting the containment against the conditions generated by such accident.'.

	The designer explained that shaking table experiments showed that the control rods could be inserted in less than 3 seconds under SSE shaking using the time window between the automatic scram initiating their insertion and the arrival of the strong ground motion phase. The PRT questions the assumption that the time window between scram initiation and the arrival of the strong ground motion phase is long enough, for all expected earthquakes, to ensure full rod insertion. This particularly applies to near-field earthquakes with magnitudes between 6 and 7 with short duration. Also, the triggering value of 0.194 g may already be part of the strong ground motion phase. The PRT indicated in the 2022 meetings that relative displacements of key internal components in the reactor core are expected to define the governing failure mode. Potential failure modes that may prevent the control rods from reaching the position of full insertion within the required time window are expected to be: (a) excessive deformation of the control rod drive mechanism; (b) excessive relative displacements of the reactor pressure vessel internals (e.g. upper support plate vs reactor pressure vessel and upper core plate vs fuel assembly head); (c) excessive fuel assembly deformation. The PRT stressed that these potential failure modes should be considered in the fragility analysis, to define the governing failure mode. Boron injection system: 12JND (emergency boron injection): **0.66 g** (motor-operated valves). The PRT noted that this value was not in line with the expectation that, according to the DiD approach, the system shall be available after failure of the control rods in a seismic event. Failure of safety systems' power supply: Fastening of cable runs: **0.41 g** 11UQZ, 12UQZ (tunnels for emergency power supply system cables and service water systems): **0.38 g** Fastenings of electrical cabinets etc.: **0.52 g** Reinforced concrete structures of EDG building: 0.72 g 11BTA Battery fastening to racks: **0.42 g** Heat removal from core (emergency core cooling):

	11JNB (emergency cooling SG): 0.59 g (SG emergency cooldown heat exchanger) Fastening of heat exchanger: 0.61 g Hanger supports for pipelines and steam lines: 0.63 g Limiting value for heat removal from the core via JNB: 0.63 g. JNJ 3rd level hydro accumulators HA3 in annulus, including piping: 1.01 g JNJ HA2 in the containment, including piping and check valve: 1.01 g Core catcher: Core catcher: not analysed. JMN (for cooling the corium in the core catcher and feeding containment sprinklers): **1.31 g**; piping of JMN pump: **1.25 g**; availability of cooling water from SFP: **0.57 g** (see below) Spent Fuel Pool (SFP) SFP (leak tightness): **0.57 g**. **Assessment**: The value is significantly lower than the fragility stated for the JMN pump used for cooling the corium and supporting sprinklers for containment cooling; the pump is fed by water from the SFP. During the site visit the licensee informed the PRT that HCLPF values will be recalculated, SMA will be performed and the seismic PSA will be updated when all SSCs are in place. HCLPF values below $1.4 \times PGA_{SSE\text{-}h} = 0.543$ g are not in line with the margins required according to the licensing basis. It is recommended that NDK not accept the margins of SSCs important for safety with HCLPF < 0.543g and require retrofitting of the respective components. The PRT recommends that NDK review the licensee's approach to ensure safe shutdown of the reactor by inserting control rods during seismic events. **Assessment:**

	The PRT considers that the seismic margins (HCLPF) are small for a significant number of SSCs important for safety, such as the inner containment (0.55 g), the emergency batteries (0.41 g), the emergency power supply tunnels (0.38 g), the SPF (0.57 g), anchoring and ventilation systems. The NR and the document *Topic_1-44.pdf* provided by NDK identify numerous SSCs with HCLPF values which are below the value 1.4×PGA_{SSE-h} = 0.543 g and several SSCs which do not meet DB requirements of **PGA_{SSE-h} = 0.388 g** (11UQZ and 12UQZ tunnels for emergency power supply system cables and essential service water systems). The **loss of the confinement function of the containment at 0.68 g (HCLPF) corresponds to an occurrence probability of about 10^{-5} per year**, according to the PSHA results. It must be clarified whether this corresponds to a cliff edge. The low seismic margins of a significant number of SSCs important for safety lead to a contribution of the seismic risk (S-PSA) to the overall plant risk (in terms of FDF) which is very large: 80% by the reactor core, 91% including reactor core, spent and fresh fuel storage. These numbers indicate an unbalanced design which does not duly account for seismic hazards. The PRT expects that the seismic contribution to the overall plant risk in terms of LRF and/or LERF is also very high. Addressing seismic issues at this early phase is therefore highly important, to meet the chance of design modifications during the construction phase and to reduce the risk contribution of seismic events to an acceptable level. With respect to SSCs, which need sufficient seismic margins above the SSE to cope with DEC, it seems clear neither whether all necessary SSCs have been identified nor if all identified SSCs are sufficiently designed to ensure their integrity and function in accordance with their role in support of the Defence in Depth (DiD) levels. The seismic margins of SSCs required to prevent fuel damage and large or early releases (i.e. those relied on in DiD Levels 3 and 4) are only acceptable if they ensure CDF and LERF values which are low enough to meet the acceptance criteria (see T1-00 and T1-01). To ensure a suitable review and assessment is carried out, a consistent concept should be available that describes and justifies the seismic margins that SCCs need in order to comply with acceptable CDF and LERF values. At present such a demonstration or documentation has yet to be presented to the PRT. **Recommendations:** **T1-10-1. High priority:** PRT expresses concern about the calculated seismic fragilities (HCLPF values calculated by the designer; see T1-09) and suggests that NDK arrange for an in-depth review of methods and calculation of HCLPF values. **T1-10-2. High priority:** The PRT questions the data reported on the seismic resistance of the reactor core and the reactor shutdown system (full control rod insertion within the required time window). The PRT recommends reviewing the fragility calculation for the reactor core

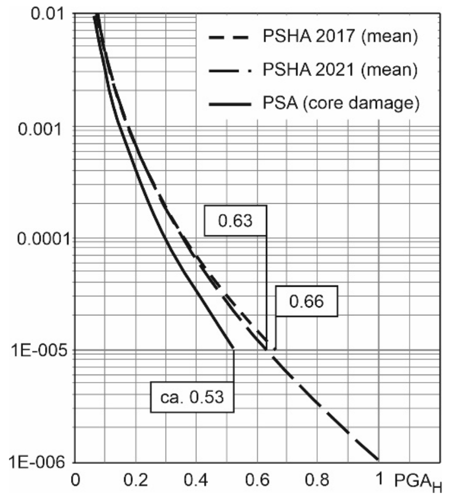
<table>
<tr><td></td><td>(needed for both SMA and seismic PSA) and the licensee’s approach to ensuring safe shutdown of the reactor by inserting control rods during seismic events.

T1-10-3. High priority. The PRT recommends that NDK:
1. not accept the margins of SSCs important for safety with HCLPF < 0.543g;
2. take necessary action to ensure compliance.</td></tr>
<tr><td>T1-11</td><td>Range of earthquake leading to severe fuel damage

Description:

The NR report gives values of the mean annual fuel damage frequency (FDF) derived from a seismic PSA. Calculations discretised the mean seismic hazard curve for the site ground motion parameter PGA, based on which the component fragilities of SSCs are developed, into 8 intervals between 0.2 g (occurrence probability = 4×10^{-4}/year) and 0.6 g (occurrence probability = 5×10^{-6}/year) with steps of 0.05 g.

The total mean annual FDF of core and pool fuel is stated with 5.59×10^{-6}; the individual FDF values calculated for the core fuel and the pool fuel are 4.64×10^{-6} and 1.84×10^{-6}, respectively. The values are below 10^{-5} per NPP unit per year, specified in the Russian nuclear standard NP-001-97.



Left: comparision of hazard curves PSHA 2017 and PSHA 2021 (broken lines) with the hazard curve used for seismic PSA and calculation of FDF, according to Figure 7 of the NR (bold continuous line).

PSHA 2017 (Site Parameters Report): Hazard curves considering all source models and using the reported σ GMPE (AKU-BDD0132 Rev. B04 20.01.2017, p. 6.6-61.

PSHA 2021 (WorleyParson): Hazard curves for PGA (truncation at 3σ) used for PSA 2021.</td></tr>
</table>

	In general, the PRT agrees with the discretisation of the mean seismic hazard curve for the site into 8 intervals, in order to evaluate the mean annual frequency of plant damage state and obtaining the seismic risk quantification by convolution of the seismic hazard curve and the plant damage state fragility curve. The mean seismic hazard curve used for the seismic PSA is shown in Fig. 7 of the NR. Remarkably, the shown hazard curve indicates significantly lower PGA values than the results of PSHA 2017 and 2021. Values compare as follows: PGA 10^{-4}/year used for PSA ≈ 0.3 g, PGA for 10^{-4}/year defining the DBE = 0.388 g (ENVY/KOERI 2013); PGA 10^{-5}/year for PSA ≈ 0.53 g, PGA for 10^{-5}/year according to PSHA 2017 and 2021 = 0.63-0.66 g. The reason for the discrepancies is currently not known. The NR shows in Fig. 8 which ground motion intervals contribute the most to the fuel damage frequency. The last bin (0.55 g – 0.60 g) appears to be the biggest contributor. This observation indicates that the hazard curve may have been truncated and the convolution with the plant damage state fragility is inaccurate – i.e. bins at higher PGA (i.e. smaller annual probability of exceedance) may have to be considered, if the conditional fuel damage probability (i.e. plant damage state fragility) shows significant increase that corresponds to a decline in the hazard curve. Simply expressed, the bar chart in Fig. 8 should look like a Gaussian bell curve (Gaussian distribution) to indicate a proper convolution. The PRT therefore assumes that the hazard curve was truncated and the convolution with the plant damage state fragility is inaccurate. In fact, the hazard curve shown in NR Fig. 7 terminates at 10^{-5} annual probability of exceedance. Hazard values for the 0.55 and 0.60 g bins appear to be extrapolated to beyond the range of the calculated hazard curve. **Assessment:** PGA values used in the seismic PSA which are significantly lower than the results of PSHA 2017 and 2021, and the apparent truncation of the hazard curve, lead the PRT to question the reliability of the stated FDF values. **Recommendations:** **T1-11-1.** It is suggested that NDK ask the licensee to review the calculated FDFs by ensuring that the hazard curve used for the PSA is appropriate and consistent with the results of the 2017 and 2021 PSHAs. It is further recommended that NDK check the conditional fuel damage probability (i.e. plant damage state fragility) if its increase is steeper than the corresponding decline in the hazard curve, for accelerations higher than 0.6 g. If their conditional fuel damage probability has a significant contribution to the total FDF, additional bins with higher PGA (i.e. smaller annual probability of exceedance) should be taken into account.

T1-12	**Indirect impacts of seismic events**
	
	Description:
	
	As regards indirect seismic impacts, NDK/the licensee provides information in Chapter 2.1 'Design basis' of the NR about the different measures or approaches taken to protect the plant in the event of a design basis earthquake:
	
	• Layout solutions are being developed for the Akkuyu NPP so as to physically separate systems into different seismic categories and exclude adverse effects caused by lower category equipment on higher category equipment .
	• If equipment and pipelines of different seismic categories are accommodated in the same room, they must be spaced, or equipment and pipelines in a lower seismic category will be additionally detached, to achieve structural integrity and stability. Thus, protection against the secondary effects of earthquake is achieved. This means that safety-related equipment should not fail during an earthquake up to the SSE level.
	• One of the following approaches is used to analyse the interface of seismic category I components with category II and III components:
	  • checking the robustness (operability) of higher seismic category component under loads caused by the failure of lower category component,
	  • lower seismic category components are designed for all external loads and impacts to be covered in the design of an adjoining higher category component. Contact interaction of structures not belonging to seismic category I with category I structures as well as category I structures is excluded by aseismic joints and layout solutions. The adequacy of aseismic joint width is verified by calculations.
	
	These measures allow the plant to prevent damage to safety-related components from indirect seismic impacts in the event of a design basis earthquake.
	
	Assessment
	
	In the NR, NDK/the licensee does not provide information regarding complementary measures or approaches to protect the plant against indirect seismic effects from earthquakes exceeding the SSE up to 1.4×SSE. The safety margin evaluation provided by NDK/the licensee in Chapter 2.2 'Evaluation of safety margins' of the NR does not explain how indirect seismic impacts are taken into account in the different measures described for SSE. So far, NDK/the licensee did not give an answer to the need to account for the influence of indirect impacts by seismic events on needed SSCs in DEC situation. However, they did state that margins are going to be re-evaluated and that walkdowns would be useful in this context.
	
	The PRT recalled that NRoT Chapter 2.2 'Evaluation of safety margins' states that 'The structural strength of seismic category I buildings and structures, equipment, process and other lines was separately tested for beyond-design basis seismic impact exceeding SSE by 40%.' Furthermore, In Topic 1-44 file, answering to PRT initial questions NDK/Licensee states that 'In the event of building failures, the model

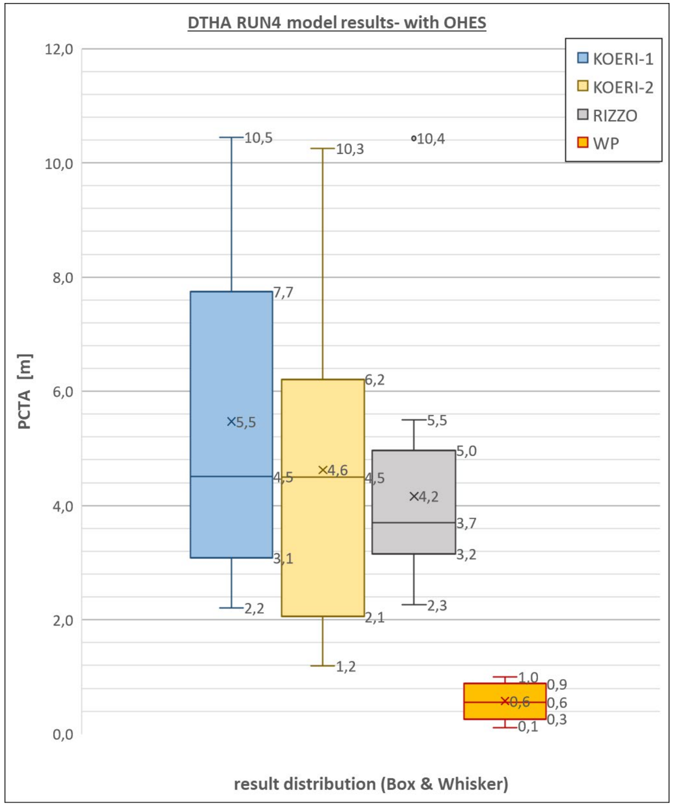
	postulates a dependent failure of all the equipment and structures located inside.‘ (AKU-VAB0101-BQB0016 Probabilistic safety assessment 16.4-60 AKKUYU NÜKLEER A.Ş. Akkuyu NPP Rev. B03 2020-12-29). It appears that indirect impacts of SSCs on neighbouring safety relevant buildings or SSCs were not evaluated for DEC earthquakes. SSCs important for safety for coping with earthquakes exceeding the SSE need to exhibit sufficient margin and the measures taken to protect these SSCs against indirect seismic effects should still apply up to 1.4×SSE. For instance, interaction of Category I or II SSCs should not impair the margin evaluation of an SSC needed in the event of a beyond design earthquake. **Recommendation:** **T1-12-1.** NDK should ask the licensee to provide an evaluation of the seismic margins (resistance, displacement) of SSCs that may have an impact (spatial interaction, pounding, failure or collapse) on safety-related SSCs needed in earthquake DEC situations. If these SSCs reveal insufficient seismic margin, their failure and impact on SSCs important for safety should be considered in the seismic margin assessment.
T1-13	**Seismic qualification of fire protection systems** **Description:** During the country visit, NDK and the designer informed about fire protection of cable tray systems under MCR and ECR. These rooms accommodate cables of different safety trains, which are not separated from each other with fire barriers or safe clearances. This makes it impossible to contain fire in cable tray systems within one train. As a result, automatic firefighting systems are designed as supporting safety systems: double-train design, 100% each, thus fulfilling the single failure principle. The systems are designed to function under extreme external hazards (SSE, hurricane, flooding etc.) as well as during design basis accidents. NDK further stated that firefighting systems are classified according to the safety class of the rooms they are installed in. The fighting systems in the two safety trains are therefore rated as seismic category 1. Fire extinguishing systems for the beyond design-basis accident management equipment have the following seismic categories: • the 0.4 kV alternative diesel generator has fire extinguishing systems since it is located outside the rooms; • equipment for managing BDBA (pumps of alternative intermediate circuit, additional cooling system of the cooling pond, heat exchangers, valves) is located in the same rooms as the safety systems and do not have fire extinguishing systems;

	• electrical equipment for BDBA management equipment is located in the UJA building, in which the fire extinguishing systems have category I seismic resistance and are designed for seismic impact up to 1.4 SSE. No information with respect to the on-site fire brigades is provided in the NR. **Assessment** During the site visit, the PRT noted that the fire station building was designed to withstand DBE loads. With regard to site accessibility following a strong earthquake, the PRT would like to point out the following: The major impact of the Niigataken Chūetsu-Oki earthquake on the Kashiwazaki-Kariwa nuclear power plant included significant damage to non-safety access roads, which impeded the response by the off-site fire brigade and other emergency responders. This significantly affected the operator's response to the earthquake. Such impacts should be considered in the safety analyses for Akkuyu. **Recommendation:** **T1-13-1. High priority.** The PRT recommends that NDK ensure that firefighting systems protecting SSCs important for safety are seismically resistant at least to the level of the SSE (compare WENRA, 2021, Reference Level SV5.6). SSCs with DEC functions, in particular SSCs necessary to prevent early or large releases, should also be protected from fire caused by DEC earthquakes. **T1-13-2. High priority.** The fire brigade building is designed to withstand $PGA_{SSE\text{-}h}$ = 0.388 g. The PRT notes that severe accident management also covers the fire brigade in DEC conditions (see Chapter 7). It is therefore recommended to upgrade the fire brigade building to at least 1.4×PGASSE-h = 0.543 g.
T1-14	**Adequacy of peak coastal tsunami amplitude (PCTA) results using the WorleyParsons (WP) model inputs and their impact on the DB tsunami** **Description**: Peak coastal tsunami amplitude (PCTA) results have been assessed for Akkuyu site using deterministic and probabilistic tsunami hazard assessment (DTHA and PTHA). Both the DTHA and PTHA rely on tsunamigenic source rupture parameters that have been determined using four different seismic models. While the use of these four different seismic models was intended to increase the robustness of the assessment, the PRT noted that one of the models (WorleyParsons - WP) leads to very low PCTA values compared to the other models. The WP model generates much smaller water level oscillations at the source level. **Assessment:**

	In the DTHA2018 report that considers the new local bathymetry (offshore hydraulic engineering structures), the WP model concluded that the mean tsunami runup for the RUN-4 most critical tsunami scenario for Akkuyu (used to define the tsunami design basis) would be only +0.58 m. Such a low max tsunami runup is not realistic and is contradicted by regional evidence of past tsunami runups. This includes paleo-tsunami evidence found in 2008 near Arkum, 2-3 meters above sea level and 1650 m from the coastline in the delta of the Göksu River (see Akkuyu paleo-tsunami study, trench ST-22). In 2017, a study was performed to check the tsunami models with the Arkum tsunami evidence. The results are in report AKU-BDD0132 'Tsunami inundation at Arkum and ST-22 paleotsunami deposit' in an appendix to the Akkuyu NPP Site Parameters Report. This report demonstrates that none of the 18 rupture scenarios from the WP model could reproduce the Arkum tsunami deposits. 'It also includes potential indications (coastal boulder displacement) of an at least 3-meter high tsunami found near Yesilovacık and Narlikuyu villages (Ögretmen et al., 2015[121]). Finally, the WP runup model results are strongly contradicted by the three other models which conclude the mean tsunami runup to be in the range between +4.2 m to +5.5 m for the same RUN-4 scenario.

[121] missing footnote [121] Ögretmen et al., 2015, Natural Hazards, 79: 1569-1589.

The Box & Whisker (right) representation of the 4 data sets shows that the WP data set is completely outside of the range of all the other data sets. The WP interquartile range (i.e. Q3-Q1 or P75%-P25%, containing the 50% most central values of each data set) of WP does not overlap any of the interquartile ranges of the other data sets. The highest value of the WP data set (+1.0 m) is below the lowest value of the 3 other data sets (+1.2 m).



Similar conclusions can be made for the later THAs.

As the tsunami DB is currently based on a statistical combination of the runup modelling results from the four models, the inclusion of the unrealistic WorleyParsons results (i.e. a statistical outlier) leads to an inappropriate reduction of the calculated tsunami DB.

Significant discrepancies are also observed between the WP model and the three others for the calculation of the lowest sea level during a tsunami event. This is also the case for other significant tsunami source zones, notably the RUN-7 zone.

The PRT questions whether the statistical combination of the four sets of DTHA results complies with standard practice by international regulators, e.g. US NRC documents RG DG-1290[122] and NUREG/CR-7223[123]. RG DG-1290[Error! Bookmark not defined.] states '… *simplified procedures may be used; however, all methods and assumptions should be clearly conservative*.'

The NUREG/CR-7223 states in Section 7.1 'Deterministic Approach': *' A number of the conservative approaches recommended here may appear so unrealistic as to be unfair to the analysis. However, this is a necessary side effect of a deterministic approach for a poorly understood hazard*

[122] U.S. NRC, 2022. 'Design-Basis Floods for Nuclear Power Plants', Draft Regulatory Guide, DG-1290.

[123] U.S. NRC, 2016. Tsunami Hazard Assessment: Best Modeling Practices and State-of-the-Art Technology, NUREG/CR-7223.

	that should have a low annual recurrence frequency. … For the deterministic approach, every allowance should be made to describe the source in the most conservative, but physically reasonable, manner. A deterministic approach, void of any measure of uncertainty quantification, requires such a conservative procedure.' **Recommendations:** **T1-14-1. High Priority.** The PRT recommends reconsidering the adequacy of the statistical combination of the results of the four DTHA studies and that the most conservative, but physically reasonable, DTHA results are used as tsunami demand in the DB. The PRT also recommends that the unrealistic WP values be removed from the tsunami DB assessment because of their significant impact on the key DB tsunami demand metrics, i.e. maximum and minimum sea water levels and maximum flow velocities.
T1-15	**Justification of the significant tsunami wave reduction in the 2018 and 2023 tsunami hazard studies** **Description**: In 2017, a first probabilistic tsunami hazard assessment (PTHA), using the natural morphology in the site area (without any structures), concluded that the calculated peak coastal tsunami amplitude (PCTA) and minimum sea level corresponding to a probability of 10^{-4}/y would be respectively +7.97 m and -7.77 m. A second study in 2018 accounted for the construction of offshore hydraulic structures in Akkuyu Bay, assessing the sea levels at 6 specific locations. This study concluded that the calculated maximum tsunami runup and minimum sea level corresponding to a probability of 10^{-4}/y would be respectively +4.37 m and -4.4 m. This means a reduction of both initial values by about 45%. In 2023, a new DTHA study (DTHA 2023) was performed, following significant design changes in the water inlet and outlet. This study also included improvements to address some of the PRT concerns, as described below. It assessed the PCTA at 32 points around the site, for the two most significant tsunamigenic sources (RUN-4 and RUN-7). Among all 72 scenarios modelled, DTHA 2023 concluded that the calculated maximum tsunami runup and minimum sea level are respectively +11.73 m (at location PIB-04) and -8.89 m (at location PIB-11). Nevertheless, DTHA 2023 uses a statistical combination of the modelled scenarios to conclude that the probable maximum tsunami runup and minimum sea level would be respectively +5.36 m and -4.6 m. When total sea level constituents (+2.08 m) are added then the maximum probable tsunami elevation at location PIB-04 is considered to be +7.44 m. **Assessment:** It was explained to the PRT that the major difference between the runup values in the 2017 and 2018 reports could be explained by the presence of very shallow areas near the shore in the original bay morphology where higher runups were obtained. Meanwhile, the site platform

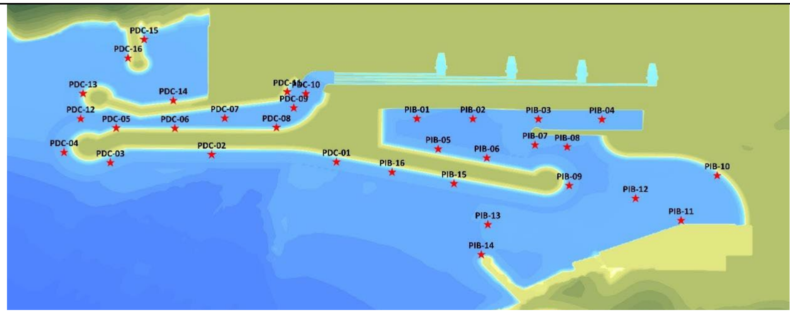
	has been expanded in the bay and the shoreline around the site is bordered by steeper sea walls or dikes. Inlet and outlet channels are being build, bordered by dikes. Dikes have a permeable structure and are made of rubble rocks, covered by large rocks and topped by large antifer concrete blocks. The stability of this structure against tsunami-induced erosion has been checked and found adequate. The seaward perimeter of the +10.50 m flat site platform consists in parts of slopes covered by antifer blocks (example: East corner), parts with a sloped shoreline (example: NW 'harbour/bay'), parts with vertical walls (example: pumping station) and a lower platform (the +3m South quay). METU confirmed that a very low bottom friction coeeficient was used in the modelling. Between DTHA 2017 and DTHA 2023, the rupture parameters in 7 key tsunamigenic scenarios for RUN-4 and 12 key scenarios for RUN-7 were reduced. There is no explanation or justification for this in the report. For instance, in these 19 scenarios the maximum displacement has been significantly reduced, as well as all resulting maximum and minimum amplitudes of the wave at the source. This in turn leads to a significant reduction in the tsunami runup predicted at the Akkuyu site. This is even more significant considering that the reduced scenarios were those leading to the highest PCTAs in DTHA 2017. The table below shows some of the differences in rupture parameters between the DTHA 2023 and the DTHA 2017 studies.

source zone	scenario	Max. Disp. (m)	Max (+)ve wave amp. at the source (m)	Max (-)ve wave amp. at the source (m)
RUN4	KOERI1-4	-5	-1,98	0,22
RUN4	KOERI1-7	-5	-2,24	0,37
RUN4	KOERI1-9	-5	-2,34	0,34
RUN4	KOERI2-7	-5	-1,63	0,94
RUN4	KOERI2-7b	-5	-1,63	3,03
RUN4	KOERI2-7c	-5	-1,63	0,98
RUN4	KOERI2-8	-5	-1,43	0,82
RUN7	KOERI1-5	-8	-2,8	0,25
RUN7	KOERI1-6	-10	-3,36	0,22
RUN7	KOERI1-9	-7	-3,26	0,52
RUN7	KOERI1-10	-5	-2,39	0,35
RUN7	KOERI2-7	-13	-4	2,36
RUN7	KOERI2-8	-8	-2,28	1,32
RUN7	KOERI2-9	-2	-0,91	0,21
RUN7	RIZZO-7	-8	-2,76	2,73
RUN7	RIZZO-8	-10	-2,81	2,77
RUN7	RIZZO-9	-15	-3,79	3,74
RUN7	RIZZO-10	-10	-3,56	3,51
RUN7	WP-10	0	-2,65	0,26

Any changes to the rupture parameters between the 2017, 2018 and 2023 THAs should be covered by a careful and sound justification.

[B] The output of the 2018 tsunami modelling consists of computed values at 6 locations around the Akkuyu site. The 2018 report does not explain the reason for prioritising these 6 locations. There is no justification provided that these 6 locations will allow the maximum tsunami runup in the Akkuyu NPP to be enveloped. The spatial distribution of runup results in the existing modelling from the 2018 study seem to show that the highest runups are obtained in the NE corners of the site bay, which were not covered in the 2018 study.

This PRT concern has been resolved by the DTHA2029 study, which now models the tsunami scenarios at 32 locations around the sites, using the revised water inlet and outlet design.



[C] Considering the orientation and shape of Akkuyu Bay and the location of the Babadil islands that partly reduce the energy of waves coming from the South, tsunami sources located W or SW of Akkuyu (notably the RUN-7 tsunami source zone) could lead to higher runups.

These sources were modelled in the 2017 study, with the original natural bay bathymetry, which led to the RUN-4 tsunami source zone being exclusively prioritised for further tsunami simulations with the new bay bathymetry. However, considering the significant impact of the offshore hydraulic engineering structures on the tsunami runup computed for RUN-4, tsunamis generated by the W or SW sources could be subject to higher local amplification, considering the modified bathymetry with the new offshore structures.

This issue has since been corrected in the DTHA 2023 report, which considers both RUN-4 and RUN-7.

Recommendations:

The PRT recommends:

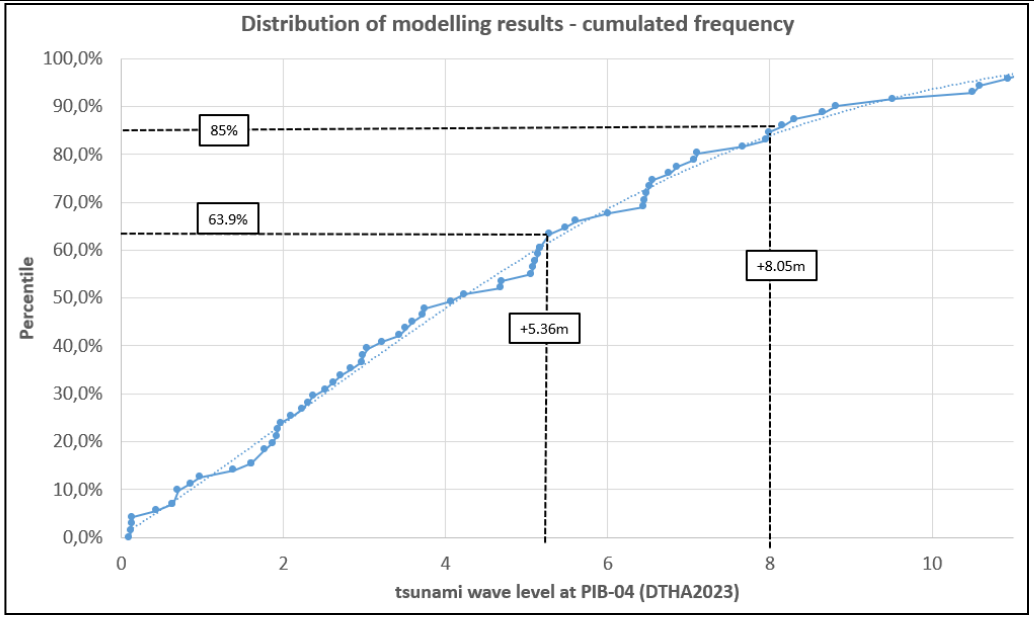
T1-15-1. High Priority. [A] Considering the significant consequences on the calculation of the tsunami DB, the PRT recommends to thoroughly review the differences between the rupture parameters used in DTHA 2017 and DTHA 2023 and correct them as necessary. Any changes to the rupture parameters should be covered by a careful and sound justification.

T1-15-2. High Priority. [B] It is recommended that Türkiye ensures that the specific locations points modelled (currently only 6) include the most penalising location in terms of tsunami runup and for instance adds additional modelling points in the N-E corners of the bay, where a splash/amplification effect is likely to increase the local wave height. *Recommendation T1-15-2 has since been solved with the DTHA 2023 report, which models tsunami propagation in 32 locations around the site.*

T1-15-3. High Priority. [C] Considering the orientation and shape of Akkuyu Bay and the location of the Babadil islands, it is recommended to model in more detail the tsunami sources located W or SW of Akkuyu (notably the RUN-7 tsunami source zone). A tsunami from these sources

	could be subject to higher local amplification. Modelling should use the revised local bathymetry and consider the offshore structures, as done for RUN-4. *Recommendation T1-15-3 has since been solved with the DTHA 2023 report, which considers both RUN-4 and RUN-7.*
T1-16	**Definition of the tsunami design basis (DB)** **Description**: The assessment of the tsunami hazards at the Akkuyu NPP site was carried out in several steps. In 2017, a deterministic and a probabilistic tsunami hazard assessment (DTHA and PTHA) were provided as part of the Site Parameter Report. Both reports were based on the modelling of 357 rupture scenarios spread over 8 tsunamigenic zones. They computed the expected peak coastal amplitudes, considering the original native bay morphology. The tsunamigenic zone generating the highest runup was 'RUN-4'. In 2018, a new report provided an update of the modelling. It used the planned new morphology of Akkuyu Bay and computed the tsunami parameters at six points around the offshore hydraulic engineering structures. This modelling update was provided only for the RUN-4 zone. It generated significantly lower runup values than those computed in 2017. The 2018 report defines the maximum probable tsunami elevation as a statistical combination of the runups from the 72 RUN-4 rupture scenarios modelled, combined with the annual total variability of the sea level. A similar approach is used in the DTHA 2023 report. **Assessment:** **[A]** For earthquake-induced tsunamis, the international consensus is that the hazard should be assessed by using either a deterministic hazard analysis or a probabilistic hazard analysis, and preferably both methods (see for instance IAEA safety guide SSG-18). In a DTHA, the DB tsunami is defined as the maximum water level from the range of all tsunami heights, from conservative numerical calculations for all the possible seismogenic sources. PTHAs are globally analogous to PSHAs but their use is not yet widespread. PTHA results are typically displayed as the annual frequency of exceedance of runup height values obtained through a logic tree approach. The method used in the 2018 report to define the DB tsunami consists of taking the mean runup value of the 72 simulations of RUN-4 (+3.71 m), and adding (only) 23% of the standard deviation (+0.66 m) and the total variability of the sea level (+2.08 m). The total DB tsunami runup is defined at +6.45 m (+3.71 m+0.66 m+2.08 m). This approach corresponds neither to a standard DTHA nor to a PTHA.

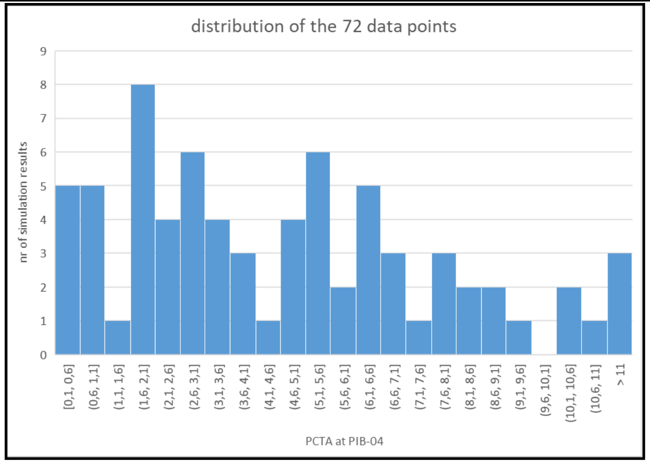
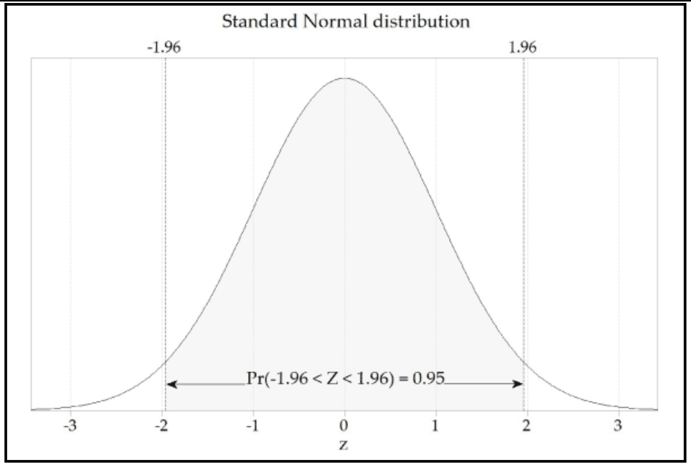
	A true PTHA should include all relevant seismogenic sources, not only RUN-4. It should provide reliable hazard curves for different confidence levels (mean, median and 84% quantile), based on a sound logic tree approach with due account of uncertainties, and extending with enough confidence well beyond an exceedance frequency of 10^{-4}/yr. A similarly unusual statistical combination is used in the DTHA 2023 report (+4.63 m+0.73 m+2.08=+7.44 m). A standard DTHA would have concluded that the DB tsunami should be based on the maximum water level from the range of all tsunami heights from numerical calculations for all the possible seismogenic sources. According to the existing modelling performed (cf. rupture scenarios KOERI-2-07c, KOERI-1-09, KOERI-1-07, RIZZO-10, etc.), this would lead to a DB tsunami around 6-7 meters higher than the current +7.44 m value. **[B]** The current definition of the DB tsunami is not 'unlikely to be exceeded with a high degree of confidence'. The graph below presents the distribution of the 4x18 PCTA data values modelled for the RUN-4 scenario, estimating the tsunami wave height at the PIB-04 point in the Akkuyu bay (in the NE corner of the inlet channel). Modelling results have been sorted from the smallest to the highest, and the percentile of each value has been calculated. Using a +5.36m value (value used in the Akkuyu DB tsunami, based on NP-064-05) would give only a 63.9% confidence that the tsunami runup will not be exceeded. This +5.36m value is exceeded by more than one third of all tsunami scenarios modelled. To have a 85% confidence of non-exceedance, a value of +8.05m should be used. To have a 95% confidence of non-exceedance (which is the intended target considered in NP-064-05), a value of +10.75m should be used.



The Non-Exceedance Probability (NEP) of 63.9% is lower than typical NEP levels used for external hazards (for instance the 85% and 95% levels commonly agreed in probabilistic assessments, or even the 100% level used in deterministic approaches in some countries, such as e.g. US NRC NUREG/CR-7223[124]) that recommend using the maximum credible calculated tsunami height as the DB tsunami.

The formula used to calculate the DB tsunami in NP-064-05 is 'mean + 1.96 × Std.Dev./√n', which assumes a normally distributed PCTA based on the Central Limit Theorem. Given that tsunami runup values represent real-world physical phenomena, it is crucial to validate the assumption that it follows a normal distribution, especially if extreme values (tails of the distribution, i.e. 95% NEP) are of concern for safety measures. However, the diagram below shows clearly that the distribution of the 72 values is not a normal distribution. Therefore, simplified formula based on the Normal distribution cannot be used without leading to very rough approximation.

[124] U.S. NRC, 2016. Tsunami Hazard Assessment: Best Modeling Practices and State-of-the-Art Technology, NUREG/CR-7223.

	  Note that the DTHA 2023 report concludes that additional safety margins may be considered and that another approach using Mean+σ (note PRT: corresponding to 82.2% NEP) instead of Mean+1.96xσ/√n (note PRT: 63.9% NEP) can be considered, i.e. increasing the NEP level in estimating the tsunami demand. **Recommendation:** **T1-16. High Priority.** The PRT recommends revising the methodology for defining the DB tsunami, to ensure it follows established international practice for DTHA and/or PTHA.
T1-17	**Landslide-induced tsunami** **Assessment:** Section 7.1 ‘Deterministic Approach’ of NUREG/CR-7223[125]**Error! Bookmark not defined.** states: *‘For many coastal locations, even for those without a nearby tsunami source region, these seismically generated tsunamis can be the source for the PMT. However, at other locations, local landslides capable of generating very large wave heights over a small area can more commonly represent the PMT source. At any location, both types of tsunamis should be evaluated for their potential to represent the PMT. In the absence of any statistical information regarding the location of*

[125] U.S. NRC, 2016. Tsunami Hazard Assessment: Best Modeling Practices and State-of-the-Art Technology, NUREG/CR-7223.

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past and future landslides, it should be assumed that any local landslides that have occurred in similar geophysical conditions as found near the NPP may in the future occur immediately offshore of the NPP. For example, if there is geological evidence of a submarine landslide along the shelf break at a location distant to the NPP but in a similar geophysical configuration, then that landslide should be considered to possibly occur at the shelf break immediately offshore of the NPP. This approach can be discarded only with a clear and convincing geological argument against permitting a change in the future landslide to a location closer to the NPP. Due to an overall lack in understanding of the evolution of large submarine landslides and the waves that they generate, a highly conservative approach must be employed to initially estimate the waves generated.'

PRT conclusions:

During the plant walkdown and subsequent discussions it was clarified that there is very low risk that a significant landslide-induced tsunami could occur following the (for instance earthquake-induced) collapse of parts of Babadil islets directly facing the Akkuyu site. This is due to the absence of evidence of past landslides on the seabed around the islets, as well as to the configuration and limited volume of the islets. This topic is considered closed.

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T1-18

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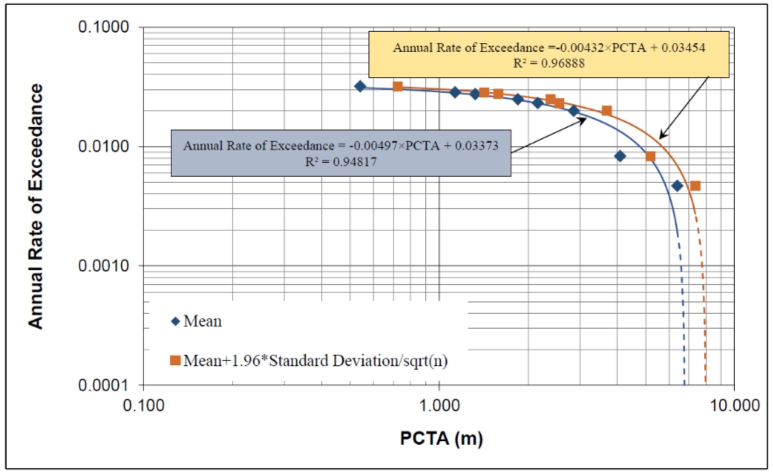
Extrapolation of the tsunami hazard up to 10^{-4}/year and beyond

Description:



A tsunami hazard curve is provided in the 2017 PTHA. Due to the limited number of 8 modelling points that define the tsunami hazard curve, and the absence of any modelling point for lower exceedance probability than 4.7×10^{-3}/yr, there is significant uncertainty on the extrapolation of the curve to lower occurrence probabilities.

The curve is extrapolated by a factor 47 on the return period of the last modelled point.

The tsunami hazard curve saturates below exceedance probabilities of 10^{-3}/year at around +8 m.

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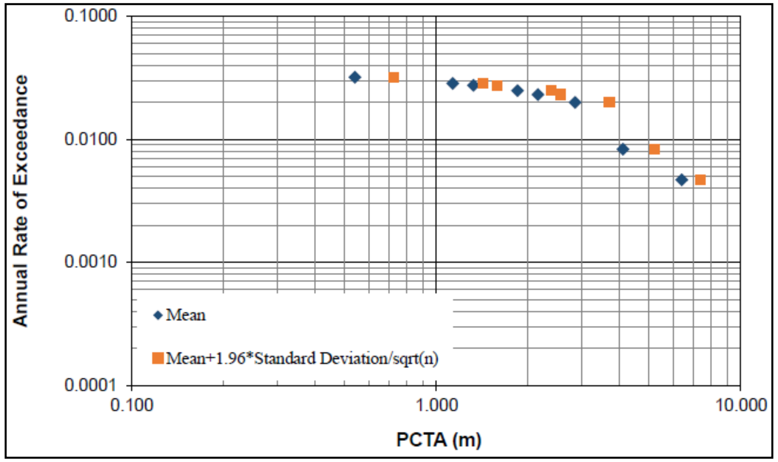
Assessment:

[A] The PRT doubts the reliability of the tsunami hazard curve for exceedance probabilities below about 4.7x10^{-3}/yr, i.e. the occurrence probability at which the hazard curves provided in the 2017 PTHA saturate. In general, tsunami hazard curves very rarely present complete saturation. When saturation is present, usually tsunami hazard curves saturate at much lower exceedance probabilities (typically below 10^{-5}/yr).



The raw distribution of 8 points from the 2017 PHTA report does not show any obvious saturation (cf on the left). The choice of an extrapolation curve that saturates seems arbitrary. As the seismic hazard does not saturate at least up to 10^{-5}/yr, there is no reason why the earthquake-induced tsunami hazard would saturate before this exceedance probability.

The hazard curve provided in the 2017 PTHA gives the impression that tsunamis higher than +8 m are impossible, whereas about 40% of the modelled tsunami scenarios for the RUN-4 zone lead to a runup higher than +8 m.

In total, 32 rupture scenarios modelled lead to runup values higher than +8 m. This shows that the probability of a tsunami runup exceeding +8 m cannot be zero.

[B] In general, there is very high uncertainty on the extrapolation of tsunami hazard up to 10^{-4}/yr and beyond (e.g. Gürpinar et al.[126], 2015), which is used as the DB tsunami in the probabilistic tsunami hazard assessment (PTHA). This in turn leads to significant uncertainty in the DB tsunami determination.

To assess the tsunami hazard corresponding to 10^{-4}/yr, a linear interpolation of the 8 computed values has first been made. This linear interpolation is relatively accurate for the lowest exceedance frequencies, but the inaccuracy becomes high (21% and 11% error) for the two

[126] Gürpinar A, Godoy A. R., Johnson J. J., 'Considerations for Beyond Design Basis External Hazards in NPP Safety Analysis', Transactions, SMiRT-23, Manchester, United Kingdom. August 10-14, 2015. Division IV, Paper ID 424.

	points with the lowest exceedance frequencies. This interpolation line has then been prolonged with a factor of 47 to a further lower exceedance frequency. Such an extrapolation so far from the distribution of the scenarios modelled unavoidably leads to very high uncertainty. Seismic hazard curves have been developed for the Akkuyu site up to very low exceedance probabilities. It is unclear why the same could not be done for tsunami hazard. A PTHA would anyway need to be based at least on modelling of all the 57 rupture parameters (which is not the case at the moment) with the new bay bathymetry. **Recommendations:** The PRT recommends that NDK do the following: **T1-17-1. High Priority.** [A] Clarify the inconsistency between the saturation of the hazard curve at +8 m and the existence of numerous modelling results above that value, and develop a tsunami hazard curve that better covers exceedance probabilities lower than 10^{-3}/year. **T1-17-2. High Priority.** [B] Reduce uncertainties over the tsunami hazard curve for low exceedance probabilities by adding modelling points for lower exceedance frequencies. **T1-17-3. High Priority.** [C] As several other PRT recommendations could impact the tsunami hazard assessment, it is recommended to derive updated tsunami hazard curves once the tsunami hazard assessment is considered up to date and integrates the other PRT recommendations. During the plant visit, the plant agreed that the current PTHA lacks data points, does not provides accurate and reliable results, and therefore is excluded from the current safety case. If this remains, then the three above-mentioned recommendations would not need to be addressed. If the PTHA was nevertheless used in a longer term future, the PRT viewpoint is that it would be important to first address the three above-mentioned recommendations on thorough improvements and develop a fully-fledged PTHA.
T1-19	**Acceptance criteria for margins against tsunami beyond the DB level** **Description**: There is currently no acceptance criteria for the margin to be provided beyond the DB tsunami and to avoid cliff edge effects. **Assessment:** Safety margins are important to offset the uncertainties inherent in tsunami hazard assessments, deriving from: • limits to the accuracy, quantity and period of time in which the historical data were collected,

	• uncertainties about the tsunami sources, tsunami modelling and other data used, such as bathymetry, • the reliability of runup heights. The design of the plant should provide for an adequate margin to protect items ultimately necessary to prevent large or early radioactive releases in the event of levels of natural hazards exceeding those to be considered for design, taking into account the site hazard evaluation[127]. IAEA safety guide SSG-67 'Seismic Design for Nuclear Installations' highlights that 'to demonstrate adequate seismic margin (for nuclear power plants), the reference review level earthquake in seismic margin assessments is typically defined by a factor of 1.4, 1.5 or 1.67 based on a peak ground acceleration corresponding to SL-2'. Russian standards apply a generic safety factor of 1.4 for margins beyond seismic DB to cliff edge effects. Gürpinar et al. (2015)[126] highlights that a margin factor of between 1.4 and 1.67 is commonly used for seismic hazard. For all other beyond design basis external hazards they recommend a standard safety margin factor of 2 for the key SSCs. Finally, they suggest higher-than-standard safety margin factors for coastal flooding hazards, considering the higher uncertainties associated with these hazards. When sufficiently accurate tsunami hazards curves are available (with reasonable uncertainty), the exceedance probability associated with the beyond design basis margin can be roughly assessed. In most countries, this exceedance probability is requested to be lower than 10^{-5}/year or 10^{-6}/year. The hazard at this low annual exceedance probability is usually linked to the targets for core damage and release frequencies (i.e. PSA results). **Recommendations** **T1-19-1: High Priority:** The PRT recommends that NDK define the acceptance criteria (if possible, in the form of a regulatory document) for the necessary margins beyond the tsunami DB (i.e. preventing cliff edge effects in the design). These criteria should be in line with international practices for external hazards in general, and for coastal flooding in particular.
T1-20	**Tsunami alert system and associated Emergency operating procedures (EOPs).** **Description**: The Republic of Türkiye has a warning system for earthquakes and tsunamis. There are 15 early seismic detection stations in the region where the Akkuyu NPP is located. The average seismic detection threshold in the region is M=1.5. All information is promptly and automatically

[127] IAEA, 2016. Safety of Nuclear Power Plants: Commissioning and Operation, SSR-2/2 Rev. 1.

	transmitted to stakeholders through various communication channels. A tsunami alert system also exists in the plant region, using tsunami detection stations based on sea level measurements in the Mediterranean Sea. The plant has emergency response procedures for external events, but no dedicated procedures dedicated to tsunamis. **Assessment:** An effective warning system should include a timely identification of the tsunami hazard, issuance of timely warnings and continuous and sustained awareness activities (training, drills and exercises). The plant has not yet decided whether to connect to the tsunami early warning system. Tsunami modelling (see 2018 report 'Tsunami Analysis Report for Akkuyu NPP with the Designed Structures') shows that there would be at least about 15 minutes available to react after initiation of the tsunami wave. This time could be decisive to take protective measures, once the tsunami is detected. To increase the available reaction time in the event of a tsunami, it may be useful to add sea level monitoring stations in key areas (around Northern Cyprus). Tsunamis can have specific consequences that may need specific emergency and accident management response instead of a generic response to external hazards. The lack of specific procedure may lead to delays in forecasting and responding to a tsunami. Having a dedicated procedure would also allow to the plant to perform more effective drills and exercises for tsunami scenarios. **Recommendations:** **T1-20-1.** The PRT recommends reviewing the suitability of using an early tsunami alert system at the plant. **T1-20-2.** The PRT recommends developing a procedure to deal with tsunami emergencies. The plant confirmed that the plant is now connected to an early tsunami alert system (based on both a seismic and a tsunami detection network), and that a specific procedure has been developed for responding to tsunami alerts. Therefore, *the PRT considers that both recommendations T1-20-1 and T1-20-2 are resolved.*
T1-22	**Protection against external flooding of means and facilities used in emergencies and accident management (elevation of storage places for equipment, emergency shelters, crisis centre etc.)** **Description**: The Akkuyu NPP has a certain amount of mobile or semi-mobile equipment to be used in DEC situations. This includes , for the whole site:

- 4 mobile pump units (MPU), which are stored outdoor near the pumping station on the platform, at an elevation of around +10.85 m.
- mobile heat exchangers (KAA25) also stored at their place of use on the platform, at an elevation of around +10.85 m.
- 4 alternative diesel generator stations (ADGS), that will be stored outdoor between the plant units, on a concrete platform at an elevation of around +10.85 m elevation.

Several underground emergency shelters for plant staff (building entrance around 20 cm above the +10.50 m platform level):

- 2 shelters with 600-person capacity (04-05UYX)
- one shelter with 1 200-person capacity (01UYX).

Several buildings are intended for use in case of emergencies or accident management, including:

- an underground crisis centre (02UYX, with its entrance at around +10.85 m)
- 4 backup buildings for emergency management teams (located on the North of each unit, at an elevation +10.85 m).

Assessment:

The elevation at which this infrastructure and equipment is situated does not provide additional protection against external flooding compared to SSCs used in lower levels of defence in depth.

Emergency facilities should remain available in credible DEC situations. External flooding has the potential to create common cause failures and lead to cliff edge effects. Due to the location and elevation of the above-mentioned facilities, external flooding may challenge multiple levels of defence in depth at the same time.

Items important for safety should be located taking into consideration common cause failure mechanisms generated by hazards[128]. Plant operators should make sure that facilities, tools, equipment to be used in an emergency and in the accident management programme is unlikely to be affected by, or made unavailable by accidents[129]. Emergency facilities should be usable under all emergency conditions[130], including flooding by tsunami.

[128] IAEA SSR-2/1 Rev.1: 'Safety of Nuclear Power Plants: Design', 2016.

[129] IAEA SSR-2/2 Rev.1: 'Safety of Nuclear Power Plants: Commissioning and Operation', 2016.

[130] IAEA GS-G-2.1: 'Arrangements for Preparedness for a Nuclear or Radiological Emergency', 2007.

	Additional details can be found in the related findings in Topics 2 and 3. **Recommendation:** **T1-22-1. High Priority.** The PRT recommends increasing the robustness against external flooding (for example, relocate to a higher elevation as appropriate) of the equipment and facilities using the means necessary during emergencies and accident management.
T1-23	**Elevation of numerous safety-related SSCs in basements below +10.50 m level** **Description**: The plant platform is built on a flat platform (+10.50 m) higher than the DBF level (+7.44 m, to be updated considering the high priority PRT recommendations impacting the DB tsunami). The plant considers that this protects the plant SSCs sufficiently from external flooding. However, a large number of SSCs important for safety are partly or fully located in basements below the +10.50 m level. Some of these SSCs have key elements that are located in deep basements below the DB flood level. **Assessment:** For new nuclear installations, all SSCs ultimately necessary to prevent core damage, an early radioactive release or a large radioactive release should be located at elevations higher than the design basis flood[131]; alternatively, adequately engineered safety features (e.g. watertight doors) should be in place to protect these SSCs. For new nuclear installations, a dry site is preferred over a site protected by permanent external barriers. If the dry site concept cannot be applied to all SSCs important for safety, the layout should include permanent flood barriers with an appropriate design basis and adequate margins. The fact that some SSCs important for safety are located below the DBF leads to the conclusion that the Akkuyu NPP does not qualify as a ‘dry site’. This stresses the importance of the ‘volumetric protection’ of all safety-related buildings, including galleries leading to these buildings. The plant indicated that it intends to install watertight doors in the openings to relevant safety-related buildings. However, due to the actual state of the construction during the plant visit, the PRT was not yet able to review the volumetric protection of plant safety-related buildings against water ingress.

[131] IAEA, 2021. Design of Nuclear Installations Against External Events Excluding Earthquakes, SSG-68.

	Recommendation: **T1-23-1:** The PRT recommends that NDK check that the plant's measures to prevent water ingress into safety-related buildings and underground galleries are robustly designed and implemented.
T1-24	**Consequences of possible rain infiltrations through turbine hall roofs** **Description**: The NR states (p. 82): 'The external initiating event 'accumulation of a pool on building roofs due to extreme precipitation' leads to the accumulation of a pool on the roof of 10UMA building. The most likely consequence of this scenario will be roof leak in the 10UMA building. It is conservatively assumed that this will lead to the failure of equipment located in that building.' **Assessment:** The turbine halls (UMA) contain the following safety-related SSCs: - main steam piping (LBA50) - turbine bypass system (MAN) - auxiliary feedwater system (LAH) (including auxiliary feedwater pump (LAJ); auxiliary feedwater pipelines, AEFWP recirculation and warming-up pipelines, including valves) **NDK clarified all open questions.** **The issue is solved.**
T1-25	**Sewer channels** **Description**: The Akkuyu site is protected against rain flooding from the surrounding catchment areas by three large peripheral storm sewers/canals (01-02-03UZN). The largest of them is the Eastern canal, which collects and ultimately discharges rain water into the sea at an elevation of +2.5 m. The design documentation on the site protection engineering structures (topic1-93.pdf) states in chapter 'Wind waves effect on discharge canals operation assessment': 'Eastern discharge canal КВ: According to the Akkuyu NPP main and auxiliary facilities layout conditions not only outlet section, but also a large part of the canal is located at lower elevations, which will inevitably lead to the canal overflow and discharge

	pour out from the open sections to the adjacent territories. Taking into account this circumstance, the alternative of implementation of eastern discharge canal as a closed tube all along at elevation below 15.0 m was considered.' The PRT was informed that higher walls will be built. A map was provided showing that the location of the joint discharge of canals 02UZN and 3UZN coincides with one of the areas where the highest tsunami runup height is expected (i.e. the North-East corner). **NDK clarified all open questions.** **The issue is solved**.
T1-26	**Design basis for extreme precipitation** **Description:** In table 4 of the NR, the maximum daily precipitation (10^{-4}/y) is indicated to be 302.7mm. However, this value is lower than the value at nearby Anamur station (page 63 - 314.22mm). Table 4.6.4 of the Site Parameters Report below indicates that the maximum expected precipitation is 321mm/day for a return period of 10 000 years, with a confidence level of 95%. **NDK clarified all open questions.** **The issue is solved**.
T1-27	**Extreme winds and tornadoes** **Description:** The Preliminary Safety Analysis Report (PSAR; AKU-PSAR040203-BQB0001) recommends using design values for wind loads corresponding to the maximum wind speed recorded at the representative meteorological station (MS) – Anamur MS. The design value for high wind (5 s gusts) is 76.1 m/s for the 10^{-4} annual exceedance probability. The 10-min averaged wind speed is 42 m/s for the annual exceedance probability (AEP) of 10^{-4}, based on the meteorological data recorded at the Anamur MS between 1967 and 2015. Safety Category I buildings and structures (classification according to PiN AE-5.6) are designed to withstand extreme wind loads of 76.1 m/s corresponding to 5 s gusts and AEP of 10^{-4} per year.

	According to the PSAR, 13 tornadoes were detected within 100 000 km^2 of the Akkuyu NPP between 1981 and 2015. Records include data from the Turkish State Meteorological Service, Directorate for Disaster Cleanup Operations, European Severe Weather Database and Karaman & Markovsky (2014)[132]. The DBT was classified as F2 on the Fujita scale. The parameters of the DBT were determined from the catalogue by Karaman & Markovsky (2014): maximum horizontal rotational speed of 60 m/s, maximum horizontal translational speed of 15 m/s, and maximum air pressure drop 44 hPa. Missiles are not considered in the NPP design since F2 tornadoes only lift up light objects, which are said to be not capable of damaging the NPP buildings/SSCs. The annual probability of a DBT passage through the district where the Akkuyu NPP is located is $7.85*10^{-7}$. Safety Category I buildings and structures are designed to withstand an intensity of F2 DBT. **Assessment:** Safety-critical category I buildings and structures (classification according to PiN AE-5.6) are designed to withstand extreme wind loads of 76.1 m/s corresponding to 5 s gusts and AEP of 10^{-4} per year (DBEW) and an F2 tornado on the Fujita scale (DBT). The annual probability that a design basis tornado will pass through the district where the Akkuyu NPP is located is 7.85*10-7. Category II buildings and structures (as per NP-031-01) are designed to withstand wind impact with recurrence once in 100 years, category II buildings and structures as per NP-031-01 not impacting safety, as well as category III buildings and structures as per NP-031-01 will withstand wind impact with recurrence once in 50 years. According to WENRA (Guidance Document Issue TU: External Hazards, Guidance on Extreme Weather Conditions), 'The exceedance frequencies of design basis events shall be low enough to ensure a high degree of protection with respect to external hazards. An exceedance frecquency, not higher than 10^–4 per annum shall be used for the design basis'. The DBT and DBEW are calculated in accordance with relevant national and international requirements and with good international practice. **Recommendations:** **T1-27-1** The PRT recommends that NDK define the necessary safety margins beyond the DBT.

[132] Karaman, A. & Markovsky, P.M., 2014. Climatology of tornado in Türkiye. Monthly Meteorological Journal, 142: 2345-2352.

T1-28	**Extreme precipitation – drainage system** **Description:** The report 'AKU-P020301-BAA0002 – Site protection engineering structures' reviews the design of the site protection against flooding by storm rainwater. The characteristics of the precipitation events used for this check are given in a table with probabilistic depth-duration curves for the Akkuyu site. The values in the table do not match the site hazard curves or the values mentioned in the NR. The eastern discharge canal KB (02ZN) passes from the north to the south between the eastern boundary of the operating nuclear island on which the main structures of the Akkuyu NPP are located and the surrounding hills. It runs over more than 1 400 meters and collects water from the largest of the surrounding catchment areas. The calculations in report AKU-P020301-BAA0002 seemed to indicate a low margin between the water depth in this canal and the top of the walls of the canal. At numerous places along its route, there is less than 20% difference between the water depth and the wall height. As showed in the table NR, p. 43, the water depth exceeds the wall height in some parts. **NDK clarified all open questions.** **The issue is solved.**

Topic 2: Loss of electrical power and loss of ultimate heat sink	
Finding nr	**Peer review team finding**
T2-1	**Reliability of active safety systems** The Akkuyu NPP design has 2 x 100% active safety trains, each powered by a single emergency diesel generator in case of LOOP. In the event of a total loss of AC power, the passive heat removal system will need to be used and a mobile diesel generator unit will need to be connected to ensure that power will continue to be supplied after the expiry of the 72-hour batteries. The availability of only two EDGs per unit raises concerns about the reliability of the emergency power supplies and the resulting reliance on the passive systems and mobile means, particularly in situations in which one EDG system is out of service for maintenance or repair, which is a main contributor to the associated risk. This concern has now been addressed by the inclusion of an additional diesel generator. This additional generator can be used to replace the diesel generators taken out of service for online maintenance, thereby retaining the 100%

	redundancy during maintenance. Given that there is only 2 x 100% redundancy on the safety systems, the PRT recommends that special attention be paid to monitoring the reliability and availability of the systems and components throughout operation. The PRT also recommends that concrete measures be taken to improve the reliability of active safety systems. Special attention should also be paid to assuring the operability of the other support systems needed for the safety functions to be able to fulfil their mission (e.g. the I&C, cooling, HVAC, etc.). In accordance with IAEA SSR-2/1 Rev. 1, Requirement 28, it is indispensable that limitations on the timing and duration of preventive maintenance and online repair, and related operational restrictions in the event that safety systems are unavailable, be specified in the OLC on the basis of an appropriate risk analysis. This applies to all safety systems, not just the EDGs. NDK is implementing actions to address this recommendation.
T2-2	**Loss of ultimate heat sink** According to SSR-2/1 Rev. 1, Requirement 53: ***Requirement 53: Heat transfer to an ultimate heat sink*** ***The capability to transfer heat to an ultimate heat sink shall be ensured for all plant states.*** *6.19A. Systems for transferring heat shall have adequate reliability for the plant states in which they have to fulfil the heat transfer function. This may require the use of a different ultimate heat sink or different access to the ultimate heat sink.* *6.19B. The heat transfer function shall be fulfilled for levels of natural hazards more severe than those considered for design, derived from the hazard evaluation for the site.* In line with the lessons learned following the accident at Fukushima Daichi NPP, the SSCs providing the normal connection to the primary UHS should be robust and highly reliable. The design of the UHS capability can be improved by assuming the loss of normal access to the UHS in normal operation to be a DEC event (IAEA TECDOC 1791, WENRA Report on safety of new NPP designs). The PRT was informed that the failure of both trains of the component cooling system (KAA) and/or the secured cooling water system (PE) during normal operation, resulting in loss of normal connection to the ultimate heat sink, has a frequency of 1.9 x 10^{-3}/year (it is not clear if this includes the contribution from external hazards or only internal failures). This is not an indicator of high reliability of the systems and is unacceptably high for the event to be considered in the category of DEC (or BDB in the Akkuyu project terminology). **The PRT recommends that the event be considered as a PIE in the design basis. It also recommends ensuring that the analyses provided in**

	chapter 15 of the safety analysis report be conducted in line with the appropriate rules and requirements for analysing design basis events. This is particularly important as the operability of the normal connection to the UHS is needed for the core cooling (primary and secondary side) as well as for the SFP cooling, meaning that the effective response of all safety systems to any disturbance is limited by the reliability of the heat removal chain. This gives the opportunity to apply the more conservative approach for deterministic safety analyses in order to assure a reliable operability of the system. NDK has requested actions and the designer has initiated actions to address concerns about heat removal during shutdown plant states. These are currently ongoing. But this does not address the general concern about the ultimate reliability of the connection to the ultimate heat sink.
T2-3	**Storage and connection of mobile means** According to IAEA SSR-2/1 Rev. 1, paragraph 5.21A: *5.21A. The design of the plant shall also provide for an adequate margin to protect items ultimately necessary to prevent an early radioactive release or a large radioactive release in the event of levels of natural hazards exceeding those considered for design, derived from the hazard evaluation for the site.* Mobile equipment to manage beyond design basis accidents (Russian terminology) is available. An alternative, air-cooled mobile diesel generator (ADGS) of limited power (2 MW), is available for each unit to support the emergency power supply system (EPS) in the event of failure of the two emergency diesel generators (SDGS). There is also a mobile diesel-driven pump unit (PEC10) with which a mobile heat exchanger (KAA25) can be fed with seawater in order to maintain long-term active residual heat removal from the SFP and from the reactor. The diesel-driven pump unit (PEC10) and ADGS could also be transported to the connection point and used for another unit should it be required. It is envisaged that the mobile devices in question will be permanently stored at the designated locations where they will be used, at elevation +10.5 m, but not permanently connected. When their use is required, they must be connected to the designated connection points, which requires up to 6-8 hours for the ADGS and around 4 hours for the PEC10 and KAA25. Their use is not required within the first 72 hours of the accident situations for which they are needed. The PRT recognises the possible advantages of storing the mobile means at or close to their place of use, as some external hazards may have the potential to jeopardise the ability to move mobile means on the site. On the other hand, storing them permanently at the +10.5 m level could lead to their vulnerability to some external hazards that may also impact the safety systems at lower levels of DiD, for example climate-induced external hazards or floods/tsunamis. The PRT understands that the mobile means have a role to play in the management of accidents in which the target is to prevent fuel damage, as well in the management of severe accidents, and they are therefore likely to be on the list of equipment ultimately required to prevent large or early releases. Unless it can be demonstrated that the mobile means are not

	ultimately required to prevent large or early releases, they should be protected against beyond design basis external hazards. **Supporting the assessment in Topic 1 on possible external hazard impacts, and supplementing its recommendations based on the assessment results, the PRT also recommends a thorough analysis of the vulnerability to BDB external hazards, of both the mobile means at their designated storage locations, and of their connecting points to the plant systems. If any vulnerabilities are identified, action to ensure adequate protection must be taken.**
T2-4	**Reliability of passive injection of second stage hydro accumulators** Part of the efforts to introduce passive features into the safety concept of the Akkuyu NPP design, has been to install two sets of hydro accumulators that are capable of providing primary side (and SNF pool) feed. There are three sets of hydro accumulators in total. The first stage (HA 1), consisting of four vessels (inventory 50 m^3 each), is a set of standard pre-pressurised fast-acting hydro accumulators, to flood the core in the event of a large LOCA. The second stage (HA2) comprises eight pieces (four pairs) of accumulators (inventory 120 m^2 each) kept at atmospheric pressure, which feed their contents into the core and/or SF pool (SFP) under the hydrostatic head generated by their height above the reactor/SFP. The third stage (HA3), consisting of 12 vessels (inventory 60 m^3 each), is located in the containment annulus. These are also at atmospheric pressure and rely on hydrostatic head to deliver their contents to the reactor or SFP. They can be connected manually to the discharge line of the HA2 accumulators and are used to provide additional water as needed. The RPV has four nozzles to enable emergency feed by the hydro accumulators. Two are located in the RPV downcomer and two in the upper part of the RPV injection. Half of the hydro accumulators of each stage (HA-1, HA-2 and HA-3) inject their coolant inventory into the RPV downcomer and the other half into the RPV upper part. The injection path into the RPV for both HA1 and HA2 is principally passive and does not require any electrical power supply, i.e. the isolation is achieved by two in-series installed check valves that are separating low from high-pressure segments. The first check valve sees the full RCS pressure on one side (i.e. about 16 MPa), and the HA1 pressure on the other side, i.e. about 6 MPa. The second check valve leading to the HA1 sees generally the same pressure on both sides, i.e. the standby pressure of HA1 (6 MPa). In between these two check valves, the feed line from the HA2 is connected and isolated by a single check valve, which sees on one side the pressure in the line from HA1 (i.e. the HA1 standby pressure of 6 MPa), and on the other side the atmospheric pressure plus the hydrostatic pressure from the water level in the HA2. When the operation of the HA2 is required, the ‘double check valve’ on a control line connecting the cold leg of the RCS with the top of the HA2 will open (when the pressure in the primary circuit is below 1.5 MPa), effectively equalising the pressure between the upper part of the HA2 and the RCS. As the HA2 is located at a higher elevation (the bottom of the HA2 is at the 26 m level and the RPV flange at about 15 m), in a case of the pressure-equalising line being opened (by the ‘double check valve’), there will be a hydrostatic pressure of the water column for the injection into the RPV/cold leg. In the best-case scenario, the hydrostatic pressure of about 0.15 MPa could be envisaged. This pressure

	would be available to (force) open the above-mentioned check valve. Considering that the check valve will experience a backpressure of 60 bars for a prolonged period of time (up to 18 months), any accumulation of boron crystals (from borated RCS coolant), dirt, or even a slight corrosion, might hinder or prevent the opening of the check valve by such a low hydrostatic pressure. The check valve, under the current arrangement, does not appear to be testable in operation, i. e. until the HA1 is isolated and the connecting line drained. Nor is any operational testing envisaged, according to the information received by the PRT. The testing is carried out only during the refuelling outage (testing at the Kudankulam NPP was mentioned to the PRT as 'evidence' for the check valve reliability). This raises concerns about whether the operation is reliable, as if not, it would potentially undermine the 'passive' feature of this system. As this is the key element of the safety concept for the feed and cooling of the primary side, the PRT consider this a major potential vulnerability, especially because in the event of a LOOP (and maybe an induced loss of integrity of the RCS) there would be no other possibility to assure cooling on the primary side. The feed from the HA2 and HA3 into the SNF pool, as understood by the PRT, bypasses the above-mentioned check value. It still relies on the hydrostatic pressure/gravity feed from about 26/23 m elevation (bottom of the HA2/HA3 respectively) to 16 m elevation (upper level of the minimum water level in the SNF pool). It requires the operator to open several isolation valves, as well as air intake valves to allow containment air to enter the upper plenum of the HA2/HA3. All the valves required for these operations are fed from the 72-hour battery, assuring their operability even in the event of an SBO. **The PRT recommends that NDK request a robust demonstration of the reliability of operation of the check valve in question when it is required or the implementation of measures to ensure the reliable functioning of the system.**
T2-5	**Multiple use of active safety systems** A key lesson learned from the accident at Fukushima Daichi NPP was the need to increase the independence between different levels of defence in depth (DiD). This is reflected in the WENRA safety objectives for new nuclear power plants and in IAEA SSR-2/1 Rev. 1. According to IAEA SSR-2/1 Rev. 1, paragraph 4.13A: *4.13A. The levels of defence in depth shall be independent as far as practicable to avoid the failure of one level reducing the effectiveness of other levels. In particular, safety features for design extension conditions (especially features for mitigating the consequences of accidents involving the melting of fuel) shall as far as is practicable be independent of safety systems.* As stated in IAEA TECDOC 1791, *The correct implementation of the requirements [of IAEA SSR-2/1 Rev.1] implies that the multiplicity of the levels of defence is not a justification to weaken the efficiency of some levels relying on the efficacy of others. In a sound and balanced design, SSCs of each level of defence are characterised by reliability commensurate to their function and their safety significance.* *Multiple consecutive levels of protection achieve the objective of defence in depth if, following the failure of one level of defence, the*

	subsequent level would not also fail for the same cause. For this reason, SSCs serving different levels remains one of the main factors to threaten the overall efficiency of the defence in depth concept. During its assessment, the PRT noted that some safety systems have multiple uses spanning several different levels of DiD, including normal operation, which is not in line with the intent of the above requirements. In cases where an operational function is supported by a safety system, it causes a normally-in-standby system to operate continuously, increasing wear and tear and the need for system maintenance. This may lead to some components of the system not being available when required to fulfil its safety functions. **Acknowledging the importance of a sophisticated system of DiD levels in order to assure nuclear safety in all operational conditions based on the technical design to be realised in the Akkuyu NPP, the PRT recommends developing and implementing a comprehensive DiD concept describing the technical features and organisational measures, as well as the assurance of independency between the different levels. This recommendation is linked to the findings under topic 3 on severe accident management and to special finding nr 4 under topic 3.** A DiD concept report has been provided by the licensee to NDK and is under NDK review. If the report is able to justify the categorisation of the safety/support systems to a satisfactory degree, this recommendation could be considered resolved. **The need to assess the potential consequences of the use of SSCs for multiple purposes and in different levels of DiD applies to all concerned SSCs.**
T2-6	**Reliability of the passive heat removal system (PHRS)** The design concept of the Akkuyu NPP, due to a low level of redundancy and no diversity of the active safety systems, and the fact that there is no emergency feedwater system envisaged, places a disproportionate and heavy reliance on the PHRS to remove the heat and cool the core in a case of a spectrum of sequences. The PHRS is a (largely) passive system and in the event of an SBO its operation is initiated by opening the flaps of the PHRS air intake due to loss of electrical power to the solenoid drives keeping them closed. The louvres controlling the PHRS airflow are also operated passively, based on the pressure in the steam-water circuit, and only require electrical power, supplied by the 72-hour batteries, to force them open when switching the system into cooldown mode. The PHRS has to continue to operate for the duration of the SBO (72 hours per analysis) to prevent the core from overheating. Section 15.6 of the Preliminary Safety Analysis Report was provided to the PRT and additional explanations were provided during the team's visit to Ankara. The operation of the PHRS was analysed for the most critical initiator, an SBO and induced loss of heat sink with RCS intact. According to the results, the pressure in the RCS and in the secondary circuit would not exceed the design limits, and the fuel cladding and the fuel (pellets) temperature would remain within the acceptance criteria. While those results support the conclusion that the safety parameters, even for a BDBA, are met, they are directly dependant on the actual capacity and reliability of the operation of the PHRS. The PRT was informed that the functionality of the PHRS itself was modelled by a specially developed computer code. The code was validated experimentally against

	results from a half scale, partial mock-up in one of the supplier’s research facilities. The experiments were conducted at reduced operational conditions, and the model was used to extrapolate to full-scale PHRS geometry and operating conditions. Commissioning tests in a real-scale PHRS at Kudankulam plant were carried out, which also provided additional data for validating the computer code, but full power operating conditions could not be achieved during those commissioning tests. **Recommendation: Due to the high importance of the PHRS for the safety justification for the Akkuyu NPP design, the PRT recommends a thorough assessment of all the analyses undertaken.** The PHRS is intended as an additional system to cover the DEC/BDBA events. Considering that the frequency of the loss of heat sink for the Akkuyu NPP design is above 1E-3, and that the PHRS will be the only remaining possibility to remove the heat, it is questionable whether the assumption that all four trains of the PHRS are operating in parallel, as well as best-estimate analysis assumptions, are acceptable. Scenarios such as failure to open the intake flaps (e. g. due to mechanical blockage by dirt, foreign objects, etc.), which would lead to a loss of one or more trains also need to be considered. Furthermore, clogging of some of the tubes in the PHRS would reduce its capacity. **A thorough review, followed by conservative DBA analyses (as opposed to the best estimate, which is acceptable for a DEC/BDBA) is recommended. This should include envisaged maximum allowable tube plugging in the SG and in the PHRS itself. Such analyses must include not just the PHRS steam-water cooling loop, but also the PHRS air-cooling loop.** With the diameter of PHRS air ducts being around 3 m and a maximum air flow needed (according to information received by the PRT) in excess of 35 kg/s, this leads to velocities of more than 10 m/s in the ductwork. Such velocities, through a densely packed air heat exchanger, through louvres and an uneven configuration of the ductwork, lead to (large) pressure drops in the duct. The fact that all four ducts come together at the top of the containment, and the air from all four PHRS trains flows through a common space (with the air trap requiring a 90/180 degree change of direction of the flow), with the discharge area of no more than around 25 m2 (estimated from schematic diagrams in the absence of the detailed drawings) leads to even higher velocities and pressure drop. As the driving force for the air circulations is the temperature gradient, when this is high (e.g. SG steam at 275 °C at the initiation vs 50 °C ambient temperature) it may be expected to be able to overcome the resistance. However, as the PHRS cools the SG, and the temperature drops to 150 °C (which is said to be the lowest temperature when the PHRS is operational), some difficulties in the air flow may arise. **The air flow and the pressure drop analysis across the full spectrum of operational conditions needs to be corroborated and values independently confirmed for the credit to be taken for the PHRS operation in a case of SBO.**
T2-7	**Non symmetrical injection of the JNA system into the core** The JNA system is designed in such a way that one train’s high-pressure injection is made into two of the cold legs and the other train’s high-pressure injection is made directly into the RPV down comer (joining a common injection pipeline used also by the hydro accumulators). The loss of water injected in case one or the other train fails (of a total of two), might have very different consequences. In the event of a LOCA

	caused by a break of the connecting pipe to the RPV shared with the hydro accumulators, it is not just the JNA train that would be lost, but also the hydro accumulators that use the same flow path. The fact that the flow paths are not symmetrical may lead to a very different plant response to specific initiators and JNA failure combinations. **The PRT recommends that a detailed analysis be carried out (or reviewed if one already exists) to assess the most critical conditions following the loss of either one of the trains of the JNA system and its effects. It should be demonstrated that in any combination of an initiator with a failure of either train of the JNA, the function of the system would be ensured and the design conditions to protect the core would be fulfilled.** The NDK has requested documentation from the licensee to support the justification of this design feature. The documentation is currently under review by the NDK. If the outcome of the review is satisfactory, this recommendation could be considered resolved.

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T3-1	**Description:** Several questions were raised regarding the requirements and content of the AM procedures and guides at the Akkuyu NPP during the NR review stage, which required various clarifications in both online and offline discussions with the NDK and the Licensee. The related discussions took place throughout the entirety of the review process and were eventually concluded with the PRT site visit to Akkuyu. During the site visit NDK informed the PRT that they had just received the first version of the EOPs, BDBAMGs and SAMGs, and they were still waiting for some additional information to be able to continue their review process. The NDK's first impression is that the procedures and guidelines look structured and complete. Some modifications are still expected based on the PSA Level 2 results. A new version of the PSA Level 2 analysis was expected to be submitted to NDK in 2024. Finalised and validated versions of the procedures and guidelines were said to be ready before the fuel loading. The operator trainings have begun, using the current versions. The PRT visited the simulator during the site visit and saw the current versions of the EOPs, BDBAMGs and SAMGs that are used for operator training. It was observed that there were clear criteria and procedural steps for entering SAMGs from the BDBAMG and that the SAMGs include one general part for the plant operators and more detailed guidelines for the plant emergency team responsible for deciding on action during a severe accident. The PRT team also looked at the overall structure of the guidelines and the more detailed guidance on containment pressure management. Based on the discussions it was concluded that the EOP and SAMG are based on the requirements documented in international and Russian Federation reports, guides or recommendations listed below, and on the relevant chapter of the Safety Assessment Report of the Akkuyu NPP. **Requirements of:**

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	• IAEA SSR-2.2 • NP-001-97 • NP-082-07 • OPR-III-RG-DO-07-004-2021 • OPR-III-RG-DO-07-006-2021 • IAEA SSG-70 Operational limits and conditions and operating procedures for nuclear power plants • IAEA SSG-54 Accident management programs at nuclear power plants • RB-023-23 Recommendations for development, structure and content of design-basis emergency operating instructions at nuclear power plants • RB-102-15 Recommendations for the structure and content of the beyond design-basis accident management guidelines, including severe accidents General structure of the accident management programme document:

EOP	**DBA** Emergency protection, failure to comply with safe operating limits and/or safe operating conditions	**EOI** RP emergency operating instruction	Part 1. Event-based Part 2. Symptom-based
	BDBA Failure to comply with Critical Safety Function (CSF)	**BDBAMG** Beyond design-basis accident management guide (SBEOI format)	
SAMG	**SA**	**SAMG**	

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	Fuel damage or spent fuel assemblies	Severe accident management guidance

Safety Assessment Report of the Akkuyu NPP:

- The list of accidents presented in Chapter 15
- The general structure of the emergency documentation is given in Section 13.3
- Emergency procedures include the monitoring of non-exceeding of safety limits given in Chapter 16

Validation of the emergency operation procedures and guidelines shall be carried out on the Full-scale Simulator (FSS) of the Akkuyu NPP according to individually developed validation programmes. Correction shall be made to the developed documents after their validation.
The emergency operating procedures and guidelines shall be verified according to the individually developed verification programmes.
In conclusion, the PRT could see progress in the development of the EOPs, BDBAMGs and SAMGs and there are plans to finalise and validate them before the fuel loading. This is an important issue for NDK to follow up. Justification for and validation of these procedures must be submitted to NDK and reviewed before the fuel loading.

Deadlines for developing and sending an accident management document to NDK:

Stage 1	Stage 2	Stage 3	Stage 4	Stage 5	Stage 6
Development of version 1	**Development of calculation justifications**	**Development of version 2** Based on documents developed at Stages No. 1 and 2	**Development of programmes for verification and validation**	**Validation and verification**	**Development of final versions** Based on documents and reports received

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				Report preparation following validation and verification	following Stages No. 4 and 5
The EOP and SAMG document was sent to NDK on 22.1.2024	8 months before the start of FAC	6 months before the start of FAC		Before the FAC stage	No later than 4 months after the end of the pilot commercial operation

According to the Akkuyu Nükleer JSC, ‘The specified documents are currently at Stage 2 and are ready for dispatch to NDK under the submission of documents to obtain a license for operation’.

Assessment:
The information provided by NDK and the Licensee filled in the gaps and answered most of the questions raised by the PRT during the stress test process. During the visit to the simulator on the site and the brief review of the SAMG, however, it was observed and discussed with the representatives of the Licensee that it seems that certain instructions have set points above the design pressure of the containment. While it not necessary to assume that the containment fails immediately after exceeding the design pressure, it was considered difficult to justify delaying depressurisation actions only until after the design pressure of the containment is exceeded. In order to ensure that such actions are carried out in a timely manner, and before exhausting or reducing the safety margins, the set point of such instructions should be changed to be at or below the design pressure of the containment, or the related instructions should be justified.

Recommendation:

Instructions in the SAMG must be subject to in-depth regulatory review and set points should be at or below the design pressure of the containment in order to ensure that the operator can take action before containment integrity is threatened, or the exceedance of the design values must be robustly justified by the Licensee.

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T3-2	**PSA scope and important contributors** **Description:** PSA Level 1 and Level 2 are required by Turkish regulations and are currently under review by NDK as part of the operation licensing. PSA Level 1 and Level 2 are used in the development of EOP, BDBMAG and SAMG. There are no strict criteria for the acceptance of the results of PSA in licensing the plant. Instead, values are specified as targets, for which efforts should be made to ensure that are reached. These values have been adopted from the Russian regulation **NP-001-97 and include:** - For Level 1, that the core damage frequency does not exceed the value of 1E-5/year - For Level 2, that the probability of evacuation beyond the precautionary zone does not exceed the target value of 1E-7 /year. The urgent protection zone is defined as a radius equal to 20 km around the plant, and corresponds to the radius where no evaluation is needed, taking into account radiological dose criteria for protection measures as provided in IAEA GSR Part 7. These include a maximum effective dose and an equivalent dose to the fetus equal to 100 mSv in 7 days. However, it should be highlighted that the target value of 1E-7 /year is not required by law to be fulfilled, but is defined as a target, meaning that the licensee is required to implement all reasonable actions (modifications, design solutions, administrative measures, etc.) that will bring the facility closer to being below this target value. - **Assessment:** No criteria have been defined yet for avoiding early releases. As provided in the discussion during the mission, NDK plans to discuss a possible requirement for the operator to introduce such a criterion as a condition to licensing the operation. (Ref T1-01-1) **PRT conclusion:** **This issue can be considered as closed.**
T3-3	**Environmental qualification** **Discussions:** Before the visit to NDK in Ankara, the PRT requested information about the adaptation of the safety features used for severe accident management for the environmental conditions for which they are intended to be used. During the plant visit NDK informed the PRT that it had received the overall environmental qualification programme and that it was currently in the process of reviewing it. The programme includes calculations of the environmental conditions inside the containment during a severe accident and a plan on how the safety features

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	used for severe accident management will be adapted for the environmental conditions. The representatives of the licensee explained that the specifications for the component requirement include the environmental conditions and the qualification tests are performed after selecting the components. If no suitable component is found on the market, additional justifications will be provided. PRT saw one example of a valve specification with temperature values. During the site visit there was no time for the PRT to check some additional examples related to severe accident management instrumentation which might be operated in conditions of harsh temperatures, pressure or radiation inside the containment. **Assessment:** Since environmental qualification is a crucial part of ensuring that the designed safety features and additional design measures will be able to perform their intended function under severe accident conditions, NDK will thoroughly review the programme and the proposed approaches. **PRT conclusion:** **This issue can be considered as closed**.
T3-4	**Independence of DiD levels** **Discussions:** Defence in Depth is a foundational principle in the design of modern nuclear power plants and as such it is subjected to in-depth regulatory review and is also a topic for the stress test review. During the review process the PRT identified various scenarios in which the design seems to contradict the DiD by relying on the same systems for the same function at different DiD levels, hence breaching the requirement for independence (e.g. primary pressure reduction, which is ensured by the same SSC set at every DiD level above 3a). The licensee has given NDK a document describing the DiD concept. During the mission, it was explained to the PRT that in the case of the Akkuyu design, Level 4 of the DiD covers both DEC-A and DEC-B conditions (it includes BDBA with or without core melt). In the licensee’s opinion this concept is in compliance with IAEA framework (e.g. approach 2 in IAEA SSG-88). **Assessment:** During the review of the NR, as well as in the discussions with NDK and the experts of the licensee, the members of the PRT concluded that the separation of strategies and practices for different DiD levels is not ensured for all scenarios, and in fact several safety systems are expected to perform the same function at various DiD levels. Possible modifications should be investigated to ensure the correct implementation of the Defence in Depth principle (Ref T2-5).

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	PRT conclusion: **This issue can be considered as closed**.
T3-5	**Plans and timeline for establishing the Severe Accident Management Programme and off-site support in case of emergency** (See T3-1 regarding the discussions on EOPs, BDBAMGs and SAMGs during the plant visit and their conclusion.) **Discussions:** During the stress test process there was a specific discussion on multi-unit accidents and alternative power sources used for severe accident management. Electricity supply options, including different sets of diesel generators, are described in more detail under Topic 2. However, the design of the facility is based on the assumption that external help will reach the site within 72 hours which, while in line with the international practice, requires a robust justification in the form of detailed plans and agreements on: • What types of equipment may need to be delivered to the site? • Who is responsible for delivering this equipment, and by what means? • Is this equipment actually present or available in the region, either from governmental organisations or at industrial facilities? • Is there agreement between these parties and the government or the licensee to provide this equipment in case of an emergency? **Assessment:** Since the time limit of 72 hours can be considered critical for the facility (e.g. this is the depletion time of the 3rd stage hydroaccumulators), these issues must be resolved in advance, and generic statements about external support may not be considered sufficient. **Recommendation:** The assumed external support/intervention after 72 hours should be robustly justified (analysed, implemented and trained) both by the licensee and the emergency response organisations in Türkiye.
T3-6	**Safety margins for SAM equipment** **Discussions:** Information on SAM equipment and the PRT raised questions about these values in relation to external hazards (earthquake, tsunami, extreme temperatures, etc.) regarding already progressing accidents as well as maintaining safe shutdown conditions. During the discussions and the site visit the following information was presented for the PRT:

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	• It is important to have the numbers on the safety margins for the SAM equipment, for instance: JMN spray; core catcher systems; annulus ventilation systems; MCR/ECR/PCEP and their supplies: ADGS - including its connection points and the systems and equipment for its fuel replenishment; and PARs and the severe accident monitoring system (hydrogen, radioactivity, temperature and pressure monitoring). • As it was concluded the comprehensive safety measures implemented for severe accident management, encompassing equipment, monitoring systems, and infrastructure, are engineered to withstand seismic impacts of 1.4 SSE. These measures extend to the strategic placement of critical facilities, including entrances to buildings housing severe accident management equipment and the protected emergency command post, at an elevation exceeding the potential tsunami wave level by a significant margin. After an earthquake, the category I equipment retains the ability to perform functions related to ensuring the safety of the NPP. The seismic resistance is confirmed by the supplier of the equipment on the basis of test results. • Connection points for crucial equipment are situated on structures engineered to endure seismic events and exceed tsunami wave levels, ensuring operational continuity during crises. Operational components, such as process equipment and electrical systems, are designed with flexible connections to enhance resilience. • The severe accident management equipment housed within the reactor building, including the sprinkler system, core catcher, and annulus ventilation system, operate autonomously without the need for consumables. This means that they are self-sufficient and do not rely on external resources to perform their crucial functions. The main control room (MCR), emergency control room (ECR), and protected emergency command post are engineered for self-sufficiency without reliance on consumables. • The mobile pumping unit (MPU) is a pivotal component, providing a supply of seawater to cool the essential intermediate circuit during and after an impact at the SSE level. It features a diesel booster pump equipped with a built-in storage tank capable of sustaining operation for 24 hours. Provision is made for diesel replenishment through external mobile tanks or stationary intermediate fuel storage, ensuring sustained functionality during emergencies. PNU 11(21,31,41) PEC10AP003 is designated as belonging to seismic resistance category I per NP-031-01 standards. Its design includes provisions to withstand seismic impacts exceeding those anticipated for the nuclear power plant site, specifically up to 1.4 times the SSE level. • Additionally, the ventilation system of the KLB22 (annulus ventilation, which is required for the fulfillment of the acceptance criteria for DBC3-4 conditions and operated, if available, in DEC-A and DEC-B in line with the ALARA principle) is designed with 2 x 100% redundancy. The SS channels are strategically spaced apart, enhancing system reliability. • All system components are categorised according to seismic resistance standards, with an additional review applied to ensure functionality after an earthquake scenario. The core catcher (CC) system is specifically engineered to withstand seismic forces up to 1.4 SSE. The impact of a tsunami on the CC is excluded by the designer.

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	• The pumping equipment, specifically identified as 11PEC10AP001 and 11PEC10AP002 (12PEC10AP001, 12PEC10AP002), within the 11(12)UQC pumping station, serves as the primary source of the chilled water supply for the BDBA. The impellers of these pumps are strategically submerged below the level of tsunami rollback, ensuring operational integrity even during extreme events. The PEC pump-motor units are designed to withstand seismic effects up to 1.4 SSE. • Additionally, the motors of the PEC pumps are situated at an elevated level relative to the tsunami wave run-up, maintaining operational functionality with a considerable safety margin. This placement, approximately 0.8 meters higher than the tsunami wave run-up level, further mitigates the risk of pump failure. Tsunami wave run-back and run-up do not lead to PEC pumps failure. The assessment of a tsunami impact is considered in the PSAR of the facility. • ADGS tsunami protection is provided by the elevation of the site chosen for the NPP. **Assessment:** As stated in the answers of NDK and explained in the meetings during the mission, the safety margins of the systems, including those used for severe accident management, have been determined and have been approved by NDK. Cooling of the core catcher and depressurisation of the containment, including in the long term, are based on the function of the sprinkler system. Due to the paramount importance of this system for the protection of the containment its power supply should be robustly guaranteed. **Related to Topic 1, finding T1-21.** **PRT conclusion:** **This issue can be considered as closed**.
T3-7	**Annulus ventilation and filtration (KLB22)** **Description:** As provided in the answers to the PRT's questions and the discussion during the visit, the leakage of the primary containment shall not exceed 0.3% of its air volume per day at the maximum design pressure. The KLB22 annulus ventilation system is used to reduce pressure in the annulus between inner and outer containment to prevent releases of radionuclides. As stated, the filtering capacity is 99.99% for radioactive aerosols (0.3µm), 99.9% for molecular iodine and 99% for organic iodine. The activity of radionuclides during accidents and the maximum possible leakage of the primary containment are provided in section 15.7.4.4 'Accidental release limit' of the PSAR. The system consists of two trains (2 x 100%), is activated by the operator or automatically, and can be powered by the diesel generators. Gas-aerosol emissions and dose exposure for personnel due to emissions are described in section 15.7 of the PSAR. Values of public exposure doses due to releases in the

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	initial period of the accident are also presented in PSAR 15.7. As stated, the assessment of radiation doses has shown that the quantitative criteria for the radiological consequences of beyond-design-basis accidents are met. The KLB22 system maintains underpressure only in the annulus and it does not affect the pressure in the primary containment. In the case of a beyond-design basis (BDB) accident due to loss of power supply from all AC sources and/or loss of the ultimate heat sink (blackout), system operation is not required in the first 72 hours. To reliably cut off the medium from common facilities, the hermetic valves 11KLB22AA001 and 12KLB22AA001 are powered from the emergency power supply of the corresponding safety channel of group I (EPSS batteries for 2 hours). Furthermore, no more than 72 hours later, the operation of the system is ensured by connecting power from an alternative diesel generator set (ADGS). **Assessment** During the plant visit it was clarified that KLB22 is used only for maintaining underpressure (subatmospheric pressure) and preventing uncontrolled releases of radioactivity into the atmosphere, including in BDBA. The KLB22 system filters the releases from the primary containment to the annulus. It is not used for depressurisation of the containment during severe accidents. During the visit it was explained that the KLB22 system has not been taken into account in the accident analysis and is not included in the EOPs or SAMGs to control the containment pressure. It was explained that in the PSAR accident analysis, the KLB22 system is used for filtering the DBA accidental releases to fulfil the public dose criteria. It is not needed to fulfil the criteria for the BDBA accidents, but if the system is functional, it should be used to fulfil the ALARA principle. The KLB22 can also be provided with electricity from alternative mobile diesel generators. The PRT has no additional questions or comments on this finding. **PRT conclusion:** **This issue can be considered as closed**.
T3-8	**Habitability of MCR/ECR** **Description** MCR/ECR air conditioning and life support systems have two independent trains (safety class 2 and seismic category 1) that perform the following functions to maintain conditions that ensure the habitability of the personnel. The life support cylinder station has 18 cylinders, each with a capacity of 185 litres, that can support full isolation for five persons for not less than 6 hours. ECR support systems are independent from those of the MCR. Isolation time can be extended by 2 hours by using personal protective equipment.

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	Assessment As provided in the report and answers, the duration of isolation is assessed so that it is ensured that the personnel are able to remain in MCR/ECR long enough to perform the necessary actions for severe accident management and to restore habitability of the control rooms without isolation. No further elaboration was provided in the answers. The issue of MCR/ECR habitability was further discussed during the mission, and additional explanations were provided. On the issue of shift change, as explained, the shift change during a severe accident will be made according to the prevailing conditions. Moreover, the personnel are trained and the conditions are monitored, thus making it possible to assess the situation during the shift change. As explained during the discussions, although the MCR radiological conditions have been assessed by calculating the potential dose rates, such an assessment has not been performed for the ECR. The issue of the habitability of the MCR/ECR and accessibility of control points be further assessed by the licensee and reviewed by NDK, including a radiological assessment, to ensure that access is feasible and that the conditions and duration of residence are sufficient for the personnel to implement the actions required in a severe accident. **Recommendation** It is recommended that the assessment of the habitability of the ECR and accessibility of control points is performed and submitted to NDK for regulatory review, including a radiological assessment.
T3-9	**ADGS reliable operation** **Description:** The ADGS is used as a single power source in severe accident management that provides power to all safety features. There is one ADGS per unit. It is a mobile containerised diesel generator that can be moved by using an individual vehicle. It is located near the EPSS diesel in a seismic category 1 building. In normal operation, the ADGS is in standby mode (the storage period). The ADGS can be used to supply power after the period of 72 hours, during which time passive means (SG PHRS, HA) are used to cool the reactor. **Assessment:** The adequacy and reliability of the ADGS (power, supplies, protection from severe internal and external hazards and severely disrupted conditions on-site) and a reliable strategy for connecting the various consumers should be robustly verified, including in the long term. If such verification is not achievable then backing-up these systems should be considered. This issue is further discussed under Topic 2 (T2-2, T2-4) and the related conclusions are provided there. **PRT conclusion:** **This issue can be considered as closed**.

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T3-10	**Hydrogen burning** **Description** The containment hydrogen monitoring and emergency removal system (JMU-JMT) consists of passive catalytic hydrogen recombiners located where hydrogen accumulation is possible, to prevent hydrogen detonation. The system also includes hydrogen monitoring instrumentation. The detonation of hydrogen is prevented in the case of a situation of beyond design basis and severe accidents. Deflagration of hydrogen is enabled provided that the localising safety systems perform the functions created by the design of the NPP design. The PRT was concerned that deflagration of hydrogen may threaten the integrity and functionality of systems and instrumentation used in severe accident management. Moreover, the operation of sprays may de-inert the containment atmosphere, thus increasing the possibility of hydrogen deflagration/detonation. **Assessment** As explained during the exchange of information, sprays are initiated when the volume concentration of hydrogen in the containment is less than 2%, which is specified as the percentage that ensures that hydrogen will not detonate., The impact on the hydrogen conditions of spraying and the subsequent condensation of the steam was discussed during the meeting. As stated, conditions where the hydrogen concentration exceeds the hydrogen detonation value (2%) are not expected. This is because the greatest amount of hydrogen will be produced during the in-vessel phase, before the sprays begin to operate. This gives enough time for the hydrogen recombiners to reduce the hydrogen in the containment atmosphere. Thus, while the sprays are functioning the hydrogen concentration is significantly reduced, below the detonation limit. Even if this is not the case and the hydrogen concentration reaches the detonation level, it was explained that the spray system stops functioning until the concentration of hydrogen drops below this level. This is provided in specific guidance in the SAMGs. Deflagration of hydrogen is possible during severe accident conditions. As explained during the discussions, the qualifications of the systems and the features of the containment, including monitoring instrumentation (e.g. hydrogen or radiation monitoring), ensure that they can withstand such conditions, due to the range of the conditions for which are designed and because, if these qualications are exceeded, this will be only for a very short time, which will not jeopardise their function. The relevant analysis is included in another report, which has been submitted to and will be reviewed by NDK. **PRT conclusion:** **This issue can be considered as closed**.
T3-11	**Substantiation of core catcher** **Description**

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	The core catcher is a central feature of the management of the ex-vessel phase in a severe accident in the Akkuyu NPP. The function of the core catcher is to provide retention, subcriticality and long-term cooling of the core melt. The core catcher prevents direct interaction of the core melt with the concrete and basemat melt-through, and stabilises the core melt, providing also for its long-term cooling. The Initiation of cooling of the core catcher is based on passive valves. As mentioned, if there is no water or an insufficient volume of water in the shaft, the staff have 2 hours to supply water to the reactor shaft. Cooling of the core catcher is addressed in T3-14, as part of strategy of heat removal of the containment in severe accidents. The PRT recognised that the core catcher performs a critical function during a severe accident. Its operation and effectiveness relies on a series of complex physical and chemical interactions and mechanisms, which should be clearly substantiated. During the visit, it was stated that a dedicated line exists as an alternative means to provide water, from where the fire brigade can supply water to the core catcher if the other cooling capabilities are lost. However, the fire brigade has not received the appropriate training for this operation. **Assessment** The function of the core catcher has been substantiated computationally. It has also been tested experimentally by the manufacturer, JSC Tyazhmash, including the issue of initiation of the cooling by the passive valves. A number of reports were mentioned where the substantiation is provided, for example the 10JKM-MDB0002 report ‘Core catcher. Explanatory note. Computational and analytical justification’. As stated in the information exchange, the amount of sacrificial material and the concentration of Gd_2O_3 ensure the subcriticality of the core melt in the core catcher. During the visit, the PRT was shown the flowchart of the new connection and noticed that there is only one isolation valve outside the containment in the event of a pipebreak. The PRT asked about plans to install a second check valve inside the containment but there are no such plans at present. The PRT was told that the fire brigade will be trained for this operation and that the next revision of the SAMGs will include this measure.

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	In order to ensure that the fire brigade can reliably carry out the operations for supplying water from the alternative connection during severe accidents, the firefighters must be informed and trained. **Recommendation** The licensee should develop procedures and provide training to the on-site fire brigade and carry out the appropriate joint drills to make sure that the firefighters are well informed and trained as regards their roles during severe accidents.
T3-12	**Practical elimination** **Description** The issue of practical elimination was not addressed in the National Report. In the information exchanged and during the visits, it was explained that the overall concept of practical elimination is described in a dedicated document, ’T3-12, Practical exclusion”. The list of the event sequences that must be practically excluded is provided. The design provisions that are intended to cope with such scenarios are also described in the report. According to the PSA Level 2 results, the probability of a large release is lower than 1E-7 per year for internal IEs and internal hazards. However, external hazards and, in particular seismic events, play a significant role with a probability of occurring that exceeds the value of 1E-7 per year. According to the answers provided, the licensee states that the seismic contribution to overall PSA results is hard to determine, and that, due to significant uncertainties, it is difficult to obtain a cumulative probability lower than 1E-7/year. Thus, the licensee applied to NDK to revise the large release criterion to 1E-6 per year. NDK is still assessing this issue. The latest PSA study was expected to be submitted to NDK in September 2024. **Assessment** The list of scenarios included in the report that must be practically eliminated aligns with the scenarios provided in the IAEA standard and covers all operational states, including scenarios with containment bypass or open containment. As acknowledged in the report, practical elimination is based primarily on demonstrating that for scenarios that may lead to large radioactive releases a very low probability is calculated in the PSA Level 2. Target values for this probability are specified in the report both for the probability of individual scenarios and for the cumulative probability. Different acceptance criteria are specified for different types of accident. For example, the approach differentiates between sequences of accidents that can lead directly to large releases (e.g. ejection from the reactor vessel under high pressure) or other sequences where additional conditions (failures) are necessary for a large release to occur. In general, the overall

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	approach needs to be more thoroughly explained. For example, the general criteria specified in the document to define 'extreme unlikeliness' for the different scenarios and types of scenarios need greater elaboration. It is also not clear whether external hazards, including seismic ones, are covered by these criteria. The general approach and criteria, the scenarios leading to an early or large release, the design and operational provisions for their mitigation, as well as the need to implement additional reasonable measures to improve safety must be reviewed and evaluated by NDK, along with the PSA Level 2 study, prior to the operational licensing of the reactor. This evaluation should take into account IAEA standards and WENRA requirements (WENRA Report: Practical Elimination Applied to new NPP Designs – Key elements and expectations, 2019). **Recommendation** NDK should thoroughly review the general approach applied for the practical elimination of scenarios resulting in unacceptable consequences.
T3-13	**Containment bypass** **Description:** Scenarios with the containment bypass going through the SG (or other ways) must be identified and carefully analysed, and the practical elimination or the proper management of these scenarios must be demonstrated by the applicant and assessed by NDK. Before the NDK visit in Ankara, the PRT asked whether it is possible for radioactive aerosols or gases to be released through the paths that circulate atmospheric air to the PHRS heat exchangers (or other ways), and if so, how the isolation of these paths is ensured in the event of a severe accident. During the initial visit to Ankara, further discussions and the site visit, the PRT was informed by NDK that radioactive aerosols or gases cannot be released into the atmosphere through the PHRS heat exchangers because: - the PHRS circulation circuit is connected only to the second circuit, so for radioactive aerosols and gases to be released into the atmosphere, it is necessary to have a combination of a leak in the primary circuit and a leak in the PHRS heat exchanger. This combination significantly reduces the probability of radioactive release from the PHRS heat exchangers; - for the release of radioactive aerosols and gases into the atmosphere, a combination of a leak in the primary circuit and a leak in the PHRS heat exchanger is required; - a leak detection system for the PHRS heat exchanger is included in the design. Also, in each circuit of the PHRS, two serial isolation valves have been installed in both the steam and condensate lines of the PHRS. When a leak is detected, the radioactive release into the atmosphere can be stopped by isolating the damaged steam generator through feed water and steam with protection and blocking signals by those valves. More general explanations on containment bypass were given during the discussions:

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	If a bypass occurs, for example, if there is activity in the KAA industrial circuit, routine actions take place to localise leaks and avoid their consequences. Various possible scenarios with the containment bypass have been taken into account in the design. To eliminate in practice the bypass of the containment through the SG, there is an automatic algorithm to control an accidental leak from the first circuit to the second one by localising the emergency SG, as well as by triggering active and passive cooling systems that reserve each other (SG ECS and PHRS), preventing an increase in pressure in the SG to the setpoints of the SG safety valves. Fittings have been installed on the steam discharge lines from the SG through steam discharge devices (BRU-A), which close when the BRU-A fails to close and releases emissions into the atmosphere. The automatic algorithm is described in Section 12.1.6 of Chapter 12 of the PSAR and in Section 15.4 of Chapter 15 of the PSAR. To exclude bypass of the containment through pipelines and ducts of ventilation systems passing through the boundary of the localisation zone, at least two localization valves are installed on them, which automatically close at increase in pressure in the containment. The description of the insulating fittings is given in Section 12.2.2.6 of Chapter 12 PSAR. In order to exclude the bypass of the containment as a result of damage in a severe accident to the overlap of the lower elevation being part of the localization contour, the fuel melt components provide a core catcher that prevents the interaction of the melt with concrete. The description of the JKM molten core capture and cooling system is given in Section 12.2.3/5 of Chapter 12 of the PSAR. **Assessment:** There is a lack of detailed information on containment bypass in the National Report. However, during the discussions the additional information was provided, mostly by NDK, and the PRT was able to understand the issue. The general approach seems to be reasonable and systematic. During the final discussion, during the site visit, the PRT was informed that the issue of containment bypass is covered under the practical elimination, and the relevant document, entitled ‘Practical exclusion’ was shown. The document presented seems to be comprehensive and systematic, however NDK should review the consistency and correctness of the approaches, and the completeness and sufficiency of the information. **PRT conclusion:** This topic is closely related to the **practical elimination** concept which is covered under issue **T3-12**. There was no need for additional discussion on T3-13 and it can be **considered closed**.
T3-14	**Cooling strategies** **Description:** There is no means dedicated to DiD Level 4 for the containment, core catcher and SFP cooling. According to the information provided there is the possibility of conditions arising (the associated scenarios were not identified) in which steam condensation may not be possible (see the

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	note below). These scenarios must be identified, and management strategies have to be ensured. The overall strategy of cooling the containment, including the function of the annulus ventilation system, the core catcher and SFP, seems very challenging and should be verified. *Note: a temperature in the containment equal to 200 C is mentioned, for which the condensation of steam is possible only at very high pressures, well exceeding the design containment pressure.* Additional information on technical means and strategies was provided during the review and the conversations at the Ankara and plant visits. It was stated that according to the information presented in the PSAR for the DiD Level 4, the means of beyond design basis accident management are provided, which ensure pressure reduction in the containment. These comprise: (i) an additional FP cooling system (SFPCS), which performs the function of a BDBA sprinkler system; (ii) an alternative industrial circuit system that removes heat from the SFPCS; (iii) a mobile pump unit supplying cooling sea water to an alternative intermediate circuit; (iv) an alternative air-cooled diesel generator that provides power to the mechanisms of these systems; (v) a sprinkler pump of the first channel, connected to an alternative diesel generator, and topping up the spent fuel pool and the supply of coolant to the reactor through a dike with HE-2 with flow rates that stop boiling in the spent fuel pool and the reactor. At the initial stage of the BDBA with a primary circuit leak, the temperature in the containment rises to 200 °C as a result of the release of a large amount of the primary coolant into the containment area. Subsequently, the temperature in the containment decreases due to steam condensation on the structure of the containment building and a decrease in the coolant outflow from the primary circuit. In the event of a blackout due to the failure to start all DGs and, accordingly, a failure of the active emergency systems that remove heat from the containment, heat is removed from the fuel in the reactor for 72 hours by passive systems (HE-2, HE-3, PHRS) and from the SFP in the spent fuel in a boiling mode, which are the worst conditions for the parameters occurrence in the containment. The operation of these systems also provides conditions under which the pressure in the containment within 72 hours from the moment of the initiating BDBA event does not exceed the calculated value of 0.5 MPa, and the temperature does not exceed 150 °C by the end of 72 hours (considering the above peak of 200 °C at the beginning). After 72 hours, the BDBA control means are activated, which perform heat removal from the containment area and pressure reduction. The licensee informed NDK and the peer review team that the results of the BDBA analysis (with maximum parameters under the containment at a time interval of up to 100 hours) are presented in Subsection 15.6.2.2.2 of Chapter 15 of the PSAR. This section also presents the results

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	of the analysis of parameters when cooling water is supplied after 72 hours by BDBA control means only in SFP and when supplied to FS and to the reactor. However, the team was not able to look at this information in detail. **Assessment:** As clarified during the discussions, cooling of the core catcher and the containment is achieved by the operation of the sprinkler system (see also T3-6). The sprinkler system can be operated by two independent pumps (JMN10 and JMN61), using the water in the sump of the reactor and water from the spent fuel pool. The containment pressure does not exceed the design pressure during all severe accident scenarios calculated and presented in Chapter 15 of the PSAR. The operation of the sprinkler system for depressurisation of the containment is included in the SAMGs. However, both during the review of the NR and the site visits it was concluded by the PRT that in the long term the use of the sprinkler system is the only available means to ensure long-term heat removal from the cac in the core catcher and the depressurisation of the containment. While according to the design this function may be fulfilled by various sets of pumps dedicated to the different DiD levels, this is all based on the assumption that the piping and sprinkler heads (which have a double redundancy) will be available/undamaged during a accident (e.g. after a major earthquake). To decrease the SA strategy's dependence on this assumption additional strategies could be identified that ensure diverse means of depressurisation and heat removal (e.g. filtered venting strategies, long-term containment cooling solutions, passive containment cooling etc.). In addition, considering the possibilities of double containment and consequential low cooling through the containment walls, the only mean of heat transfer from the containment area is the heat exchanger of the sprinkler system. Despite the fact that in the initial phase of BDBA the condensing of steam at the containment walls and equipment would be possible, this process will be stopped after the heating up of these structures above the condensing temperatures. The boiling in the core and SFP will only additionally heat up the containment inventory. So, during the BDBA, without the use of the spray system's heat exchanger, incondensable conditions can occur. As the use of the spray system is not provided in the BDBA (before the core melt), the temperature and pressure will only increase. During the core melt, the large amount of thermal energy will be released to the containment due to the steam-zirconium reaction ($Zr + 2H_2O = ZrO_2 + 2H_2 + 6530$ kJ/kg) and the action of the recombiners. It is necessary to mention that during this phase the use of the spray system will be prohibited due to the hydrogen safety conditions. So, the containment environment will be overheated even more, and the incondensable conditions margin will increase.

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	In these conditions, the action of the sprinkler system, at the initial phase, will not be in a closed loop. Due to the overheated medium (especially at the upper part of containment) sprayed water will be heated into steam before it will reach the bottom of the containment. In the discussions during the plant visit, the team was informed that there is a dedicated feedwater line to the core catcher, providing the possibility to connect a fire engine or other mobile means to supply water to the core catcher. Despite this positive development, it does not change the general issue that, due to the fact that supplied water will evaporate, the heat will just transfer from the core to the overheated containment environment. In the absence of any other technical means for cooling the containment, this issue can be considered as a cliff edge situation. So, the issue should be carefully analysed, and, if appropriate, appropriate strategies should be developed and technical measures implemented. **Recommendation:** The licensee should investigate the possibility of alternative long-term cooling/depressurisation strategies and means for the containment under severe accident conditions
T3-15	**Containment conditions** **Description:** Before the visit to NDK in Ankara, the PRT requested a list of conditions and severe accident sequences, subjected to detailed deterministic assessment, that may threaten the confinement function (both early and late) of the containment and the time frame before this happens. During the review the PRT was informed that an analysis of the following severe accidents, which may threaten the integrity of the containment, is presented in the PSAR: 1) Minor leakage of coolant from cold leg; 2) Break of RCPL at the reactor outlet; 3) Cold leg rupture in the lower part of loop seal 1; 4) Break of RCPL at the reactor inlet.

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	In addition, information was requested from the licensee regarding the specific conditions that might cause an early or large release, and an explanation of why, among all these conditions in (A), only those presented in the relevant section of the PSAR are subject to deterministic assessment and the others are not. After the visit to NDK in Ankara, the PRT asked for the list of scenarios in line with the results of Level 1 and Level 2 PSA; which scenarios can lead to overpressurisation of the containment in core melt accidents (DEC 2) in the in-vessel and ex-vessel phase; what measures there are to protect the containment in such scenarios or conditions; and for the additional scenarios calculated within the framework of the PSA (compared to those given in Chapter 15). The team was informed that the in-vessel design solutions practically exclude scenarios that could lead to exceeding the design pressure in the containment such as the destruction of the reactor vessel at high pressure in it, steam explosions, etc. This is due to the availability of the PHRS and HAs at in-vessel stage. At the ex-vessel stage, the sources of pressure increase are steam generated during cooling of the CC and steam from the SFP during its boiling. Other causes of pressure increase are either absent or excluded. An increase in pressure above the design level can occur only if special hardware fails (active systems + either PHRS or HAs). Such scenarios have an extremely low probability and are not considered in the project. The failure of special hardware means the transition to Level 5 DiD and plans for the protection of the population should be activated. **Assessment:** The containment cooling strategy was extensively discussed during the plant visit. More details can be found under **T3-14**. A detailed description of the practical elimination (exclusion) issue can be found under **T3-13.** There is no need to discuss this issue separately. **PRT conclusion:** **This issue can be considered as closed**.
T3-16	**PHRS (Related to Topic 2, finding T2-7.)** **Description:** The team requested the information regarding redundancy provided for the systems, for instance the valves, used for initiation and operation of the PHRS system and whether actions by the personnel were needed. During the visits and the discussions, a great deal of information was provided to the PRT by NDK and the licensee.

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	The PHRS provides redundancy of system components, a 33% reserve is provided for the heat removal capacity (4 loops, 33% each). According to the accepted design requirements, three operable PHRS circuits are enough to perform the system function in any mode that requires it to function. The opening of air locks is automatic and does not require any action on the part of the operator. Air locks are not redundant. At the inlet to the heat exchanger PHRS there are two air locks, at the outlet of the heat exchanger there is one, but structurally consisting of two independent gates. Incomplete opening of the gates or failure to open one of the gate valves or gates has little effect on the power output of the heat exchanger as a whole. So, for example, when opening one gate of the upper gate by 15 degrees, and the second by 75 degrees, the output power is approximately 84% of the nominal. **Assessment:** As clarified in the discussion, the PHRS system belongs to Level 4 of the DiD concept of the Akkuyu nuclear plant. This is a passive means used when other systems are lost. The operation of the system has been tested and verified both computationally and experimentally, while it has adequate redundancy. **PRT conclusion:** **This issue can be considered as closed**.
T3-17	**Dose criteria issues** **Description:** As provided in the answers of NDK before the Ankara visit, the dose criteria accepted as the maximum emergency release during PSA-2 development are 50 mGy and 500 mGy for the effective dose and 500 mGy to the thyroid, respectively, for the first days. The PRT presented the following additional questions: a. Please specify the distance in km where these criteria apply according to the license and safety analyses of the plant. Which are the definition and range of the emergency zones around the plant? b. While the effective dose complies with the IAEA GSR Part 7 generic criteria for stochastic effects, the criterion for the thyroid dose seems to be much higher than the respective IAEA criterion of 50 mSv in 7 days. Can you please explain the basis for specifying the thyroid dose criterion? c. Are there accidents/scenarios approaching or exceeding the above criteria? Please also specify the distance of exceedance. The PRT was informed that the precautionary Action Zone (PAZ) is 5 km from the NPP's limits. (Currently, the internal boundary of the PAZ is the site limits, the external boundary is a ring of 5 km radius from the vertical axis of the reactor of Power Unit 1). The urgent protective action planning zone (UPZ) is 20 km around the NPP limits. (Currently, the internal boundary of the UPZ is the PAZ external boundary, and the external boundary of the UPZ is a ring of 20 km radius from the vertical axis of the reactor of Power Unit 1).

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	Russian criteria for evacuation do not consider the thyroid for evacuation. In compliance with the requirements of the International Atomic Energy Agency, PSA-2 is being revised; however, this report has not yet been submitted to the Nuclear Regulatory Authority (NDK). According to the PSA results, there are some scenarios that exceed the prescribed dose limits beyond the action zones according to Russian dose limits. The licensee has been asked for these scenarios and a report has been presented. According to this report, the frequency of these scenarios is very low (between 1E-9 and 1E-17). The PSA Level 2 study will be revised in accordance with Turkish regulations and was due to be submitted in September 2024. Because of that, the review and assessment of this issue continues. **Assessment:** During the plant visit NDK explained that the Turkish national requirements for protective action limits and public doses in accident conditions are in line with the IAEA GSR Part 7. The details of the PAZ and UPZ and the related criteria are given above. PSA Level 2 analysis are due to be revised in 2024 in accordance with Turkish regulations, and NDK will follow-up on this topic. **PRT conclusion:** **This issue can be considered as closed**.
T3-18	**Functioning of PRZ PORV together with KTP during overpressurized scenarios** **Description:** According to the stress test specification, quick depressurisation to 1 MPa must be ensured. In addition, on answers 22, 60 and 66 NDK commented that it is waiting for a detailed analysis for further review of the functioning of PRZ PORV, together with KTP during overpressurised scenarios in the primary circuit, and that the applicant should justify the quick depressurisation to 1 MPa to avoid failure of the core catcher. According to the safety analysis report, ‘in the context of design-extending accidents involving KTP (EGRS) core meltdown, this system is used to reduce the primary pressure by up to 1 MPa by venting the medium from the PRZ to the containment, according to the accident management instructions.’ PSA calculations are based on the assumption that PRZ PORV and EGRS are opened together by the operating personnel to reduce the pressure below 1 MPa. This information is provided in the reply to the additional information requested from the Akkuyu NPP and in the safety analysis report. The situation of important events related to severe accidents was calculated according to time and the relevant results were presented in tables and graphs by the Akkuyu NPP. According to the calculations made, PRZ PORV and EGRS are turned on at 13.7 hours after an initiating event occurs, and it is indicated in the relevant graph that the primary circuit pressure falls below 1 MPa.

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	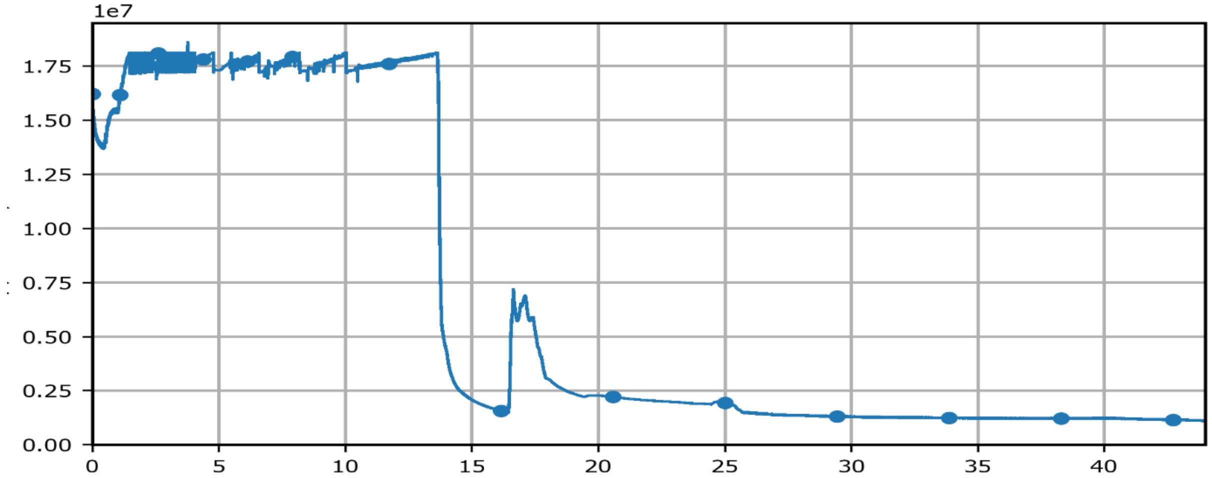 **Assessment:** In the event that cooling of the reactor is lost, depressurisation is performed by the joint operation of PORV and the emergency gas removal system. As both systems are needed, no diversity is ensured for the depressurisation of the reactor in the event of loss of cooling capability. In addition, these systems are not dedicated for this function (and this DiD level). This lack of diversity seems to weaken the practical elimination of a high-pressure ejection of the core. **Recommendation** The issue of the primary circuit depressurisation by the means dedicated to Level 4 DiD must be carefully analysed and, if relevant, the appropriate measures must be developed and implemented.
T3-19	**ADGS restricted set of mechanisms** **Description:** During the desktop review the PRT received the information that an 0.4 kV air-cooled standby diesel generator supplies power to a restricted set of mechanisms, which are used for the management of beyond design-basis accidents: - an alternate intermediate circuit pump installed in the first train of the essential load intermediate system KAA; - a primary train spray pump; - two emergency boron input pumps of the first train of the JND system;

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	- an additional cooling system pump of the fuel pool (included in the JMN sprinkler system); - primary train motor-operated valves of the JMN, JNA, KAA systems; - a post-accident monitoring system During the further analysis and conversations, it was identified that other equipment (e.g. the annulus ventilation system, MCR/ECR, some pumps) are also powered from these ADGs. **Assessment:** It is crucial to ensure that during the BDBA/SA the power, consumed by the ADGs consumers will not exceed the ADGs capacity. During the plant visit NDK provided the list of ADGs consumers, including all mentioned consumers, batteries and even some Level 3 DiD equipment, which can be used in SAM strategies, plus a capacity margin of 25%. **PRT conclusion:** **This issue can be considered as closed**.
T3-20	**SFP water supply** **Description:** During the desktop review on answers it was stated that during BDBA with primary circuit leaks the JMN system supplies water from the emergency sump to the reactor and simultaneously to the FP at a rate sufficient to remove the heat without boiling. Thus, the steam inlet is terminated in the containment volume and a pressure increase in the containment is prevented. Water supplied to the reactor and FP is returned to the emergency sump, from where it again supplies the reactor and the FP, thus a continuous coolant circulation is created. The JMN60 system can be used for both SFP cooling and MC pressure reduction. The pump is powered by an alternative diesel generator set, which also supplies the 0.4 kV power supply sections of the first or second SS channel, which in turn indicates that if the JMN61 pump is operational, then the JMN11 and JND10,20 pumps are highly likely to be operational to help the JMN61 pump. **Assessment:** During the discussions, the use of the water of the SFP, which is also connected with the overall strategy of cooling the containment, was clarified. This issue, therefore, is related with the use of water to cool the containment and is already addressed in the previous question (T3-14). There is no need for a separate discussion on this issue. **PRT conclusion:** **This issue can be considered as closed**.

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T3-21	**Additional measures to maintain containment integrity** **Description:** According to the national report (Section 6.3.10), the following measures shall be envisaged to enhance the capability to maintain the containment integrity after heavy fuel damage: • development and analytical substantiation of severe accident management operator actions (SAMG) to protect containment integrity; • analysis and measures to be taken in the event of a simultaneous accident at several NPP power units and the impact of several hazards on the entire site, the loss of emergency power supply - complete blackout for more than 72 hours (loss of all AC and DC sources), loss of instrumentation and post-accident monitoring systems, and also additional measures for the use of mobile means to restore emergency power supply and severe accident management; • development of additional equipment for managing severe accidents at power units remotely from the NPP crisis centre (PECP) in the event of loss of control or life support facilities at MCR and ECR. During the discussions the PRT was informed that analytical substantiation of severe accident management operator actions is under development and is a subject for the regulatory review. Additionally, it was stated that identical sets of safety systems and special hardware for controlling beyond design-basis accidents are provided for each power unit. Therefore, each power unit is independent of the other power units to overcome accidents, including in the case of a simultaneous accident at several NPP power units due to a common cause (external influence). Protected emergency management points (PEMP NPP) are designed to be followed up in the implementation of action plans to protect personnel and the public in case of an accident at the NPP. PEMP NPP is the location of the crisis centre, the NPP Emergency Situations Commission, the Head of Emergency Operations (HEO), and the NPP emergency rescue unit. PEMP NPP has the autonomy to function for 5 days for the entire staff stationed in it. The PEMP NPP is equipped with: - communication facilities (radio, telephone, satellite communications, television (including industrial); - means of receiving, transmitting, processing and storing information about the state of NPP power units in their operation, in case of an accident (its development and course); - means of receiving, transmitting, processing and storing information about the radiation situation at nuclear power plants and the area of operation of the ASKO. Additional equipment for the remote control of severe accidents from the PEMP NPP crisis centre is not provided. **Assessment:**

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	The issue of additional equipment for the remote management of an SA from the crisis centre (PECP) belongs to the SA strategies, and was discussed with NDK during the discussions on SAMGs, and there is no need to discuss it separately (T3-1 and T3-5). The other issues, mentioned above, were already discussed under issues T3-14 and T3-20. **PRT conclusion:** **This issue can be considered as closed**.
T3-22	**Additional measures to maintain containment integrity** **Description:** During the discussions with NDK and the licensee it was stated that the PHRS system, in combination with the second stage hydroaccumulators, is expected to operate during BDBA primary circuit leak scenarios (LOCA) as a mean to cool/condensate the steam accumulating in the primary circuit, as well as to remove heat from the system through heat exchange between the non-condensable gases in the primary side and the steam/water mixture in the secondary side while removing it to the alternative heat sink (external air). As described by the licensee it is considered a mode not inherent to the SG design basis, namely a mode of condensation of the steam generated in the core and transferred to the inner space of the SG tube bundle. According to the licensee there is research available on this phenomenon and the functionality of the PHRS under such conditions which is validated via experiments. The research and experiments were performed at the SG prototype in the steam-gas mixture condensation mode to define the steam generator condensation power with three stages of hydrogen and nitrogen generation in the core, in order to: • simulate the mixture outflow to HA-2; • determine the minimum amount of steam-gas mixture outflow to HA-2 required for maintaining the SG condensation power; • determine the impact of gas composition (nitrogen, helium, helium-nitrogen mixture) on the steam generator condensation power. The sufficiency of SG condensation capacity, allowing for the abovementioned peculiarities of the process for the PHRS efficient operation, has been proved by IPPE (A.I. Leipunsky Institute for Physics and Power Engineering) using a full-scale GE2M-PG test bench, which has a RP V392M steam generator prototype (scale 1:48). The SG prototype has full-scale tube length and tube bundle height. In the course of experiments, as a result of condensation-induced natural convection, the steam-gas mixture from the mixture preparation system, containing steam, nitrogen and helium (it substitutes hydrogen), was supplied to the SG prototype. The proportions of gases in the mixture were specified based on the results of analysis of gas generation in the reactor. These calculations were carried out via the certified MORAVA-N2 software. Similar to the

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	full-scale plant, the steam-gas mixture was withdrawn from the SG prototype cold headers with a volumetric flow rate scale corresponding to the mix outflow from SG to HA-2 of the full-scale plant. The experiments were used to verify the correctness of the SG condensation capacity analysis using the TETCH/ANGAR; SOKRAT + ANGAR software packages and RELAP5/MOD3 software code. The mentioned software packages were used for comparative calculations to justify NPP safety in the event of LOCA scenarios with a concurrent failure of ECCS active parts. According to the verification results, these software packages, in terms of quality and quantity, provide a correct description of SG model condensation capacity behaviour. Moreover, they give even under-valued/conservative results according to the licensee. Based on this information it was concluded by the licensee that the PHRS is capable of heat removal in the amount specified in the design under the conditions of a loss-of-coolant accident and a failure of the ECCS active parts, even while taking into consideration the non-condensing gas supply to the SG heat transfer area. **Assessment:** Due to time limits on the review process the PRT did not review the studies mentioned by the licensee, but the PRT recognised the efforts of the designer to assess and validate the adequacy of the PHRS design. During the discussions on different topics, however, the PHRS system came up with various non-trivial roles (e.g. as a mean to remove heat from the containment after RPV failure in the late stages of a severe accident), which the PRT considers should be subjected to in-depth regulatory review, since these modes seems to be widely out of scope of the original design purposes. **PRT conclusion:** **This issue can be considered as closed**.
T3-23	**Accident management and restoring containment leak tightness in shutdown conditions** **Description:** The National Report (Revision 2) did not include information on the consideration of operational shutdown states in the severe accident management strategy (DEC with core melt). Measures for ensuring containment isolation in shutdown conditions were not explained in the NR. **Assessment:** NDK has clarified that, for severe accidents initiating at shutdown states, the approach is to practically eliminate the possibility of fuel damage by ensuring core cooling so that no damage to the core is possible within at least 72 hours. After 72 hours, the provision for continuing core cooling are the same as those for power operation.

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	Concerning the containment isolation in shutdown conditions, NDK has clarified that the leak tightness of the containment is maintained. If, for technical reasons, both doors of the airlock were open, then there is an option to quickly close them using an electric or manual drive. **PRT conclusion:** This issue should be covered in the topic of practical elimination (T3-12) and this finding can be considered closed.
T3-24	**Effectiveness of hydrogen mixing in the containment** **Description:** Section 6.3.2 of the National Report (Revision 2) states that '*Passive catalytic hydrogen recombiners are located in places (rooms) of possible hydrogen accumulation with the required capacity, which allow performing a given function at any state of the gas-vapor mixture in such a way that it does not require mixing the medium in the containment to create a homogeneous atmosphere*.' However, in a typical VVER geometry, the local hydrogen and steam concentrations in the steam generator compartment may be high in the early stages of LOCA (not homogenous). Release of gases from the SG compartment to the containment dome and the following mixing of the containment atmosphere is necessary to level out the H2 concentration, even when PARs are available. **Assessment:** NDK has clarified that the highest hydrogen concentration is formed in the steam generator room at the location of the postulated leak of the main coolant line. A report with justification of the hydrogen explosion safety has been reviewed and approved by NDK. Hydrogen combustion risks have been considered in operating the containment spray (sprinkler) system, so that hydrogen concentrations are checked before spraying the containment atmosphere. In general, the hydrogen management system based on PARs meets the international requirements on managing explosion risks in severe accident conditions. **PRT conclusion:** The issue can be considered as closed.
T3-25 **PRT conclusion**	**Description:** During the visit site visit and the short visit to the radiation centre it was observed that the area to the north of the facility seems to have no gamma detectors at the moment, and it remained unclear whether it is planned to deploy additional detectors in the near future, or that the current state of the system can be considered the final version. It was also briefly mentioned that the high number of detectors in the eastern sector is due to the typical wind direction in the area. **Assessment:**

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Finding nr	**Peer review team finding**
	The PRT considered that the high number of detectors in the eastern sector is a fair point but not a reason not to have detectors in the northern sectors. In order to ensure that the organisations responsible for emergency response have adequate and comprehensive information about the radiation levels around the site, the possibility of installing further radiation detectors should be investigated. **Recommendation** The licensee should investigate the need to install addtional radiation detectors in the vicinity of the facility to ensure comprehensive coverage of the area.