



Multinational Design Evaluation Programme

Technical Report

TR-VVERWG-08

Lessons Learnt from Review of Preliminary Safety Analysis Report

Regulators involved in the VVER working group discussions:	HAEA (Hungary), NNSA (China) and NSC as its TSO, SEC NRS (Russia) as TSO of Rostekhnadzor, NDK (Türkiye), STUK (Finland)
Regulators that support the present technical report:	HAEA (Hungary), NNSA (China) and NSC as its TSO, SEC NRS (Russia) as TSO of Rostekhnadzor, NDK (Türkiye)
Comments:	The development of this report began in 2019. Although STUK withdrew from MDEP in 2022, it has raised no objections to its publication.

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List of abbreviations and acronyms

CR	Control Rod
DiD	Defence in Depth
HAEA	Hungarian Atomic Energy Agency
MDEP	Multinational Design Evaluation Programme
NDK	Nuclear Regulatory Authority of Türkiye
NNSA	National Nuclear Safety Authority of China
NPP	Nuclear Power Plant
PSAR	Preliminary Safety Analysis Report
SAR	Safety Assessment Report
STUK	Radiation and Nuclear Safety Authority of Finland
TSO	Technical Support Organisation
VVER WG	VVER Working Group

1. Introduction

VVER Working Group (VVER WG) was established to foster communication among the members on regulatory approaches for safety review for VVER-1200 within the frame of the national licensing procedure and to promote understanding of each country's regulatory decisions and the basis for the decisions.

The preparation of the report on the safety review experience began in 2020 as part of the activities of the subgroup on the analysis of transients and accidents and was completed after the merger of this subgroup with the subgroup on severe accidents. At the time the report preparation began, all countries participating in the subgroup, except for Russia, were conducting safety assessments at various stages of the licensing of VVER-1200 nuclear power plants. In the Russian Federation, the safety assessment for VVER-1200 nuclear power plants had already been completed, and operating licences had been issued.

The basis for the licensing is the Preliminary Safety Analysis Report (PSAR), which presents information on accident and transient analysis among others. This information may vary from country to country based on the national requirements, but it is also a good basis to exchange approaches and information on accidents and transients and to foster a common understanding of safety related problems and their possible solutions.

2. Objective

This technical report discusses lessons learnt from the limited number of questions originated from PSAR review during the safety assessment of VVER nuclear power plants by Hungarian Atomic Energy Agency (HAEA), Radiation and Nuclear Safety Authority of Finland (STUK), Nuclear Regulatory Authority of Türkiye (NDK), and National Nuclear Safety Authority of China (NNSA). For each question, the opinions of the subgroup participants on their relevance are given, taking into account national safety regulation features and national regulatory requirements. The questions originated from the area of accidents and transients and mainly from the review of chapter 15 from PSAR (Transient and Accident Analysis). The objective is to discuss and compare approaches, views and experiences of national regulators related to accidents and transients related topics.

In 2022, STUK withdrew from Multinational Design Evaluation Programme (MDEP), and therefore not all questions are accompanied by comments from STUK. Nonetheless, the report includes STUK experts' questions on the safety justification of the Hanhikivi nuclear power plant, along with comments on these questions from other subgroup participants who found this information valuable for the exchange of experience in conducting safety assessments.

3. Background

HAEA, STUK, NDK, and NNSA reviewed and assessed information submitted by the vendor to determine whether VVER-1200 design complies with regulatory requirements. During the review a limited number of questions raised by regulatory experts were distributed among the VVER WG members for the discussions.

HAEA’s 6 questions (see Appendix A) related to:

- analysis for the unintentional movement of the control rod assembly with a subcritical reactor,
- justification for the fulfilment of acceptance criteria for leaky fuel rods,
- analysis of unjustified startup of the ECCS,
- the difference in the Hydro Accumulators System pipeline diameter with comparison to the reference unit,
- decay heat removal process by Low Pressure ECCS during maintenance water levels,
- hydro accumulator’s adiabatic gas expansion.

STUK’s 2 questions (see appendix B) related to:

- containment isolation (localization) function,
- defence in depth (DiD) for JNG-1 Heat exchangers.

NNSA question (see appendix C) related to Fuel cask drop accident.

NDK’s 2 (see appendix D) questions related to:

- demonstration of subcriticality in reactivity-initiated accidents,
- the difference of the number of pumps in steam generator Emergency Cooldown System (JNB10-40) in comparison to the reference unit.

The questions are presented in tables in Appendixes A – D with the following template:

Question	Short description of question which should help the quick understanding of the question.
Description	Detailed description of the question which contains important information to understand the question.
Requirement	Related regulatory requirements in the member country.
Comments	Comments by other members containing national approaches, views and experiences related to the question.

The short description of VVER-1200 system identifiers is presented in appendix E.

4. Lessons Learnt

The communication between regulators during safety review ensured a common understanding of safety issues. For each question presented in the report national regulators requested additional clarification from the vendor and licensee and questions are being resolved in relation to each issue question in the framework of the national licensing procedure.

The discussion of questions made it possible to identify differences in national regulatory requirements. The questions and the comments of the report may be also useful for design vendors to gain deeper understanding of national regulatory basis as well as expectations from the regulatory body experts on safety justification of VVER-1200 design. The results can also be used by the regulatory bodies of other countries who are not the member of VVER WG but implementing the VVER-1200 technologies.

Such regulatory cooperation and review of practices and results exchange are mutually beneficial for training and knowledge transfer among the experts of regulatory bodies and their Technical Support Organisations (TSOs). The results of discussions have been used to facilitate the regulatory reviews and assessment work of various VVER designs and to enhance the scope, level of detail, and quality of VVER-1200 transient and accident analysis and safety justification documentation.

Appendix A – HAEA questions

HAEA Question 1	Missing analysis of the unintentional movement of the control rod assembly with a subcritical reactor
Description	<p>The chapter 3.1.4 of PSAR contains the list of DBC2. For the fulfilment of the requirement for analysing unintentional movement of the control rod assembly with subcritical reactor and with operation on full power the list shows the following event category:</p> <p>“Unintentional movement of one or more control rod assemblies at minimum controllable power and full power states.”</p> <p>It is not explained that this analysis covers the case of a subcritical reactor.</p>
Requirement	<p>3a.2.2.4500. Among the various DBC1-4, at least the following internal events shall be taken into account in the design of the nuclear power plant:</p> <p>[...]</p> <p>c) DBC2 operating conditions:</p> <p>ca) unintentional movement of the control rod assembly with a subcritical reactor,</p> <p>cb) unintentional movement of the control rod assembly with operation at full power,</p> <p>[...]</p>
Comments	<p>STUK:</p> <p>STUK has not received updated PSAR analysis for construction license. Thus, the scope of the analysis is still a bit unclear. As YVL guides do not give a detailed list of analysis to be provided a direct requirement for this cannot be given. All plant initial states have to be covered with specific analysis or other justification. There are expectations and requirements for shutdown margin calculations to be done to subcritical core.</p> <p>SEC NRS:</p> <p>According to the NP-001-15, p. 1.2.15, the preliminary list of initiating events is set in the requirements specific for each reactor type. For VVER-1200 such list is set in Appendix 7 to NP-006-16. This list includes unintentional movement of the control rod assembly but does not specify criticality/subcriticality or the power level at which the initiating event occurs.</p>

According to NP-001-15, p. 1.2.9 conservative approach shall be applied to the analyses of the DBA. This means that the worst-case scenario shall be analysed, including selection of worst combination of the plant parameters (including, of course, initial power level). So, according to Russian requirements, it is not required to provide both analyses (at critical and subcritical state) as long as it is justified in Safety Assessment Report (SAR) that the worst-case scenario was selected for the analysis. But if it is not possible to justify the selection of the worst-case scenario, both (or even more if needed) scenarios (unintentional movement of the control rod assembly with a subcritical reactor and at full power) shall be analysed and their result presented in SAR.

NDK:

The expression "minimum controllable power state" that the reactor is kept at a certain power level and that this condition will be met with criticality since there is power generation, that is, the reactor is critical. (This should be seen from the initial value given for power in SAR. If such information is not available, this information must be requested.)

So, the scope of the analysis does not cover subcritical state.

The analysis submitted for "Uncontrolled Withdrawal of CPS Rod Group" does not cover subcritical state in Akkuyu Nuclear Power Plant's (NPP) SAR either. It covers calculations at hot zero power-HZP (which is 1% of nominal power and has the same meaning as minimum controlled power level) and at power (which is 104% of nominal power).

To meet this requirement, an analysis is needed to be submitted for the start-up stage i.e. the stage of bringing to power from the hot state (in which reactor is in subcritical state and subcriticality of at least 1% is ensured by boric acid concentration in the primary coolant without consideration of RCCAs submerged in the core) to HZP (in which reactor is in critical state). Since, during this transition state, the subcriticality needs to be ensured by RCCAs.

Otherwise, justification on exclusion of this condition needs to be submitted by the licensee.

NNSA:

In PSAR of Chinese VVER1200 project, unintentional movement of one or more control rod assemblies in subcritical condition is also not analysed. Designer shall supply adequate evidence to prove this scenario is enveloped by other accident scenario.

HAEA Question 2	No justification for the fulfilment of acceptance criteria
Description	<p>In chapter 15.5.3.2 “Analysis of Core Cooling and System Pressure for a Decrease of Reactor Coolant Flow” the fulfilment of the acceptance criteria for leaky fuel rods is not described in the case of the chapter “Instantaneous Seizure or Break of the Shaft in One Reactor Coolant Pump Set”.</p> <p>The acceptance criterion is the following: The number of leaky fuel rods during an accident shall not exceed 10 % of the total number of fuel rods in the core.</p> <p>The table summarizing the acceptance criteria contains this criterion but only refers to chapter 4.2.“Fuel system” as fulfilment, but this chapter only gives a general description which is not specific for this analysis.</p>
Requirement	3a.2.3.0900. For all initiating events included in the design basis or in the extended design basis, the fulfilment of the relevant acceptance criteria shall be demonstrated by deterministic safety analyses
Comments	<p>STUK:</p> <p>STUK would also expect such an analysis to be performed to justify effects of shaft seizure or break. The acceptance criteria depend on the classification of the accident. PSAR chapter 15 analysis will be delivered later.</p> <p>SEC NRS:</p> <p>According to Russian regulations, for DBA it is required that fulfilment of acceptance criteria shall be justified in SAR.</p> <p>Reference:</p> <ul style="list-style-type: none"> • NP-001-15, p. 1.2.1: “The NPP meets the safety requirements if the following conditions are met: the radiation impact of the NPP on the personnel, population and the environment during normal operation and violations of normal operation up to design basis accidents, inclusive, does not lead to an excess of the established radiation doses for personnel and population, standards for emissions and discharges, • nuclear power plant radiation impact on personnel, population and environment is limited in case of beyond design basis accidents, • the probability of accidents at the NPP is limited.”

	<p>NP-082-07, p. 2.1.12: “In Reactor Facility and NPP design, design limits for design basis accidents shall be set to each design basis accident or a group of accidents, which must not be exceeded taking into account the operation of safety systems.”</p> <p>NP-082-07, p .5.1.5: “For each initial event being analysed, the following shall be provided:</p> <p>...</p> <p>Conclusion of fulfilment of design limits and acceptance criteria.</p> <p>...”</p> <p>NDK:</p> <p>Agreed.</p> <p>In PSAR of the Akkuyu NPP, the mentioned criterion is defined with the following footnote:</p> <p><i>“Under the conditions of categories 4 for which by calculation results DNB on the surface of fuel rod claddings is reached, the number of the fuel rods subjected to DNB, shall not exceed 10 % of the total number of fuel rods. In cases when the number of the fuel rods subjected to DNB, more than 10 % of the total number of the fuel rods being in the core, it shall be demonstrated that number of leaky fuel rods does not exceed 10 % of the total number of the fuel rods in the core. In this case loss of integrity of a fuel rod means a direct contact of nuclear fuel with the coolant as a result of a loss of integrity (break) of cladding of fuel rod under action of “stretching” pressure differential onto the cladding and high temperatures.”</i></p> <p>From this paragraph, it is understood that the criterion requires demonstration only if DNB occurs in more than %10 of the total number the fuel rods in the core.</p> <p>About fulfilment of this criterion for “Instantaneous seizure or break of one reactor coolant pump set shaft” the following statements were provided (instead of referring to Chapter 4):</p> <p><i>“In performing calculation analysis assessment of boundary value of relative heat rate of fuel rod of the core at which DNB does not occur and the minimum DNBR is not more than 1.0 is carried out. The results of the assessment showed that DNB does not occur when relative heat rate of fuel rods with regard for engineering safety factor is less than 1.46 (Kr·Keng).</i></p> <p><i>According to relative heat rate selection of fuel rods distribution (with regard for engineering coefficient) the number of fuel rods for which</i></p>
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	<p><i>DNB can take place with accepted assumptions, does not exceed 7.14 % of total number of fuel rods in the core.</i></p> <p><i>At the same time the calculations provided in [1] demonstrate that DNB does not result in depressurization of fuel rods.”</i></p> <p>So, there is some information presented without any justification.</p> <p>It has been specified that the DNB occurs on the fuel rods with relative heat rate with regard for engineering safety factor is over a pre-determined value. However, there is no explanation on how this value is determined.</p> <p>There is also no explanation on how to achieve the result on the amount of fuel rods which experience DNB.</p> <p>Also, it is stated that no fuel damage occurred despite the existence of DNB.</p> <p>And upon additional information request of NDK, a report including fuel behaviour analysis under this transient is presented (and referred as [1]). However, there is still quite a bit of information missing and NDK will request additional new information.</p> <p>NNSA:</p> <p>According to Chinese regulations and standards, the acceptance criteria for the operation conditions of category IV is that the radioactive consequences meet the limit requirements of GB6249. One of the commonly used decoupling criteria for radiological consequence evaluation is 10% fuel burn out share. It is generally believed that if DNB occurs in the fuel, it is considered that the fuel rod is damaged. If the accident analysis shows that the DNB share under the accident does not exceed 10%, there is no need to conduct a subsequent radiological consequence evaluation specifically for this initiating event. Decoupling criteria are not mandatory.</p> <p>In the PASR report of China’s VVER1200 project, the share of DNB occurrence is not given for Category IV operating conditions, but the decoupling criterion is adjusted. 10% fuel burn out share is still considered as one of the decoupling criteria for radioactive consequence evaluation, but it is not assumed that DNB occurs when fuel damage appears. Instead, it is believed that <i>Report on Fuel Rod Mechanical Design Method</i> has been provided to explain the judgment method for fuel rod unsealing. In actual calculation, investigate local cumulative axial strain of the fuel cladding to evaluate whether the fuel rod has lost its seal.</p> <p>During the review by the Chinese side, a detailed review was conducted on the acceptance criteria of the Russian side, and the evaluation method</p>
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	for fuel rod unsealing during PSAR stage has always been a remaining issue for the review.
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HAEA Question 3	The analysis of unjustified startup of the ECCS is not complete
Description	<p>In chapter 3.1.4, Table 3.1, the PSAR names the following events as fulfilment of the requirement to take into account the unjustified startup of the ECCS in DBC2:</p> <ul style="list-style-type: none"> • Inadvertent actuation of emergency core cooling system hydroaccumulator • Malfunction in chemical and volume control system leading to increase in primary coolant inventory <p>The Table and the PSAR does not take into account the unjustified startup of the high pressure (JND) and low pressure (JNG-1) ECCS.</p>
Requirement	<p>3a.2.2.4500. Among the various DBC1-4, at least the following internal events shall be taken into account in the design of the nuclear power plant:</p> <p>[...]</p> <p>c) DBC2 operating conditions:</p> <p>[...]</p> <p>co) unjustified startup of the emergency core cooling system,</p> <p>[...]</p>
Comments	<p>STUK:</p> <p>Spurious startup of ECCS is expected to be analysed.</p> <p>YVL B.3 304. The inadvertent actuation of every system accomplishing a safety function shall be addressed as an initiating event.</p> <p>Updated PSAR 15 analyses are not yet received.</p> <p>SEC NRS:</p> <p>According to Russian requirements, analyses results shall be provided in SAR for each initial events included in the DBA initial events list</p> <p>Reference:</p> <p>NP-082-07, p. 2.1.8: “A list of initial events for design basis accidents and a list of beyond design basis accidents, classification of design basis and beyond design basis accidents according to the frequency of occurrence</p>

	<p>and severity of consequences, as well as analysis of design basis and beyond design basis accidents and their consequences should be established in the Reactor Facility and NPP design and presented in the NPP SAR. Among beyond design basis accidents, it is necessary to consider accidents with severe core damage.”</p> <p>It can be noted also the spurious startup of ECCS (except for the piston pump driven high pressure boron injection system, which may be or may not be the part of ECCS depending on design) could influence the primary circuit only in plant states with low primary pressure, because ECCS pumps head is significantly lower than nominal primary pressure.</p> <p>NDK:</p> <p>Partially agreed.</p> <p>Since this transient may not be a concern of each design, this subject should be discussed specifically for each different design.</p> <p>Inadvertent operation of ECCS high pressure/low pressure pumps is not considered for Akkuyu design either. I also think that this transient should be excluded. Because this is not a case in Akkuyu design, as ECCS HP/LP pumps are not capable of delivering flow to the reactor when the primary circuit is at nominal pressure.</p> <p>Even so, justification on exclusion of this condition needs to be submitted by the licensee.</p> <p>NNSA:</p> <p>Abnormal startup of ECCS system is considered in VVER1200 in China, and provides abnormal startup of emergency core cooling system under refuelling, shut down for maintenance, RP “cold” condition, unit heating and cooling conditions (see PSAR Rev. A15.1.25 for details). In normal operation conditions, nominal pressure of primary loop is 16.2Mpa, max pressure of high-pressure injection is 7.9MPa (JND), max pressure of low-pressure injection is 2.5MPa (JNG-1). ECCS injection pressure is too low to inject water into reactor core in unjustified startup of the ECCS condition, thus it is not necessary to perform such kind of analysis for this scenario.</p>
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HAEA Question 4	Hydro Accumulators System pipeline diameter was drastically reduced compared to the reference unit
Description	Compared to the reference unit (LNPP-II) the diameter of the hydro accumulator pipelines was drastically reduced. The reason for this change in the design of the NPP is not explained, however DSA

	calculations show that this design modification seems to meet all applicable criterion.
Requirement	
Comments	<p>STUK:</p> <p>There has been change in design concerning Hydroaccumulator line diameter for FH1 as well mentioned in PSAR 1.4. Reason for change has been mentioned is to facilitate manufacturing of the RPV and to improve its structural strength.</p> <p>Finalized analyses justifying the design solution acceptance has not yet been received. Analyses received so far show that there the modification seems to meet the criterion. There has been related TSO made analysis ordered by STUK on LBLOCA, and this issue needs further investigation.</p> <p>SEC NRS:</p> <p>According to p. 1.2.6 NP-001-15, all technical measures taken for safety shall be validated by the previous experience, research, prototype operation experience. According to the official comments to this point given in RB-152-18, this requirement shall be interpreted that if not all of the above validation components are used, there shall be justification that others are enough for validation.</p> <p>So, in this specific case, there is no contradiction to the requirements as the research done by the calculations justified that this design decision is safe.</p> <p>As this question is about the design decision that differs from the reference design of LNPP-2, we can't provide any specific comments.</p> <p>NDK:</p> <p>If the criteria are met and if there is no problem arising from this change in the analyses, it is considered that there is no issue that needs to be questioned further.</p> <p>Additional information for Akkuyu NPP design:</p> <p>The same difference in hydroaccumulator nozzle diameter reduction exists between Akkuyu NPP and its reference NPP.</p> <p>NNSA:</p> <p>The size of main safety injection pipeline from safety injection tank to pressure vessel of JNG-2 system (passive part of emergency core cooling system) of TNPS units 1-4 and unit 7&8 is 351x36, with an inner diameter of 279. TNPS unit 1&2 adopts alloy steel with the weld deposited cladding, while TNPS unit 3&4 and unit 7&8 adopt stainless steel.</p>

	According to the safety analysis in PSAR Chapter 15, the mass flow rate of ACC can meet the ECCS criteria.
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HAEA Question 5	Decay heat removal process by Low Pressure ECCS during maintenance water levels is not well understood
Description	It is described in the Chapter 6.1.5 of the PSAR, that during maintenance water levels inside the reactor vessel is reduced to the level of the top of the cold leg. Then the removal of the decay heat by the Low-Pressure Safety Injection System is performed in such a way that the pump is fed from the cold leg and injecting into the hot leg, which is exactly opposite to the natural circulation as it is arranged at VVER-440 NPPs in such cases. This arrangement raises some questions that should be answered. How effective is this direction of cooling taking from the cold leg instead of the hot leg? How is the distribution of the cooled down water added to the top of the water surface? These processes should be included in the models as well.
Requirement	
Comments	<p>STUK:</p> <p>PSAR chapters concerning maintenance situations have not yet been received. JNA system description presents similar functionality as presented in the question. For practical reasons it would be worthwhile to point out to user experiences in this regard.</p> <p>SEC NRS:</p> <p>According to p. 1.2.6 NP-001-15, all technical measures taken for safety shall be validated by previous experience, research, prototype operation experience. According to the official comments to this point given in RB-152-18, this requirement shall be interpreted that if not all of the above validation components are used, there shall be justification that others are enough for validation.</p> <p>So, in this specific case, there shall be justification in SAF that this design decision is safe.</p> <p>In the case of LNPP-2, there were calculations results provided in the SAR and supporting documents to justify that this design decision ensures the decay heat removal in case of accident, including the loss and recovery of the LP ECCS.</p>

	<p>NDK:</p> <p>The adequacy and effectiveness of the design need to be demonstrated in all operating modes, and all necessary information must be provided.</p> <p>Additional information for Akkuyu design:</p> <p>The same cooldown mode is also used for Akkuyu NPP. For Akkuyu NPP, information on water levels in different operational states is available in the PSA document. Unfortunately, information related to the questions raised up by Hungary is also not available for Akkuyu NPP.</p> <p>NNSA:</p> <p>In Tianwan unit 7&8, the phenomena in question above is normal during operation.</p> <p>When the water level of the first loop is lower for maintenance, before the water level is lower below the main joint surface of the reactor, the first loop residual heat removal system is switched from the planned cooling condition to the maintenance cooling condition. At this time, the circulation direction of the core cooling circuit is changed, and the residual heat removal system will take water from the cold leg inject into the hot leg after cooling. Then, the coolant is injected into the core from the top of the core to remove core residual heat. During the switching process, the residual heat removal system is on operation. This method is only allowed 48 hours after reactor shutdown.</p>
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HAEA Question 6	Hydro Accumulators adiabatic gas expansion problems
Description	Hydro Accumulators have more water volume and less Nitrogen gas volume compared to VVER-440 design. This raises concerns regarding the adiabatic expansion of the gas in case of a fast discharge of the HAs, which could cause PTS in the upper vessel walls or maybe freezing of the water, clogging etc.
Requirement	
Comments	<p>STUK:</p> <p>This issue may be connected to the design change in the hydroaccumulator pipeline diameter and with slower discharge. Volume ratio is approximately 5:1. Expectation from STUK is that dimensioning, and volumes should be justified. Freezing and related phenomena is not expected due to stored heat in water and structures (not fully thermally isolated expansion), but this topic will be further studied.</p>

	<p>SEC NRS:</p> <p>According to p. 1.2.6 NP-001-15, all technical measures taken for safety shall be validated by previous experience, research, prototype operation experience. According to the official comments to this point given in RB-152-18, this requirement shall be interpreted that if not all of the above validation components are used, there shall be justification that others are enough for validation.</p> <p>So, in this specific case, there shall be justification in SAR that this design decision is safe.</p> <p>In the case of LNPP-2, there were calculations results provided in the SAR and supporting documents that shows that although the nitrogen gas temperature in the hydro accumulators drops well below zero due to adiabatic expansion, the heat capacity of the remaining water and the metal of the hydro accumulator wall is much more than the heat capacity of the nitrogen gas, so this does not lead to any significant water freeze or PTS.</p> <p>NDK:</p> <p>The safety of the design must be justified. So, it would be helpful to request a calculation of the tank's response for the highest possible discharge rate.</p> <p>NNSA:</p> <p>The water volume and Nitrogen gas volume in ACC are the same between TNPS unit 7&8 and TNPS units 1-4. Considering the thermal capacity of the water in accumulator and the ACC wall, frozen or clogging may not happen when pressure decreases, but analysis on thermal-hydraulic for ACC pressure drop should be needed.</p> <p>In TNPS commissioning tests, a lower pressure test (2MPa) is arranged. But the purpose of this test is to identify the resistance coefficients of ACC line. When the pressure of ACC is decreased from 2MPa, the phenomena of frozen and clogging do not exist.</p>
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Appendix B – STUK questions

STUK Question 1	Improper consideration of containment isolation (localization) function
Description	<p>Containment isolation as safety function is not recognized correctly in system descriptions. The safety analyses are not in line with design presented in system descriptions.</p> <ul style="list-style-type: none"> • Documentation received in package 3 does not fulfil STUK regulation Y/1/2018 10§ paragraph 3 subparagraph c) point i. (<i>In order to ensure containment building integrity, i.e. the containment shall be designed to maintain its integrity during anticipated operational occurrences and, with a high degree of certainty, during all accident conditions</i>) • LBA-system does not have, in accordance with system description, any containment isolation valves. • the motor operated isolation valves of JND and JNG-1 are not, contradictory to YVL B.6 requirement 324 controlled by protection system. • in system descriptions of JND, JNG-1, JNA, LAR/LAS, LBA, JNK and KAA control of CIVs needs to be better explained, taking into account YVL B.6 requirement 329a (<i>The control function of the containment isolation valves shall be ensured with systems designed for severe reactor accident management. Ensuring shall be tolerant of a single failure for the pipes that are not automatically isolated in connection with an initiating event leading to containment isolation.</i>) • JNG-1 takes part in reaching safe state after severe accident. Systems functionality, maintainability and accessibility need to be justified as it is required in YVL B.6 requirements 340a, 340b and 340c. This means that environmental conditions after severe accident need to be assessed and systems functioning in this environment need to be justified. • in suction lines of JNG-1 from JNK-pool there is no isolation valve inside the containment, in accordance with PI-diagrams. This deviates from YVL B.6 requirements 323 and 324. • in JNA each division has two containment isolation valves that are both motor operated. Diversity requirement given in YVL B.6 requirement 323 is not fulfilled. • In JNK system description there is one manually controlled motor operated isolation valve in two JNE system suction lines. Sufficiency of one valve has been justified in a way that does not fulfil YVL B.6 requirements 323, 324 and 325.

<p>Requirement</p>	<p>STUK regulation Y/1/2018 10§ paragraph 3 subparagraph c) point i</p> <p>YVL B.6 requirements 323,324, 325, 329a, 340a, 340b, 340c</p> <p><i>323. The design shall allow reliable closing of every line penetrating the containment pressure boundary and communicating with the primary coolant or containment atmosphere. Such lines shall be equipped with at least two independent isolation valves in series. The diversity principle shall be applied to the isolation valves in series.</i></p> <p><i>324. The isolation valve according to requirement 323 shall be locked closed or have provisions to be closed automatically, in which case it shall be controlled by the plant's protection system or be of the passively closing type (check valve). There shall be at least one isolation valve both inside and outside the containment.</i></p> <p><i>325. Each pipeline penetrating the containment pressure boundary that is not connected to the primary coolant or directly connected to the gas space of the containment shall be provided with at least one isolation valve outside the containment.</i></p> <p><i>329. A check valve may not be used as a containment-external isolation valve.</i></p> <p><i>340a. Systems used in bringing the plant to a safe state following a severe reactor accident are not required to be independent from the systems designed for normal operation, anticipated operational occurrences and postulated accidents. When using systems that do not fulfil this independence in order to bring the plant to a safe state or to maintain it following a severe reactor accident, the operability of the systems shall be demonstrated in conditions following a severe reactor accident.</i></p> <p><i>340b. It shall be possible, if necessary, to restore the operability of the systems required to bring the plant to a safe state or to maintain it following a severe reactor accident in conditions following the accident.</i></p> <p><i>340c. Functions required to bring the plant to a safe state or to maintain it following a severe reactor accident shall be implemented by one or several systems. If the systems are located in facilities that are not accessible in conditions following a severe reactor accident, the single-failure criterion shall be applied to the function in terms of components requiring an external power source.</i></p>
<p>Comments</p>	<p>HAEA</p> <ul style="list-style-type: none"> • In our case the LBA system has containment isolation valves. • The motor operated isolation valves of JND and JNG-1 are automatically operated but also can operated manually.

	<ul style="list-style-type: none"> • In suction lines of JNG-1 from JNK-pool there is no isolation valve inside the containment in our PSAR also. • In our case the JNA each division has also two containment isolation valves that are both motor operated. • There is no JNE system, maybe it is a mistype. In the case of JNK the JNK11AA801 and JNK41AA801 isolation valves do not have pairs inside. The usage of one valve is not justified. • Related requirements: <ul style="list-style-type: none"> ○ <i>3a.4.6.1900. The pipelines penetrating the containment wall, directly connecting to the pressure boundary of the main circulation loop or the containment atmosphere shall be supplied with at least two fast activation isolation valves, operated reliably and independently by external energy, in serial arrangement, one placed within the containment and the other outside of it.</i> ○ <i>3a.4.6.1910. In the design of isolation function the following shall be ensured:</i> <ul style="list-style-type: none"> <i>a) in the case of double wall containment, its properties shall be considered,</i> <i>b) the isolation valves shall be placed as close as possible to the containment wall penetration,</i> <i>c) considering the consequences of a failed isolation function, leakage monitoring of the pipe section between the isolation valves shall be provided,</i> <i>d) functionality of the isolation valves and the penetrations shall be regularly inspected and tested,</i> <i>e) the operation of the isolation valves shall be sufficiently fast to effectively perform its function,</i> <i>f) the specification of the isolation valves shall be determined by taking into account all the relevant DBC and DEC plant states</i> ○ <i>3a.4.6.1920. Solutions different from 3a.4.6.1900. can be designed supported by deterministic analysis supplemented with probabilistic analysis.</i> ○ <i>3a.4.6.1925. The isolation valves operated by external energy shall be fast operated, automatic and equipped with remote operation, status signals shall be displayed in the main and in the backup control room, and it must be provided to perform its function in a fail-safe manner in case of loss of external energy or loss of operating medium.</i> ○ <i>3a.4.6.1930. The isolation valves operated by external energy shall be remotely activated. Status signals about their position shall be displayed in the main control room and in the backup control room.</i>
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	<ul style="list-style-type: none"> ○ <i>3a.4.6.1940. If an isolation valve other than 3a.4.6.1900. is applied, it shall be provided that the valve can be automatically operated remotely or lockable.</i> ○ <i>3a.4.6.1950. Within the containment a check valve can also be accepted, if the status of the valve is continuously monitored.</i> ○ <i>3a.4.6.1960. In the pipe section that are not directly connected either to the pressure boundary of the main circulation loop or the containment atmosphere, the isolation valve can be omitted if justified and if the integrity of the pipe section is ensured via a technical solution.</i> ○ <i>3a.4.6.1965. Deviation is allowed from 3a.4.6.1900. in case of impulse lines or in cases when it is justified by analysis that the isolation solution based on 3a.4.6.1900. would reduce the reliability of the safety system whose pipeline passes through the containment wall. In the latter case the use of check valve within the containment is acceptable as an alternative to isolation valve if the status of the valve is constantly monitored or the omission of the valve inside the containment is allowed if the integrity of the pipeline section is ensured via technical solution. In case of deviation from the 3a.4.6.1900. the appropriateness of the used alternative isolation valve shall be justified via deterministic and probabilistic safety assessments.</i> ○ <i>3a.4.6.1970. In the case of isolation valves operated by external energy, efforts shall be made to fulfil the function in a fail-safe design if the external energy or the operating medium is lost.</i> ○ <i>3a.4.6.1975. In case of pipelines passing through the containment wall, which are not directly connecting to the pressure boundary of the main circulation loop or the containment atmosphere, at least one isolation valve shall be used which is external energy operated and located outside the containment. In justified cases two, serial arranged, manually operated, locked in closed state valve can be acceptable, from which one shall be located inside the containment and the other outside.</i> <p>SEC NRS</p> <p>The requirements in Russia are similar in general, but diverse in some details.</p> <p>NP-001-15, p. 3.6.3 “Confinement safety systems must be provided for each NPP unit and perform the specified functions for design basis</p>
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	<p>accidents, as well as beyond design basis accidents accounted for in accordance with paragraph 1.2.1”</p> <p>NP-001-15, p. 3.6.5: “All communications that cross the boundaries of the containment, through which, in an accident, radioactive substances may escape beyond the boundaries of the confinement, must be equipped with isolation elements in accordance with the requirements of federal rules and regulations in the field of atomic energy use.”</p> <p>NP-010-16, p. 68 “All pipelines crossing or connecting to containment should be equipped with isolating devices installed as close as technically possible to the point of intersection of the containment. The number of isolating devices and their installation locations should be selected so that for any initiating event of a design basis accident and independent of the initial event failure of one of the safety system elements taken into account in the NPP design, isolation of all crossing pipelines for which this is provided for by the NPP design.”</p> <p>NP-010-16, p. 70 “It is allowed not to install isolating devices on pipelines laid through the containment or connected hereto and used for intake of working medium from the primary circuit or accident localization area rooms with further return to them, as well as for measurements during accident, provided that these pipelines and equipment connected by them meet requirements of these Rules for containment elements”</p> <p>There is no specific requirement to have different types of power for isolation valves (i.e., no requirement to have one motor-operated valve and one pressurized air operated valve on every line), but there is a general requirement to apply the diversity principle in the design of all safety systems (NP-001-15, p.3.1.9).</p> <p>NDK</p> <p>Related requirements used in Türkiye are listed below:</p> <p>Turkish Legislation</p> <p>REGULATION ON DESIGN PRINCIPLES FOR SAFETY OF NUCLEAR POWER PLANTS</p> <p><i>Article 19- (1) The plant is designed to be capable of retaining the bulk of the radioactive material that might be released from fuel during the entire range of accidents.</i></p> <p><i>(2) A containment system is designed unless this function is handled by other means. Containment system is designed to withstand high internal pressure, temperature and dynamic effects during accident conditions. For the accidents when containment system leakage is a major concern, entire piping systems contain automatic isolation systems. Pressure</i></p>
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	<p><i>differences through the containment upon an accident, shall not harm the systems that prevent serious accidents and containment interior.</i></p> <p><i>(3) The structure maintains its integrity in both short and long term under the pressure and temperature conditions that could prevail in design basis accidents. The confinement capability of meeting the design basis targets for limiting the leakage of any radioactive material is demonstrated.</i></p> <p>IAEA Requirements</p> <p>SSR-2/1 SAFETY OF NUCLEAR POWER PLANTS: DESIGN</p> <p><i>Requirement 56: Isolation of the containment</i></p> <p><i>Each line that penetrates the containment at a nuclear power plant as part of the reactor coolant pressure boundary or that is connected directly to the containment atmosphere shall be automatically and reliably sealable in the event of an accident in which the leak-tightness of the containment is essential to preventing radioactive releases to the environment that exceed acceptable limits.</i></p> <p><i>6.22. Lines that penetrate the containment as part of the reactor coolant pressure boundary and lines that are connected directly to the containment atmosphere shall be fitted with at least two adequate containment isolation valves or check valves arranged in series and shall be provided with suitable leak detection systems. Containment isolation valves or check valves shall be located as close to the containment as is practicable, and each valve shall be capable of reliable and independent actuation and of being periodically tested.</i></p> <p><i>6.23. Exceptions to the requirements for containment isolation stated in para. 6.22 shall be permissible for specific classes of lines such as instrumentation lines, or in cases in which application of the methods of containment isolation specified in para. 6.22 would reduce the reliability of a safety system that includes a penetration of the containment.</i></p> <p><i>6.24. Each line that penetrates the containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one adequate containment isolation valve. The containment isolation valves shall be located outside the containment and as close to the containment as is practicable.</i></p> <p>Russian Federation Regulations</p> <p>NP-010-16 "RULES OF DESIGN AND OPERATION OF LOCALIZING SAFETY SYSTEMS OF NUCLEAR POWER PLANTS"</p> <p><i>Isolating devices</i></p>
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68. All pipelines crossing or connected to the containment shall be fitted with isolating devices to be installed as close as technically possible to the crossing location with containment. Number of isolating devices and installation locations shall be selected so that isolation of all pipelines crossing the containment, for which this is stipulated by the NPP design, shall be provided in case of any initiating event of design-basis accident and initiating event-independent failure of on safety system elements considered in the NPP design.

69. Permissibility of omission of equipment with isolating devices of pipelines not connected to the reactor plant pipelines and equipment or atmosphere of the accident localization area rooms and protected against potential external and internal impacts in case of accidents, shall be justified in the NPP design.

70. It is allowed not to install isolating devices on pipelines laid through the containment or connected hereto and used for intake of working medium from the primary circuit or accident localization area rooms with further return to them, as well as for measurements during accident, provided that these pipelines and equipment connected by them meet requirements of these Rules for containment elements.

71. Installation of pipeline hand-operated valves equipped with lock or welded plugs can be provided as isolating devices on pipelines used only during repair with reactor shut down.

72. If the need arises to close isolating devices, these shall be closed for such time that release of radioactive substances to the environment does not result in excess of design-basis limits and criteria of safety.

73. The NPP design shall specify conditions for each of active isolating devices at the border of accident localization area, under which relevant isolating device closing signal shall be generated.

74. The NPP design shall justify for each isolating device the design-basis value of leakage through it beyond the accident localization area border in closed position of the isolating device.

75. In case of compressed air pressure loss pneumatic isolating devices shall turn to position for performance of safety function.

76. Isolating devices being active elements shall actuate automatically based on conditions determined in the NPP design.

77. The NPP design shall provide for measures to exclude unauthorized opening of isolating devices both during an accident, and in post-accident period, also in case of the drive power failure.

78. Pipeline valves used as isolating devices shall comply with the requirements of the Federal standards and rules in the field of atomic

energy use, governing design, manufacture, tests, installation and operation of pipeline valves for nuclear power plants.

79. No check valves are allowed to be used as isolating devices.

IAEA Guidelines

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

4.86. Piping crossing the containment walls should be equipped with containment isolation devices and devices necessary to isolate leaks in the external recirculation loops to maintain sufficient water inventory for cooling. Non-isolatable leakage (e.g. between the containment penetration and the isolation valve) should be prevented by design (e.g. by the provision of a guard pipe).

4.101. The external extensions of the penetrations should be installed in a confined building, at least until the first isolation valve, to collect and filter any leaks before a radioactive release occurs.

4.102. Leak rates of isolation devices, air locks and penetrations should be specified, with account taken of their importance to safety and to the integral leak-tightness of the containment.

4.103. The design should include appropriate isolation devices to ensure the isolation of the containment in the event of an accident, as described in paras 4.154–4.166.

4.154. Each line that penetrates the containment that is not part of a closed loop and that either (a) directly communicates with the reactor coolant during normal operation or in accident conditions or (b) directly communicates with the containment atmosphere during normal operation or in accident conditions should be provided with two isolation valves in series. Each valve should normally be closed or should have provisions to close automatically. If the line communicates directly with the reactor coolant or the containment atmosphere, one valve should be provided inside the containment and one valve outside. Each valve should be reliably and independently actuated. Isolation valves should be located as close as practicable to the containment.

4.155. Loops that are closed either inside or outside the containment should have at least one isolation valve outside the containment at each penetration. This valve should be an automatic valve, a normally closed valve or a remotely operated valve. If the failure of a closed loop is assumed as a postulated initiating event or as a consequence of a postulated initiating event, this recommendation will apply to each line of the closed loop.

4.156. *Loops that are closed both inside and outside the containment envelope should have at least one isolation valve, an automatic valve, a normally closed valve or a remotely operated valve located outside and as close as practicable to the containment at each penetration.*

4.157. *Small dead-ended instrumentation lines that penetrate the containment should have at least one isolation valve outside the containment.*

4.158. *Containment isolation valves for instrumentation lines that are closed (i.e. not in communication with the atmosphere) are not necessary, provided that the lines are designed to withstand the accident conditions for which confinement is necessary. The rooms where these lines emerge should be equipped with a filtration-ventilation system to maintain subatmospheric pressure. Such rooms and the equipment within them should be designed to withstand increased levels of temperature and humidity due to possible leakage from these lines.*

4.159. *The need for an automatic isolation of the containment in accident conditions should not prevent the systems necessary to mitigate those accidents from accomplishing their intended functions.*

4.160. *Overpressure protection should be provided for closed systems that penetrate the containment and for isolated parts of piping that might be overpressurized by an increase of the temperature inside the containment atmosphere in accident conditions.*

4.162. *For specific operational conditions (e.g. conditions with an open containment or inhibited containment automatic isolation), the risk to safety should be assessed and temporary provisions should be implemented as necessary to ensure the containment isolation function can be accomplished in a timely manner.*

4.163. *Particular consideration should be given to the containment isolation features of the following systems that potentially could create a bypass of the containment:*

- a) *Systems designed for removing heat from the core, from the core debris or from the containment that can transport radioactive material outside the containment in accident conditions.*
- b) *Systems that can transport airborne radioactive material from the containment atmosphere to outside the containment in accident conditions (e.g. systems used in some designs to mix the atmosphere inside the containment to prevent the ignition of hydrogen).*
- c) *Supporting systems or auxiliary systems (inside the containment) for which, in the event of leakage, fluids with a high activity might be released outside the containment (in some designs, the*

	<p><i>component cooling water system, the containment sump purge system or the sampling systems).</i></p> <p><i>4.164. Systems connected to the primary circuit in normal operations (i.e. primary circuit filtration systems or, in some designs, the chemical and volume control system) and systems connected to the containment atmosphere should be automatically isolated in accident conditions when they are not necessary for safety.</i></p> <p><i>4.165. To achieve the objective of limiting any radioactive release outside the containment, the isolation devices should be designed with a specified leak-tightness and closure time.</i></p> <p><i>4.166. Design provisions for leakage tests (e.g. nozzles, instrumentation test lines) should be made such that each isolation valve can be tested.</i></p> <p><u>Comments for each item:</u></p> <ul style="list-style-type: none"> • LBA-system does not have, in accordance with system description, any containment isolation valves. <p>LBA-system will provide a path to the environment only if an accident where simultaneous coolant leak from primary to secondary circuit and steam pipeline break between containment wall and steam isolation valve take place. Besides, the release will be mitigated to some extent by the deposition of radioactive particles in the reactor building or its premises (steam chamber).</p> <p>SAR should include a radiological consequence analysis of this specific case justifying the acceptability of the design solution against the relative acceptance criteria.</p> <p>In Akkuyu design, LBA system does not have any specific containment isolation valves as your case.</p> <p><i>(On the other hand, steam isolation can be ensured by SG PORV, BRU-A, motor operated isolation valve upstream the BRU-A, FSIV and motor operated isolation valve downstream the FSIV in the LBA system. Although there is a disputable issue that they are not close enough to the wall of containment, according to SSR-2/1 Article 6.24, it may seem reasonable.)</i></p> <ul style="list-style-type: none"> • the motor operated isolation valves of JND and JNG-1 are not, contradictory to YVL B.6 requirement 324 controlled by protection system. <p>It does not feel right for a protective safety system to include a CIV and the mentioned requirement applies for CIVs.</p> <ul style="list-style-type: none"> • in system descriptions of JND, JNG-1, JNA, LAR/LAS, LBA, JNK and KAA control of CIVs needs to be better explained, taking into
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	<p>account YVL B.6 requirement 329a (The control function of the containment isolation valves shall be ensured with systems designed for severe reactor accident management. Ensuring shall be tolerant of a single failure for the pipes that are not automatically isolated in connection with an initiating event leading to containment isolation.)</p> <p>JND, JNG-1, LAR/LAS, JNK: It does not feel right for a protective safety system to include a CIV and the mentioned requirement applies for CIVs. (Extra: Systems with same functions but different designs exist in Akkuyu NPP)</p> <p>LBA: This system doesn't include any CIVs.</p> <p>JNA, KAA: The requirement should be fulfilled.</p> <ul style="list-style-type: none"> JNG-1 takes part in reaching safe state after severe accident. Systems functionality, maintainability and accessibility need to be justified as it is required in YVL B.6 requirements 340a, 340b and 340c. This means that environmental conditions after severe accident need to be assessed and systems functioning in this environment needs to be justified. <p>Agreed.</p> <ul style="list-style-type: none"> In suction lines of JNG-1 from JNK-pool there is no isolation valve inside the containment, in accordance with PI-diagrams. This deviates from YVL B.6 requirements 323 and 324. <p>It does not feel right for a protective safety system to include a CIV and the mentioned requirement applies for CIVs. Besides: Flow is into the containment. An isolation valve may be needed if a reverse flow occurs for a reason (such as emptying of the tank or a suction line break) and there exist one which can be used if needed.</p> <p>(Extra: For the system with the same function existing in Akkuyu NPP, the water source is different, and the suction line doesn't cross the containment boundary.)</p> <ul style="list-style-type: none"> In JNA each division has two containment isolation valves that are both motor operated. Diversity requirement given in YVL B.6 requirement 323 is not fulfilled. <p>The requirement should be fulfilled.</p> <p>(Extra: For such systems in Akkuyu design, motor operated valves are installed inside the containment, whereas pneumatic-driven valves are installed outside the containment.) (The Akkuyu design has a low-pressure pump of JNA, which has the same function as the JNG-1 in your design. In emergency mode, water to JNG-1 (suction lines of JNA LP</p>
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pump) is supplied from the spent fuel pool and the sump. In cooldown mode, water to JNG-1 is supplied from specified primary circuit loops. There are motor operated isolation valves that are installed both inside and outside the containment. Therefore, diversity requirement is not fulfilled, and it is needed to place isolation valves which have different working principles from each other.)

- In JNK system description there is one manually controlled motor operated isolation valve in two JNE system suction lines. Sufficiency of one valve has been justified in a way that does not fulfil YVL B.6 requirements 323, 324 and 325.

In Akkuyu design, JNK and JNE systems do not exist.

NNSA

STUK question No.1 item 1:

STUK proposed that the main steam system (LBA) does not have containment isolation valves inside and outside the containment.

The review team believes that in HAD 102/06 and the standard review program (NUREG-0800) section 6.2.4 of the containment isolation system, it is mentioned that “The pipes penetrating the containment boundary are neither part of the pressure boundary of the reactor coolant, nor it is not directly connected to the containment gas. At least one locked manual remote-control valve or automatic isolation valve must be configured on the outside of the containment.”

This can be used as a basis for the main steam system not to install a containment isolation valve inside the containment. In addition, the main steam isolation valve can be used as the isolation valve of the main steam system on the outside of the containment. This valve isolates the leaking steam generator under SGTR and plays an isolation role. In other cases, the isolation valve needs to be kept normally open to bring out the core heat.

Therefore, the review team believes that the main steam system (LBA) need not be equipped with a containment isolation valve inside the containment. The main steam isolation valve can be used as an isolation valve for the main steam system on the outside of the containment, meeting the requirements of relevant domestic regulations and guidelines.

STUK question No.1 item 2:

All JNG-1 isolation valves have isolation valves that are not controlled by the containment isolation signal and maintain the valves open after the accident. 6.2.4 post-accident state.

	<p>STUK believes that the electric isolation valve and protection system control of the high-pressure safety injection system (JND) and low-pressure safety injection system (JNG-1) does not contradict the requirements of Article 324 of YVL B.6. Article 324 of YVL B.6 requires "The isolation valve required by 323 should be locked or have automatic closing provisions, in this case it should be controlled by the protection system of the nuclear power plant, or it should be a passive closing type (check valve) There should be at least one isolation valve inside and outside the sealing device.</p> <p>The review team believes that in the standard review outline (NUREG-0800) section 6.2.4 of the containment isolation system, it is mentioned that "used for special safety facilities or special safety facilities. The containment isolation device on the pipeline of the related system usually consists of two isolation valves in series. If it can be proved that only one isolation valve on the pipeline can adapt to the failure of a single active component, then this single isolation valve solution is acceptable. "This is not inconsistent with my country's current requirements and practices.</p> <p><u>STUK question No.1 item 3:</u></p> <p>In China’s VVER1200 project,</p> <p>JNG-1 does not have functional requirements in severe accidents.</p> <p>STUK believes that “taking into account the requirements of Article 329a of YVL B.6: The control function of the containment isolation valves shall be ensured with systems designed for severe reactor accident management. Ensuring shall be tolerant of a single failure for the pipes that are not automatically isolated in connection with an initiating event leading to containment isolation. The system descriptions of JND, JNG-1, JNA, LAR/LAS, LBA, JNK and KAA need to be better explained. The review team believes that the requirements of Article 329a should actually be met.</p> <p><u>STUK question No.1 item 4:</u></p> <p>STUK believes that "JNG-1 needs to demonstrate its availability, system functionality, maintainability and accessibility under severe accidents." The review team agreed with this statement. In the domestic nuclear power project review, we also paid attention to the system's availability, system functionality, maintainability, and accessibility under serious accidents.</p> <p><u>STUK question No.1 item 5:</u></p> <p>In China’s VVER1200 project, the valves mentioned by STUK can be used for the containment isolation when used as containment isolation valves.</p>
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	<p>STUK believes that "JNG-1 was directly cited from the JNG pool. The pit pool is located at the bottom of the containment vessel. Therefore, there is no isolation valve inside the shell, and the isolation valve outside the shell does not meet the requirements of Article 323 and Article 324 of YVL B.6." This situation of the review team is acceptable but requires that the upstream pipeline between the isolation valve and the containment should have a protective sleeve.</p> <p><u>STUK question No.1 item 6:</u></p> <p>STUK believes that "each zone of JNA is isolated from the containment by two electric isolation valves, which does not meet the diversity requirements".</p> <p>In China's regulatory and standard system, the requirements mentioned above are not mandatory.</p>
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STUK Question 2	Questionable separation of Defence-in-Depth for JNG-1 Heat exchangers
Description	The heat exchangers in JNG-1 have functions in Defence-in-Depth levels 1 to 4. The heat exchangers are used in normal operation in cooling the plant to cold shutdown, as well as in anticipated operational occurrences. In anticipated accidents the heat exchangers function in emergency heat removal chain and in severe accidents they are used to cool the core catcher. In accordance with STUK regulation STUK Y/1/2018 9§ paragraph 3 the levels of defence required under the Defence-in-Depth principle shall be as independent of one another as is reasonably achievable.
Requirement	STUK Y/1/2018 9§ paragraph 3 <i>The levels of defence required under the defence-in-depth principle shall be as independent of one another as is reasonably achievable.</i>
Comments	HAEA The situation is the same in our PSAR, the mentioned heat exchanger is used in more DiD levels. Requirements: <u>Govt. Decree 118/2011 (VII. 11.) 7 § (4a):</u> <i>The independence of the levels of Defence-in-Depth shall be ensured to the extent reasonably achievable.</i>

	<p><u>Govt. Decree 118/2011 (VII. 11.) Annex 3a:</u></p> <p><i>3a.3.1.0300. In order to ensure a greater independence, it shall be systematically implemented to a reasonable extent that irrespective of the initiating events, a safety system, structure or component can be associated with only one specific level of the Defence-in-Depth.</i></p> <p>SEC NRS</p> <p>Differences in requirements.</p> <p>According to the Russian requirements, this is allowed as soon as the fulfilment of safety functions is justified. NP-001-15 p.3.1.13 requires that “Multipurpose use of safety systems and their elements must be justified. Combining safety functions with normal operation functions should not lead to a violation of NPP safety requirements and a decrease in the required reliability of safety functions performance.”</p> <p>NDK</p> <p>In Akkuyu design, there are redundant HXs for the corresponding system and there exists diverse systems. Besides JNG-1 system, there are some other disputable systems regarding independence of level of defence in depth such as SG PHRS, Containment Sprinkler System.</p> <p>As in your provision, IAEA documents propose the approach for the <i>Defence-in-Depth</i> levels independence implementation based on “reasonably achievable”, in other words, such approach provide some flexibility to applicant/licensee in terms of design.</p> <p>NNSA</p> <p>STUK question No.2:</p> <p>The review team recognized the principle that "the various levels of <i>Defence-in-Depth</i> must be as independent as possible from each other". However, as a passive facility, the heat exchanger of JNG-1 is subject to normal operation, expected operational events, DBC and DEC. It works, and its function does not conflict under different working conditions, we think it is acceptable.</p>
<p>Action</p>	<p>The regulatory body has requested that as according to the documentation received so far, the JNG-1 heat exchangers are used in DiD levels 1 to 4, the approach shall be justified. In the justification the fulfilment of obligatory STUK regulation (STUK Y/1/2018 9§ paragraph 3) shall be assessed and justified.</p>

Appendix C – NNSA question

NNSA Question	In PSAR 15.5, Fuel cask drop accident is not properly analysed.
Description	<p>In VVER 1200 project of China, max drop height of spent fuel cask during on-site transfer is 26m, it is far beyond the acceptance criteria 9m. Impact limiting layer on ground is designed to absorb impact force during fuel cask drop accident, so radio-logic consequences analysis is not performed even when cask drop height exceeds 9m (about 30 foot). Instead, fuel cask integrity is theoretically analysed using ANSYS software.</p> <p>But review staff believe, regulatory requirement is not met by only performing integrity analysis.</p>
Requirement	Requirement in SRP (standard review plan): NUREG-0800 Section 15.7.5 requires that during handling and transferring process of the spent fuel container in the plant, if the maximum possible drop height or configuration of damping device exceeds the test range, a radioactive consequence analysis is required to be conducted.
Comments	<p>HAEA:</p> <p>In Hungary there are no specific criteria to this case. The calculated maximum drop height in case of the planned Hungarian VVER-1200 is also 9 meters.</p> <p>Related to this, the licensee presented the following in the PSAR:</p> <p>“The design of the cask TK-13 satisfies the recommendations of the IAEA, including for the accidents with cask falling from a height of 9 meters on a rigid base. Due to the fact that the lifting height of the cask is bigger than 9 meters, the NPP design provides shock absorbers that reduce the impact on the cask, when dropped, to values less than the impact on cask, when it is dropped from a height of 9 meters on a rigid base. However, when the cask is dropped, as a result of impact on the fuel assembly, it is possible for fuel assembly to be deformed.”</p> <p>There is no information in the PSAR on the maximum drop height and the fuel cask is licensed in a separate procedure later than the construction license procedure.</p> <p>SEC NRS:</p> <p>According to the p. 2.3.6 NP-061-05, the rise height of the transport cask shall be as low as possible. The maximum allowed rise height shall be</p>

justified in the design. According to the p. 2.3.7 NP-061-05, the rise above the allowed height foreseen in the design is also allowed in case of at least one of the following requirements is met:

- the step-by-step rise if foreseen in such a way that the actual height is not above the maximum allowed on each step,
- there rise is done above the damper or with the use of damping device, which lowers the load to the level equal to the load arising with drop from the design height to the hard base,
- the independent, full load capable secondary fall arrest system is foreseen.

So, in this specific case, there shall be justification in SAR that this design decision is safe.

In the case of LNPP-2, the technical measures foreseen in the design to ensure the spent fuel cask integrity in case of drop. There is a damping asphalt-concrete-sand cushion under the whole cask transportation path. There is a justification provided in the SAR and supporting documents that shows that the mechanical loads to the cask in the case of drop to the cushion from 27 m are equal to the loads in the case of drop from 9 m to the hard base. This is allowed according to Russian regulations (p. 2.3.7 NP-061-05, second bullet).

NDK:

Obviously, the criterion in NUREG-0800 is not met.

Also, even if it is demonstrated that in case of drop of the cask at a height more than 9 m the integrity is preserved, radiological analyses should be presented in order to identify possible consequences.

Additional information for Akkuyu NPP design:

No analysis of this condition is given in the safety analysis report. And instead,

- It is indicated that the drop of spent fuel cask is prevented (with a probability of drop of separate FAs, bottles, casks with FAs during fuel handling not more than 10⁻⁷/year) and this is ensured by the equipment design.
- While describing the design features available to achieve this, it is stated that the transport at a height up to 9 m is permissible and transportation cask was designed against a drop from this height.
- Also, it is stated that in the places of lifting the cask at a height exceeding 9 m; to exclude the failure of FA and fuel rods, shock-absorbers are present to be used to decrease the loads in case of the drop.

	<p>STUK:</p> <p>There is no set height limit in regulation in Finland that indicates when radiologic consequences analysis would be needed. Firstly, the aim is that the container should stay intact but potential consequences need to be analysed to assess threat and for emergency preparedness purposes. STUK has not yet received documentation that would present and justify the issues mentioned in the question.</p> <p>Related requirements in YVL D.3 and YVL C.5:</p> <p>YVL D.3 470. Transfer routes shall be kept short and lifts low. [2020-03-17]</p> <p>YVL D.3 471. At a nuclear fuel storage facility and encapsulation plant, transfers of heavy or otherwise dangerous objects shall be avoided in areas where a dropped load or a malfunction can damage the nuclear fuel or a component or structure important to safety. [2020-03-17]</p> <p>YVL D.3 472. The systems used for handling nuclear fuel shall be single-failure tolerant insofar as they might cause spent nuclear fuel damage or other significant radiation exposure in the event of a failure. [2020-03-17]</p> <p>YVL D.3 473. As a result of the failure, nuclear fuel handling systems shall enter a state that is favourable from the perspective of safety. [2020-03-17]</p> <p>YVL D.3 474. The nuclear fuel transfer device shall be equipped with safety functions in accordance with requirement 523 of the Guide YVL E.11. The grabs of hoisting device units involved in transferring the nuclear fuel shall be designed in accordance with requirement 525 of the Guide YVL E.11. [2020-03-17] [These requirements concern for example failure tolerance].</p> <p>YVL D.3 475. If the equipment transferring a nuclear fuel transfer cask or disposal canister cannot prevent dropping, the transfer cask or disposal canister shall be able to retain its leak-tightness in the postulated drop with a high degree of confidence. [2020-03-17]</p> <p>YVL C.5 309. The radiation situation of the plant and site area shall be analysed to plan the operations and radiation protection to be initiated during emergency situations. [2013-11-15]</p>
<p>Additional NNSA comment</p>	<p>China's VVER 1200 project applicant points out that there was a drop height of over 9 m during the transfer process of the spent fuel transportation container. When it was lifted at the gantry crane, the maximum height was 26m. In the design, in order to reduce the load borne when falling from 26m to the allowable value, the damping device was designed on the ground below gantry crane to form a buffering effect. In the meantime, an analysis on the drop of the spent fuel transportation</p>

	<p>container was conducted and <i>Report on Analysis of Drop of Spent Fuel Transportation Container of TNPS</i> was prepared. The conclusion of the report is as follows:</p> <p>The stress analysis results of transportation container falling from 26m to the buffer layer under 4 conditions are all lower than the stress of falling from 9m to hard ground.</p> <p>During the process of falling from 26m height to buffer layer, transportation container can maintain its structural integrity without damage.</p> <p>The report mentioned above is being improved, and the issue is currently under review as a remaining issue during the PSAR stage.</p>
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Appendix D – NDK questions

NDK Question 1	Demonstrate that subcriticality is achieved against the highest possible positive reactivity
Description	<p>According to SSG-2 (Rev.1) / Article 7.27, subcriticality needs to be maintained during and after anticipated operational occurrences or design basis accident conditions. It is expected that the most challenging conditions for subcriticality should be covered in the analyses of PSAR Chapter 15. As it is known and stated in Chapter 4, the highest possible positive reactivity in the reactor core occurs when the plant is in the cold state. However, Chapter 15 analyses are generally performed when plant state is in hot state.</p> <p>In line with the above-mentioned remarks, it is necessary to demonstrate that how the subcriticality is maintained against the highest positive reactivity under the most conservative DBA conditions, which are listed below.</p> <ul style="list-style-type: none"> • Plant is in cold state or transition period from hot to cold state. • Boron concentration is zero (EOC). • Loss of offside power (LOOP) is considered. • Due to LOOP, chemical and volume control system (KBA) pumps are unavailable. This causes the boron concentration in coolant to not be increased by KBA. • Sticking in the upper position of one RCCA with maximum efficiency is considered for scram.
Requirement	<p>SSG-2 (Rev. 1) Deterministic Safety Analysis for Nuclear Power Plants</p> <p><i>“7.27 - The conservative analysis of anticipated operational occurrences and design basis accidents should demonstrate that the safety systems alone in the short term, along with operator actions in the long term, are capable of achieving a safe state by fulfilling the following safety conditions:</i></p> <p><i>(a) Shut down the reactor and achieve subcritical condition during and after anticipated operational occurrences or design basis accident conditions;....”</i></p>
Comments	<p>SEC NRS:</p> <p>SEC NRS: Russian regulation NP-006-16 «Requirements for the content of safety analysis reports for nuclear power plant units with VVER reactors» requires analysing the following initiating events with changes of reactivity:</p>

- ejection of control rod (CR),
- uncontrolled withdrawal of the group of CRs from the reactor core with the operating rate,
- misalignment of the CR (falling of CR, erroneous withdrawal of CR, static misalignment of CR),
- connection of a non-operating loop of the reactor coolant pipeline without prior power reduction,
- Inadvertent dilution of boric acid in the primary circuit coolant,
- error in the course of fuel assembly loading into the reactor,
- human error in the course of xenon oscillations suppression (movement of CR resulting in the maximum possible deformation of the power density field).

The NPP design shall provide for protective safety systems aimed to assure reliable emergency shutdown of the reactor and its maintenance in sub-critical state in case of any operational occurrences up to design basis accidents inclusive.

The minimum required sub-criticality margin for the promptly shut-down reactor shall be substantiated for different moments of the campaign with due regard for uncertainties of such margin and experimental verification in operating reactors.

The justification of the most conservative DBA conditions is performed by licensee. It should be demonstrated that SCRAM efficiency is enough to make the reactor subcritical.

For Russian VVER-1200 the analysis of CR and group of CR withdrawal were performed for nominal power, reduced power and zero power levels both at the beginning and at the end of fuel cycle. According to the operating instructions the cold state of the reactor can be reached only after the boron concentration in coolant is increased to the certain level. The operation of reactor in the cold state with zero boron concentration is not allowed.

The SEC NRS experts performed the audit calculation of VVER-1200 SCRAM-efficiency. The results of calculations are presented in the table and confirmed that subcritical condition during and after anticipated operational occurrences or design basis accident conditions will be achieved.

Efficiency of SCRAM $\Delta k_{3\phi}/k_{3\phi}$,	Zero power RAINBOW-TPP (BIPR-7A)	The beginning of fuel cycle, nominal power (licensee calculation)	The end of fuel cycle, nominal power
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Full	10,82	12,74	13,0
Without most effective CR	8,48	10,04	10,92

HAEA:

- 1. Requirement 3a.4.1.0500. The nuclear reactor and the active core shall be so designed that in the case of events resulting in DBC1-4 and DEC1, the mechanical failures of the systems, structures and components of the nuclear power plant unit and the physical behaviour of the coolant of the nuclear reactor shall not hinder the shutdown, the maintenance of the subcritical condition and the cooling of the nuclear reactor.*

Guideline N3a.32.: In order to keep the reactor safe and to keep it in a subcritical state, the following bounding parameters are introduced, and the active zone is designed to comply with them:

- closure reactivity,
- the maximum value of the re-criticality temperature,
- the minimum value of the effectiveness of emergency protection,
- subcriticality during refuelling and subsequent tests,
- shut down boric acid concentration as a function of time after shutdown and heat transfer medium temperature.

For the first four above, the most valuable absorbent is assumed to fail. The definition of the closure reactivity bounding parameter should take into account the temperature drops immediately after shutdown. The determination of the shutdown boric acid concentration should take into account the subsequent processes, the further decrease in temperature, and decrease in the xenon concentration after a while.

- 2. Requirement 3a.4.1.0500 3a.4.1.0800. In the shutdown state and during the refuelling of the nuclear reactor, it shall be ensured that the subcriticality continuously remains at the prescribed level even during the placement of fission materials or absorbents in the active core or their removal from there.*

Guideline N3a.32.: The reactor physics calculation for the calculation of subcriticality gives an indication of the effect of placement of new fissile material to the core and the removal of absorbent from it.

NNSA:

China's VVER 1200 project applicant points out that there was a drop height of over 9m during the transfer process of the spent fuel

transportation container. When it was lifted at the gantry crane, the maximum height was 26m. In the design, in order to reduce the load borne when falling from 26m to the allowable value, the damping device was designed on the ground below gantry crane to form a buffering effect. In the meantime, an analysis on the drop of the spent fuel transportation container was conducted and *Report on Analysis of Drop of Spent Fuel Transportation Container of TNPS* was prepared. The conclusion of the report is as follows:

The stress analysis results of transportation container falling from 26m to the buffer layer under 4 conditions are all lower than the stress of falling from 9m to hard ground.

During the process of falling from 26m height to buffer layer, transportation container can maintain its structural integrity without damage.

The report mentioned above is being improved, and the issue is currently under review as a remaining issue during the PSAR stage.

NDK Question 2	The number of pumps in SG Emergency Cooldown System (JNB10-40) is reduced compared to the reference unit													
Description	<p>It has been observed that the number of pumps in the SG ECS system has been reduced from 2 to 1 for each safety train compared to the reference plant. The reason for this change in the design of the NPP is roughly explained as “Due to increased capacity of PHRS(JNB50-80) heat exchangers, pump redundancy of SG ECS can be eliminated.”</p> <p>Increasing the capacity of a passive safety system (PHRS) is a sufficient reason to eliminate the redundancy of any components in the active safety system or not? It is important to evaluate this issue from the point of view of implementing the principle of redundancy in design.</p> <table border="1" data-bbox="459 840 1465 1594"> <thead> <tr> <th data-bbox="464 846 694 1265" rowspan="2">Brief description regarding above mentioned safety systems. Safety functions</th> <th colspan="2" data-bbox="699 846 1177 1003">Safety systems and additional engineered safety features</th> <th data-bbox="1182 846 1460 1265" rowspan="2">Brief description</th> </tr> <tr> <th data-bbox="699 1010 949 1265">Active</th> <th data-bbox="954 1010 1177 1265">Passive</th> </tr> </thead> <tbody> <tr> <td data-bbox="464 1272 694 1588">Heat removal from the reactor through secondary circuit</td> <td data-bbox="699 1272 949 1588">JNB10-40 SG emergency cooldown system (ECS) consisting of two trains (2×100%).</td> <td data-bbox="954 1272 1177 1588">JNB50-80 passive heat removal system consisting of four trains (4×33%).</td> <td data-bbox="1182 1272 1460 1588">Residual heat removal and primary circuit cooldown during design and beyond design basis accidents.</td> </tr> </tbody> </table> <p><u>Passive Heat Removal System</u></p> <p>The system functions during design basis accidents and beyond design basis accidents requiring heat removal from the reactor plant. The system performs passive core heat removal from SG secondary circuit to the atmosphere as alternative ultimate heat sink. The system consists of four identical and fully independent trains based on natural air circulation. Each train comprises two heat exchanging modules, pipelines of steam condensing circuit with valves, supply and exhaust air ducts, air flaps and control devices.</p>				Brief description regarding above mentioned safety systems. Safety functions	Safety systems and additional engineered safety features		Brief description	Active	Passive	Heat removal from the reactor through secondary circuit	JNB10-40 SG emergency cooldown system (ECS) consisting of two trains (2×100%).	JNB50-80 passive heat removal system consisting of four trains (4×33%).	Residual heat removal and primary circuit cooldown during design and beyond design basis accidents.
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	Active	Passive												
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	<p>Steam Generator Emergency Cooldown System</p> <p>The emergency SG cooldown system consists of two trains, each of them connected to two steam generators. The system trains are physically separated and fully independent. Each train of the system consists of pipelines, emergency cooldown heat exchanger, two emergency cooldown pumps and condensate return pipelines in two steam generators. Emergency cooldown of each SG is made through a closed circuit.</p>
<p>Requirement</p>	
<p>Comments</p>	<p>SEC NRS:</p> <p>The reduction of number of in the SG ECS from two to one can be accepted if new PHRS performance justified by experiments and safety demonstration of new design for relevant accident scenario is provided. Redundancy principle should be additionally justified based on safety analysis and description of technical means applied for maintenance and service of SG ECS trains.</p> <p>HAEA:</p> <p>According to PSAR Chapter 6.1 the Emergency Feedwater System (LAR/LAS) consists of four identical completely independent from each other trains (LAR10, LAR20, LAR30, LAR40) with capacity 100 % each. (Each train provides emergency feed pump LAS10(20,30,40) AP001.)</p> <p>At selection of train capacity, it was assumed that one train can be in repair, the second train has independent from the initial event failure, the third train - independent failure.</p> <p><i>Requirement 3a.4.3.0500. Under DBC2-4, the removal of the residual heat from the reactor, the spent fuel pool and the containment shall be ensured by means of one or more redundant heat removal systems in such a way that jointly they are capable of performing heat removal even if one of the systems or a redundant branch of a system is lost due to a failure and, simultaneously, another system or redundant branch is inoperable due to maintenance or testing. If the heat removal system or its service system contains passive design solutions, for which an extremely low probability of failure can be demonstrated for the given operating condition, it is sufficient to design the passive system components to be single failure tolerant.</i></p> <p>NNSA:</p> <p>In China, the VVER-1200 unit is the VVER-1200/491 type reactor device, which has some differences in design compared to the VVER reactor in Turkey. The VVER1200 unit in China is equipped with the emergency</p>

feedwater system (LAR/LAS) and the steam release valve (BRU-A) to jointly perform heating transfer function on the steam generator side after the accident. The LAR/LAS consists of four trains, each with 100% heating capacity, and each train is equipped with one water supply pump. The LAR/LAS is designed to respond to design basis accidents (DBA). In addition, the VVER-1200 units in China are designed with the passive residual heat removal system (PHRS) for steam generators, consisting of four columns, each with 33% heat capacity. PHRS is designed to cope with design extension condition (DEC).

In China's VVER-1200 units, LAR/LAS responds to DBA, PHRS responds to DEC, LAR/LAS and PHRS respond to different types of accidents, belonging to different defence-in-depth levels. Redundancy requirements cannot be discussed between these two systems.

If the LAR/LAS and PHRS jointly perform SG emergency cooldown function under the DBA, it can be discussed that “Due to increased capacity of PHRS (JNB50-80) heat exchangers, pump redundancy of SG ECS can be eliminated.” It should be noted that when reducing SG ECS pumps, it is necessary to evaluate whether the relevant requirements for the DBA are met, such as single failure criteria, multiplicity (redundancy), etc.

Appendix E – System identifiers description

Country	System identifier	System description
HAEA	JND	High pressure emergency core cooling system, based on centrifugal pumps
HAEA	JNG-1	Low pressure emergency core cooling system
NDK	JNB10-40	Steam generator emergency cooldown system
NDK	JNB50-80	Passive heat removal system
NDK	KBA	Chemical and volume control system