European Stress Tests for Nuclear Power Plants

National Progress Report

FINLAND

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Contents

A. Background information on National Progress Report

A.1 General view on NPPs in Finland

Currently there are two nuclear power plant (NPP) sites in Finland, one on Hästholmen island in Loviisa and the other one on Olkiluoto island in Eurajoki. The Loviisa NPP is owned and operated by Fortum Power and Heat Oy (Fortum) and the Olkiluoto NPP by Teollisuuden voima Oyj (TVO).

Loviisa site has two VVER-440 reactors of Soviet design equipped with specific safety features including leak tight steel shell containment to meet Western nuclear safety requirements. The first unit Loviisa 1 (LO1) achieved its first criticality in January 1977 and it started commercial operation in May 1977, the second unit (LO2) in October 1980 and in January 1981, respectively.

In Eurajoki, there are two BWRs of AB Asea Atom design. The first unit Olkiluoto 1 (OL1) achieved first criticality in July 1978 and started commercial operation in October 1979, and unit 2 (OL2) in October 1979 and in July 1982, respectively. The third unit Olkiluoto 3 (OL3) currently under construction is AREVA's pressurized water reactor, European Pressurized Reactor (EPR). There is also Decision in Principle for construction of the fourth unit (Olkiluoto 4) at site, but the construction license has not yet been applied.

In future, Finland will have a third site for Fennovoima's NPP either in Simo or Pyhäjoki, which are located on the northern part of Finnish coast of Gulf of Bothnia. There is a Decision in Principle for this plant, but no selection of site between those above has been made yet.

The Stress Tests considered here focus on the units in operation or under construction, but not on those that have not yet been licensed, as implied in the specifications on the Stress Tests [1].

The two sites considered here, are dealt with in chapters B and C. This National Progress Report does not include STUK's conclusions on the stress test issues, but they will be included in the final National Report.

A.2 Actions initiated in Finland after the Fukushima Daiichi accident

Radiation and Nuclear Safety Authority (STUK) received a letter from Ministry of Employment and the Economy (MEE) on the 15th of March 2011. The letter gave directions for STUK to carry out a study how Finnish NPPs have prepared against loss of electric power supply and extreme natural phenomena in order to ensure nuclear safety.

STUK sent a letter to Licensees on the 21st of March 2011 with specific directions to give a report to answer MEE's letter due on the 15th of April 2011. In some issues Licensees' reports go beyond the EU Stress Tests, but they support the Stress Test work, as well.

Based on the Licensees' reports, STUK gave a report to the ministry on the 16th of May 2011. The report was followed by letters to Licensees (the letter to TVO is still under preparation) with requirements on further safety assessments as well as requests to improve safety where possible.

Due to this, many parts of the work necessary for European Stress Tests have already been carried out in connection with national studies, but they still have to be reported in English to respond specifically to the stress test issues.

B. Fortum – General overview on topics to be discussed in the final Licensee Report

B.1 General data about site/plant

B.1.1 Brief description of the site characteristics

The Loviisa NPP is on the coast of Gulf of Finland, Baltic Sea, and the sea is the ultimate heat sink for the plant. The sea level changes at the site are moderate, and the seismic activity is very low.

B.1.2 Main characteristics of the units

The Loviisa power plant was the first nuclear power plant in Finland. The power plant has two units: the first started commercial operation in May 1977, and the second one in January 1981. The units are VVER-440/213 type pressurized water reactors.

The engineering, procurement and construction of the Loviisa power plant was a multicultural project. The reactor, turbine, generator and other main components are from the former Soviet Union. Safety systems, control systems and automation systems are of Western origin. The steel containment and its related ice condensers were manufactured using Westinghouse licences.

Originally, the reactors had thermal power of 1375 MW, but in late 1990's they were upgraded to 1500 MW. The net electric power of the units is 488 MW each.

Severe accidents were not considered as a design basis when designing and constructing the plant. An extensive severe accident management (SAM) strategy was developed and systems supporting it were implemented in both of the units between 1986 and 2004. Wide changes and upgrades were carried out to enable the management measures and monitoring the course of a severe accident.

B.1.3 Systems for providing or supporting main safety functions

B.1.3.1 Reactivity control

Control rods and boration systems will be addressed, including power supply, pump and motor cooling, and necessary air-conditioning.

For injection of borated water AC power is needed, and thus in case of SBO boration is not possible without provisional arrangements. The possibility to improve the reliability of boration still needs further considerations, in connection of which the need for further ensuring the primary coolant pump (PCP) seal water will be evaluated.

The fuel in the spent fuel pools does not become critical even in case of presence of pure water. Administrative procedures for achieving necessary subcriticality margins will be summarised.

Control of criticality during severe accidents will be dealt with in Section B.6.2.4.

B.1.3.2 Heat transfer from reactor to the ultimate heat sink

The normal systems to transfer heat from the reactor coolant system (RCS) through steam generators to the condenser will be described, and the possibility to dump steam either into the condenser or into the atmosphere will be introduced. When dumping steam in to the atmosphere, it is possible to use either blow out from the steam header or directly from the steam generator by the pilot operated safety valves. The diverse means of removing heat from the RCS by the steam generators or via feed & bleed will be described. In order to be able to ensure closed circuit for the heat removal, alternative means to transfer heat into the atmosphere has been studied.

As a part of the stress test, information on

- power supply, cooling and equipment room air-conditioning for the heat removal systems;
- physical separation and protection;
- the water resources at site for cooling/heat removal;
- power supply for heat removal chains; and
- cooling and power supply for air-conditioning systems

will be given.

A summary of diversification of cooling function will be introduced.

B.1.3.3 Heat transfer from spent fuel pools to the ultimate heat sink

Information on the normal heat removal systems and make-up water systems will be given. Heat removal by boiling the water in the pools is an ultimate procedure in case the pool cooling is not available. The pool boiling is assessed in Section B.1.3.4 when considering heat removal from the containment. The steam from boiling the water in the spent fuel storages would be released into the atmosphere.

The stress test report will be supplemented with the possibility to use the containment internal spray system or the diesel driven fire water system to add water into the pools.

As a part of the stress test the following aspects will be studied.

- the delays due to heat conduction through structures and steam flow rates from the boiling pools for fuel cooling
- power supply, cooling and air-conditioning for the heat removal systems
- the need for access into the containment or spent fuel storages to operate the systems

B.1.3.4 Heat transfer from the reactor containment to the ultimate heat sink

The information on heat removal from the sump water and from the containment atmosphere will be provided.

Stress test cases include study of a situation, where all AC power sources, including diverse SAM diesels of the Loviisa plant, are assumed to be lost. In early phases of the Loviisa SAM strategy, containment venting was ruled out as an option to remove heat from the containment due to low resistance of the containment structure against sub-pressure in case of enhanced steam condensation. A containment external spray system was applied instead.

As a part of the stress test

- power supply, cooling and air-conditioning of the heat removal systems, and
- physical separation and protection against internal and external events

are dealt with.

B.1.3.5 AC power supply

Information on AC power supply will be included in the final report. Physical separation and protection against internal and external events will be dealt with.

At the Loviisa site, possibility to use transportable AC power sources has not been applied. The need for this kind of equipment will be studied as a part of the stress test.

B.1.3.6 Batteries for DC power supply

Information on DC power supply will be included in the final report. As a part of the stress test, physical separation and protection against internal events will be dealt with.

B.1.4 Significant differences between units

The plant units have the following differences that may affect the accident management.

- There are two emergency feed water pumps in LO1 and four in LO2.
- There is an additional large-scale diesel generator common for both units at the site. The control of this device is only at LO1.
- Two boron mixing tanks for borated water production are located only at LO1.
- The spent fuel storages and their cooling is connected with LO2 systems.
- The normal fire water control systems are located at LO1, and diesel powered fire water pumps at LO2.
- The containment external spray water cooling system pumps are located at the sea water pumping station of LO2.
- Equipment and electrical system room ventilation have differences between units.

Other differences between units are minor.

B.1.5 Scope and main results of Probabilistic Safety Assessments

The scope of level 1 and 2 covers both internal and external events for all operating states of LO1. For external events there are no significant differences between the two units. Dependencies of the systems will be addressed in the final report.

B.2 Earthquakes

Description of the analyses carried out on earthquakes will be provided. The basis for the study is the seismic PSA from 2010. The details, such as adequacy of the design basis and design margins, will be assessed taking into account the frequencies of the earthquakes and the probability of the seismic intensity. During construction of the Loviisa plant, there were no specific requirements for seismic design.

Southern part of Finland belongs to area of low seismicity, and the Loviisa NPP has not been designed to take into account earthquakes. After 2001, the new buildings have been designed and constructed against 0.1 g design earthquakes (according to YVL Guide 2.6). The fraction of seismic risk of the overall core damage frequency (CDF) is small. The seismic PSA from 2010 resulted in introducing some safety improvements to decrease the risk from the current level. The Licensee's view is that protection against seismic events is adequate, and there is no need for major plant changes. Additional studies on seismic robustness of the fuel pools and the fire water systems are conducted, as required by STUK based on the report answering the letter from MEE.

B.3 Flooding

Description on plant protection against flooding, including the consequences of high water level will be provided. Additional information on flooding due to heavy rain and meteotsunamis is addressed. Possible improvements on flooding protection will be evaluated.

B.4 Extreme weather conditions

Description on plant protection against extreme weather conditions will be provided. Possible consequences will be addressed based on PSA studies.

B.5 Loss of electrical power and loss of ultimate heat sink

B.5.1 Nuclear power reactors

B.5.1.1 Loss of electrical power

The Stress Test report will include examination of heat removal systems required in different cases. The situation is close to that considered in case of loss of ultimate heat sink in Section B.5.1.3. Capacities of the batteries and their charging will be explained in Section B.1.3.6.

Time delays will be introduced in sections B.1.3.2 and B.1.3.3.

B.5.1.2 Measures which can be envisaged to increase robustness of the plant in case of loss of electrical power

In case of loss of sea water as the ultimate heat sink the possibility of transferring heat into the atmosphere with a new additional system has been studied, in order to ensure transition to closed decay heat removal. The possibility for ensuring the power supply for this decay heat removal system in case of SBO will be evaluated in the design phase of the system.

Enhancing the reliability for charging of batteries of the diesel motor driven additional emergency feed water pumps supporting the start-up of the system would ensure better operability of the system.

The need for mobile equipment will be assessed, as well, but in other sections of the stress test.

B.5.1.3 Loss of the ultimate heat sink

The protection for ensuring the reliability of the ultimate heat sink will be described, e.g. sea water screens and filters. The recirculation from the surge chamber to the suction will be explained, including the prevention of formation of frazil ice with this manoeuvre. Evaluation of the time for water heat up to a level too high for continuing operation will be given.

B.5.1.3.1 Loss of the primary ultimate heat sink

The possibility for dumping steam into the atmosphere in cases with closed RCS will be described. Water supplies for this operation are large, and thus this process can be continued for a long time. Durations will be given in the report.

If the RCS is open and the reactor shaft can be connected with the fuel pools in the containment, the water can be cooled via intermediate cooling circuit. In case the shaft cannot be connected with the pools, heat removal need to be carried out by boiling the water in the shaft.

It may be possible to reinstall the RPV lid, and start cooling through the steam generators by steam dumping into the atmosphere. Evaluation for duration of the lid reinstallation will be provided.

The possibility to use accumulator water for water supply in cases with open RCS with low water levels in the reactor shaft will be studied. Unless the cooling function cannot be recovered, the only means to remove the heat will be boiling of the water, which will be considered in Section B.5.1.3.3.

B.5.1.3.2 Loss of the primary ultimate heat sink and the alternate heat sink

The Loviisa plant does not have a specific alternate heat sink with closed heat removal circuit. Steam dumping from the secondary circuit in to the atmosphere will be considered as an alternate heat sink. From this point of view situations leading to loss of secondary circuit feed water will be studied.

During shutdown with open RCS, the situation corresponds to the case of loss of the primary ultimate heat sink as described in Section B.5.1.3.1.

B.5.1.3.3 Loss of the primary ultimate heat sink, combined with station black out

The cases correspond to situations considered in previous sections (steam dumping in to the atmosphere and boiling the water in the reactor shaft). The possibilities for water injection will be studied, including possibility to use fire water systems. When boiling the water in the reactor shaft, the containment will be pressurised.

B.5.1.4 Measures which can be envisaged to increase robustness of the plant in case of loss of ultimate heat sink

It will be studied, whether there is need for supplying power to some systems that cannot be supplied from the normal backup diesels. The need for additional power supply may arise from the unavailability of the diesel generators after some time due to loss of sea water cooling.

In case of loss of sea water as the ultimate heat sink the possibility of transferring heat into the atmosphere with a new additional system has been studied, in order to ensure transition to closed decay heat removal.

The use of water in the accumulators during shutdown could be possible by changing Technical Specifications and operating procedures. The usability and occupational safety aspects of this will be studied.

The need for further ensuring the possibilities for boration and PCP seal durability will be evaluated.

B.5.2 Spent fuel storage pools

B.5.2.1 Loss of electrical power

This issue is considered in Section B.1.3.3. In case of loss of electrical power the heat removal needs to be carried out by boiling the water in the pools. The diesel driven fire water pumps can be used to add water into the pools. The use of fixed fire water lines will be further analysed.

The containment pressure control systems will be introduced in Section B.1.3.4, and steam dumping in to the atmosphere in case of heat removal by pool boiling in the spent fuel storages in Section B.1.3.3.

B.5.2.2 Measures which can be envisaged to increase robustness of the plant in case of loss of electrical power

The possibilities for improvements are considered to be ensuring the fuel pool cooling or makeup water. In addition, the flow paths from the spent fuel storages need to be optimized in order to ensure proper routing of steam into the environment.

In order to be able to switch to closed decay heat removal not applying the sea water as an ultimate heat sink, an alternative way of transferring heat into the atmosphere has been studied. The possibility for ensuring the power supply for this decay heat removal system in case of SBO will be evaluated in the design phase of the system.

B.5.2.3 Loss of the ultimate heat sink

This is not considered to differ much from Section B.5.2.1, since the loss of ultimate heat sink results in pool water boiling, as well. The containment pressure control systems will be introduced in Section B.1.3.4, and steam dumping in to the atmosphere in case of heat removal by pool boiling in the spent fuel storages in Section B.1.3.3.

B.5.2.4 Measures which can be envisaged to increase robustness of the plant in case of loss of ultimate heat sink

The improvements introduced in Section B.5.2.2 would help also in case of loss of ultimate heat sink, especially the possibility to transfer heat into the atmosphere independently of the sea water systems.

B.6 Severe accident management

B.6.1 Organisation and arrangements of the licensee to manage accidents

B.6.1.1 Organisation of the licensee to manage the accident

Accident management arrangements, as well as radiation protection and the specific SAM control room implemented at the Loviisa NPP will be described in the final report.

B.6.1.2 Possibility to use existing equipment

Specific severe accident management and monitoring systems have been installed in both of the units LO1 and LO2 to manage severe accident of the reactor. These are introduced in more detail in Section B.6.2. Needed equipment for the fuel pools are introduced in more detail in Section B.6.3.

B.6.1.2.1 Provisions to use mobile devices

Usage of mobile equipment has not been part of the SAM arrangements for the Loviisa NPP, with the exception of backing up the containment external spray system pumps and water supply. As a part of the stress test, the need for mobile equipment to further support SAM functions will be evaluated.

B.6.1.2.2 Provisions for and management of supplies

The information on water and fuel supply is introduced in other sections. Description of the makeup water supply into the containment external spray water tank to cover water losses due to evaporation will be included in the report.

B.6.1.2.3 Management of radioactive releases, provisions to limit them

The containment isolation and ensuring the containment integrity are the main functions in limiting the radioactive releases. This will be covered in Section B.6.2. Other means to limit the releases are

- usage of the containment internal spray system to wash out the radioactive material from the containment atmosphere
- flooding of the reactor to cover the damaged core with water, which decreases the releases from the fuel and supports suspension of the released substances in water
- in-vessel retention by RPV external cooling, which results in keeping major part of the radioactive material in the RCS.

B.6.1.2.4 Communication and information systems

Information will be given in the final report.

B.6.1.3 Evaluation of factors that may impede accident management and respective contingencies

The reporting on these issues is still underway. There is an analysis being made on communication aspects. The impact of destruction of the facilities on site and severe accident situation on the neighbouring unit on accident management have not been analysed thus far. Radiation levels have been analysed on the assumption of only one unit undergoing a severe accident. The cooling circuit for the containment external spray has been designed to support one unit at a time, but the system has the possibility for external water source. Otherwise, the severe accident situation in the both units can be managed. The specific SAM diesels are dimensioned for loads of 2 × 100%, and in case of simultaneous severe accident in both of the units, there is no redundancy for the SAM specific AC power.

The containment structure does not provide efficient protection against radiation to upward direction. Therefore, in case there are large amounts of radioactive material in the upper part of the containment, the radiation levels around the containment may be rather high due to scattered radiation from the air above the containment (skyshine) although the containment walls provide shielding against direct radiation.

B.6.1.4 Measures which can be envisaged to enhance accident management capabilities

Enhancing the power supply to some communication services has been considered beneficial. Minimising the effect of skyshine would improve possibilities to carry out some SAM measures.

The situation of severe accident occurring in both units at the same time will be assessed as a part of the stress test, and conclusions drawn from this will be provided.

B.6.2 Maintaining the containment integrity after occurrence of significant fuel damage in the reactor core

The SAM strategy and the critical safety functions of SAM will be described. The SAM strategy for the Loviisa NPP has four levels:

1. Prevention of core damage
2. Prevention of core melt sequences with imminent threat of a large release (usually sequences with an impaired containment function)
3. Mitigation of severe accident phenomena that could pose a threat to containment integrity
4. Control of releases

Top level critical safety functions of the Loviisa SAM strategy:

- Containment isolation
- RCS depressurisation (Section B.6.2.1)
- In-vessel retention (IVR) of molten core by RPV external cooling (Section B.6.2.5.1)
- Hydrogen management (Section B.6.2.2)
- Containment pressure control (Section B.6.2.3)

In addition, the recovery of functionality of SAM systems in accidents occurring during shutdown states will be introduced.

B.6.2.1 Elimination of fuel damage / meltdown in high pressure

The high-capacity RCS depressurisation valves and their usage will be described.

B.6.2.2 Management of hydrogen risks inside the containment

The hydrogen management strategy for the Loviisa NPP will be described including

- containment atmosphere mixing by forcing open the ice-condenser doors,
- hydrogen removal by passive autocatalytic recombiners (PAR), and
- management of rapid hydrogen production by glow plugs.

The monitoring of recombiner catalyst activity will be explained.

B.6.2.3 Prevention of overpressure of the containment

Explanation of the principle of the containment external spray system (introduced in Section B.1.3.4) will be provided.

B.6.2.4 Prevention of re-criticality

The issue is covered in the SAM Guidelines for the Loviisa NPP. The conclusions from the analyses on injecting water, both borated and unborated, into the damaged core will be introduced.

B.6.2.5 Prevention of basemat melt through

B.6.2.5.1 Potential design arrangements for retention of the corium in the pressure vessel

IVR of molten corium will be explained as a part of the Loviisa SAM strategy.

B.6.2.5.2 Potential arrangements to cool the corium inside the containment after reactor pressure vessel rupture

The SAM Guidelines give a short description on the ex-vessel phenomena. In Loviisa, the containment integrity cannot be guaranteed in case of RPV failure (water in the cavity cannot be excluded, narrow cavity). By applying IVR, these phenomena challenging containment integrity can be excluded.

B.6.2.5.3 Cliff edge effects related to time delay between reactor shutdown and core meltdown

No such cliff edge effects due to time delay. Bypass sequences are introduced.

B.6.2.6 Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity

SAM safety functions are supported by SAM specific AC supply and DC batteries. As a part of the stress test, the possibility to carry out the SAM measures without electricity, and whether there is need for these measures or not, will be evaluated. The need for and supply of compressed air and nitrogen will be considered. Based on preliminary evaluation, the RCS depressurisation cannot be done without electricity. The analysis on containment isolation is still underway.

B.6.2.7 Measuring and control instrumentation needed for protecting containment integrity

The issue is covered in Section B.6.2.6, since the loss of electric power results in loss of instrumentation.

B.6.2.8 Measures which can be envisaged to enhance capability to maintain containment integrity after occurrence of severe fuel damage

The need and possibility for RCS depressurisation without SAM specific electricity or batteries is evaluated.

B.6.3 Accident management measures to restrict the radioactive releases

B.6.3.1 Radioactive releases after loss of containment integrity

This information will be provided in the final report.

B.6.3.2 Accident management after uncovering of the top of fuel in the fuel pool

No previous analyses are available, and the issues below are mainly carried out as a part of the stress test.

B.6.3.2.1 Hydrogen management

The PARs in the containment remove hydrogen, and the capacity of the system is evaluated based on hydrogen production in the reactor in severe accidents occurring during shutdown states. The glow plugs cannot be applied to hydrogen sources from the fuel pool.

There are no hydrogen management systems installed in the spent fuel storages. The accident management approach is to ensure reliable water injection into the pools.

B.6.3.2.2 Providing adequate shielding against radiation

The systems supplying water are introduced in Section B.1.3.3.

B.6.3.2.3 Restricting releases after severe damage of spent fuel in the fuel storage pools

For the releases from the spent fuel pool inside the containment, the same principles as mentioned in Section 0 are applied. However, the concept of IVR by external cooling cannot be applied to the fuel pool.

For the spent fuel storages the possibilities to restrict releases are flooding of the pools to a level above fuel and retention of radioactive material in the auxiliary building of LO2, as far as possible.

B.6.3.2.4 Instrumentation needed to monitor the spent fuel state and to manage the accident

The information given in the SAM Guidelines will be provided and supplemented. The usage of containment annulus radiation level monitoring system will be analysed as a part of diagnostics of the situation.

B.6.3.2.5 Availability and habitability of the control room

Detailed assessment is still underway.

B.6.3.3 Measures which can be envisaged to enhance capability to restrict radioactive releases

The improvements considered include enhancement of fuel cooling or makeup water supply, as well as improvement of instrumentation. In addition, the paths for releasing steam out of the spent fuel storages could be optimised.

Minimising the effect of skyshine would help in carrying out accident management measures.

Additional measurements to monitor the state of the fuel pools in the spent fuel storage could be beneficial, and the details will be assessed in future.

B.7 Preconception on issues which have not been discussed in previous analyses

Seismic robustness of the fuel pools and the fire water systems will be studied in more detail than has been done earlier.

Flooding due to heavy rain and meteotsunamis will be included in further studies.

Within extreme weather conditions the range of minimum and maximum temperatures will be increased to study the effect for operability of the safety systems.

Severe accident management issues have been dealt with from the reactor point of view. Therefore the approach for accidents leading to fuel damage in the spent fuel pools has not been considered in detail, and the focus has been in reliable prevention of this kind of situations.

In severe accidents, the hydrogen management has concentrated in prevention of global hydrogen combustion inside the containment. Hydrogen management outside the containment has not been considered.

Possibility for accident management will be studied in cases with severe accident situation at the both units at the same time.

B.8 Preliminary plans for safety enhancements

Based on evaluations carried out after the Fukushima Daiichi accident, it has been concluded that such threats or deficiencies demanding immediate plant modifications do not exist. Some of the issues requiring further studies are presented in the following.

Possible improvements on flooding protection will be evaluated.

In case of loss of sea water as the ultimate heat sink the possibility of transferring heat into the atmosphere has been studied. One possibility for this might be small-scale cooling towers sized for decay heat removal from both of the reactors and all fuel pools.

Improvements to ensure decay heat removal from the fuel pools inside the containment in case of loss of current systems will be considered. Ensuring decay heat from the spent fuel storages will be studied, as well.

Enhancing the reliability for charging of batteries of the diesel motor driven additional emergency feed water pumps is considered to ensure better operability of the system.

Possibilities to

- implement connections for mobile systems to supply feed water into the steam generators,
- extend the operation time of diesel generators at site by increasing the fuel storage capacity,
- extend the operation time of DC batteries, and
- mount mobile equipment rapidly during accident situations

will be examined.

Some procedural changes to enhance water supply capabilities for decay heat removal during shutdown stated will be considered.

C. TVO – General overview on topics to be discussed in final Licensee Report

Teollisuuden Voima Oyj (TVO) owns and operates two nuclear power plant units, Olkiluoto 1 and Olkiluoto 2 (OL1 and OL2). A third nuclear power plant unit Olkiluoto 3 (OL3) is under construction. Location of the plant site is by the Baltic Sea on Olkiluoto island in Eurajoki municipality.

An interim storage of spent fuel (KPA-storage) is in operation and spent fuel from OL1 and OL2 is stored there after adequate cooling time in reactor hall fuel pools. In future KPA-storage will be used to store spent fuel from OL3 also.

Because of the fact that operating units are significantly different in design compared to the unit under construction, TVO plans to divide the final licensee report into two parts:

- Part I: OL1, OL2 and KPA-storage
- Part II: OL3

Within these parts the proposed guidance by ENSREG [2] on contents and format will be followed.

C.1 General data about the site/plant

C.1.1 Brief description of the site characteristics

Location of the plant site is by the Baltic Sea on Olkiluoto island in Eurajoki municipality. The sea level changes at the site are moderate, and the seismic activity is very low.

C.1.2 Main characteristics of the units

Operating units, OL1 and OL2, are BWRs with net electrical power of 880 MW. The units were supplied by the Swedish company AB Asea Atom (now Westinghouse Electric Sweden AB). First criticalities of OL1 and OL2 were achieved in July 1978 and October 1979, respectively.

KPA-storage is equipped with three storage pools. An expansion with three more pools is being built at the moment. The expansion will be available for operational use in the beginning of 2014.

OL3 unit is under construction. It is a PWR (European Pressurized Reactor, EPR) with electric output of 1600 MW. Fuel loading into the reactor is estimated to happen at the end of 2012. Operating license is required prior to fuel loading.

C.1.3 Systems for providing or supporting main safety functions

Units in operation (OL1 and OL2) and under construction (OL3)

The main safety functions to be described here are the following:

- Reactivity control
- Overpressurisation protection of the reactor pressure vessel and OL3 steam generators
- Emergency core cooling

- Residual heat removal from the reactor and OL3 heat transfer from primary side to secondary side
- Heat transfer from the reactor containment to the ultimate heat sink and OL3 secondary side heat transfer
- Containment isolation
- Heat transfer from spent fuel pools to the ultimate heat sink

The primary systems intended for performing these functions will be described, as well as possible diverse means available. For example, the reactor may be shut down by inserting the control rods, but a system for automatic boron injection is available for the same purpose. The descriptions will be as specified in reference 2.

The systems that provide the AC and DC power supply needed for these functions will also be described. The descriptions will be as specified in reference 2.

The main source of information will be the system descriptions in the Final Safety Analysis Report (FSAR) for OL1 and OL2. OL3 FSAR is under preparation for OL3 operating license application. The system descriptions will be a part of FSAR.

C.1.4 Significant differences between units

The most significant differences between OL1 and OL2 are:
- warm cooling water from the outlet side for recirculation back to the cooling water inlet to avoid frazil ice formation is provided from OL2 for both plant units
- process water systems of OL1 are utilized by KPA-storage

C.1.5 Scope and main results of Probabilistic Safety Assessments

TVO has a comprehensive level 1 and level 2 PSA analyses for OL1 and OL2 internal and external events including harsh weather conditions and earthquakes covering normal operation and outages. For OL3, a comprehensive level 1 and 2 PSA study based on the final design of the plant is under preparation, and it will be part of the application for operating license. Main results from the existing analyses will be presented in the final report.

C.2 Earthquakes

Units in operation (OL1 and OL2)

Earthquake was not part of the original design basis of units OL1 and OL2. Since then plant modifications to fulfil regulatory requirement for PGA-value of 0.1 g have been done.

Earthquakes have been analysed in the OL1/OL2-PSA. The analysis includes the estimation of seismic hazard in Olkiluoto, list of components and equipment needed for safe shutdown, earthquake response of buildings and equipment, plant walk downs, containment analysis and probabilistic model based on the presented data. Main results of these analysis and completed major plant modifications will be presented in the final stress test report according to reference 2.

The latest analysis show that only about 2 % of OL1 and OL2 the core damage frequency is due to earthquakes. Safety margins as specified in reference 2 will be discussed in the final report.

<u>Unit under construction (OL3)</u>

All safety important buildings are designed according to the design earthquake load which is specified hereunder.

The design basis for loads for earthquake is based on requirements in Guide YVL 2.6, because the calculated PGA level in Olkiluoto site is lower than the minimum value required in Guide YVL 2.6. Thus, from the point of view of safety design and dynamic analyses, it is adequate to consider the design basis earthquake only.

The seismic loads for the design basis earthquake are defined as follows:

- The same profile is used for the horizontal and vertical directions.
  - Zero period acceleration
  - Horizontal: 0.1 g
  - Vertical: 2/3 of the horizontal
- The ground response spectrum is given for 5% damping. This spectrum is in accordance with Guide YVL 2.6.

The input motion is defined at the finished grade in the free field, in three orthogonal directions described by response spectra corresponding to statistically independent time histories. The two horizontal components are described by equal response spectra.

The probabilistic seismic hazard assessment at Olkiluoto has been estimated in connection to the seismic PSA of the existing Olkiluoto plant units and it consists of three parts: 1) source effects, 2) path effects, 3) site effects. Because there are no registered strong motion acceleration recordings of earthquakes in Finland, the earthquake recordings from Saguenay and Newcastle regions from Canada and Australia were taken as sources of initial data because of their geological and tectonical similarity to Fennoscandia.

The hazard curve shows the return period (annual frequency of exceeding) as a function of peak acceleration level. The probability of seismic activity exceeding the PGA amplitude level of 0.1 g is very low. The ground response spectra for OL3 with uncertainty bounds are presented as defined in the probabilistic seismic hazard assessment.

At the $1 \cdot 10^{-5}$ annual frequency level (100000 year return period), the median peak ground acceleration level (PGA) is 0.085 g for Olkiluoto site. Because the calculated PGA level is lower than the minimum value required in Guide YVL 2.6, the PGA value of 0.1 g is set for the design basis earthquake.

The main outcomes of the analysis of seismic events are summarized below:

- The seismic hazard analysis demonstrates that the seismic hazard at the Olkiluoto site is very low and that the design basis earthquake characterized by a horizontal PGA of 0.1 g has been chosen with adequate conservatism. An earthquake with a horizontal PGA of 0.1 g is expected to occur with a frequency of less than $1 \cdot 10^{-5}$/y.

- Relevant structures, systems and components to be addressed in the seismic fragility analysis have been selected taking into account past seismic PSA experiences and the plant-specific design. The seismic fragility assessment of the selected structures and components has proven, according to the preliminary results, that the OL3 seismic design is adequate. All determined HCLPF capacities are beyond the EUR requirements (40% beyond the horizontal PGA of the DBE, i.e. HCLPF > 0.14 g).
- The fault tree and event tree model of the plant has been enhanced to account for seismic failure modes leading to initiating events or affecting the mitigation of these events. The quantification with FinPSA shows that the contribution of seismic events risk to the overall core damage frequency is very low. The seismic induced core damage frequency of about $1.3 \cdot 10^{-8}$/y represents about 1% of the total core damage frequency only.

The sensitivity and uncertainty analysis has shown that the main sources of qualitative and quantitative uncertainty, namely

- uncertainties regarding the seismic hazard
- uncertainties regarding seismic fragilities of structures and components
- seismic failure modes neglected in the seismic system analysis
- uncertainties regarding the impact of earthquakes on operator reliability
- assumptions regarding the correlation of seismic failure modes

do not compromise the validity of this outcome. Even if conservative assumptions regarding the mentioned sources of uncertainty are applied the seismic core damage frequency risk is still more than order of magnitude lower than the overall core damage frequency.

C.3 Flooding

Units in operation (OL1 and OL2)

According to design basis a rise of sea water to the level of N60+3.5 m may not endanger safe shutdown of the plant nor the environmental radiation safety. In order to fulfil this demand following design assets are fulfilled:

- Reactor building is watertight and can withstand external water level to the level of N60+3.5 m.
- Integrity of other safety significant structures is secured either by structural robustness or flood gates.
- Systems needed to safe shut down may not be endangered of a flood reaching N60+3.5 m. In practise e.g. the so called H-rooms, in which many pumps of the safety systems are located, are water tight are least to level N60+3.5 m. Also, the waste handling systems have been designed to be water tight or they are located above the level N60+3.5 m.

According to extrapolation of measured sea level the frequency of water level exceeding N60+3.5 m is $1 \cdot 10^{-8}$ per year. According to model scenarios mean seawater level will be decreasing at Rauma region until year 2040. This happens mainly because of land uplift, which is characteristic to the plant site.

Above considerations will be presented in more detail in final report. TVO's view is that external flooding due to seawater level with current design basis is not considered as a threat to plant safety in OL1 and OL2.

Other mechanisms which could result in flooding of reactor building are heavy precipitation and blocking of seawater channel.

Amount of heavy precipitation, which could cause pool formation in the vicinity of reactor building, has been re-estimated in 2010. Some corrective actions to enhance storm draining at outside areas of the plant are in progress.

The result of seawater channel blocking, either intake or output, could eventually lead into seawater flow to the outside areas of the auxiliary building. The enhancements which are being done to avoid pool formation by heavy precipitation, will also lead the excess seawater away from reactor building.

<u>Unit under construction (OL3)</u>

Internal and external flooding routes and consequences of flooding is shown in the OL3-PSAR and related topical reports and will be presented in the FSAR which is under preparation. It is assured that flooding up to N60+3.5 m does not endanger the systems, structures or components required to maintain critical safety functions.

The general boundary conditions are as follows:

> The site level at OL3 is N60+3.2m, but water ingress into the buildings via doors or other penetrations would take place only at water levels 30 cm higher than this. Assumption of water level of N60+3.5m (equivalent to the building level +0.00m) was used, and this was seen very conservative, and therefore consideration of the effect of higher water levels was not seen necessary.
>
> The safety goal is to ensure global stability of the building structures, and no water ingress into the buildings.
>
> Subjects considered in layout and design are increased buoyancy loads and hydrostatic pressure loads on outer structures including penetrations.
>
> Possible water ingress via “open” piping system is avoided.
>
> An additional single failure in isolation means is not assumed because one division in maximum would be flooded as a consequence.
>
> The buoyancy loads and hydrostatic pressure loads on the outer structures including penetrations are considered for the building design for a seawater level of N60+3.50m.
>
> The relevant penetrations below building level +0.00 m are tightened and building joints are tightened by Omega Water Stops. The Omega Water Stops are designed for water height of 4 m or 25 m depending on the location. They prevent flooding from one division to another.

C.4 Extreme weather conditions

Units in operation (OL1 and OL2)

Design basis for extreme weather conditions is shown in the present OL1/OL2-FSAR. These are supported by OL1/OL2-PSA-analysis, where weather conditions have been analysed. About 40 phenomena have been screened and the adequacy of the design basis has been ensured. Following phenomena have been chosen for further study and the results will be discussed in the final report.

- Outside air temperature and humidity
- Seawater temperature
- Precipitation
- Wind
- Phenomena, which could endanger the intake of seawater
- External flooding and high seawater level
- Lightning
- External fire (terrain-, bush-, forest fire)
- Snow and ice

Unit under construction (OL3)

Design basis considering extreme weather conditions is shown in the OL3-PSAR and FSAR under preparation.

The following extreme weather conditions are considered in design:

- External air temperatures and humidity conditions
- Wind and wind-generated missiles
- Cooling water temperatures
- Precipitation and external flooding
- Lightning
- Hazards with potential effects on plant items such as cooling water intakes, air intakes
- Site proximity hazards

The Probabilistic Seismic assessment and the probabilistic assessment of the other External hazards are considered within the OL3 PSA study. These analyses support the evaluation of the adequacy of the design basis of OL3 unit to cope external hazards. The analysis of other External hazards cover wide spectrum of external hazards and their combinations grouped to the following categories:

- Air based external events
- Ground based external events
- Water based external events.

For the assessment of the relevance of potential external events to OL3, a set of screening criteria has been defined in the screening analysis report. The screening analysis is based on a mapping of information on plant characteristics with respect to external events and collection of information on data, methods and experiences concerning the analysis of external events, both plant specific and generic.

A set of screening criteria are defined for single and multiple external events. The criteria are applied in the screening analysis in order to eliminate non-relevant external events from further analysis.

For the Olkiluoto site it is shown through external hazard screening analysis that the hazards specified in this section cover external hazards in the proximity of the site. Table 1 below summarises the external natural hazards treated in screening analysis in the PSA.

Table 1. List of external natural hazards analysed in the PSA for OL3.

Air based		Water based		Ground based	
		Natural external events			
A1.	Strong wind	W1.	Strong water current	G1.	Land rise
A2.	Tornado	W2.	Low sea water level	G2.	Soil frost
A3.	High air temperature	W3.	High sea water level	G3.	Animals
A4.	Low air temperature	W4.	High water temperature	G4.	Volcanic phenomena
A5.	Extreme air pressure	W5.	Low water temperature	G5.	Avalanche
A6.	Extreme rain	W6.	Under-water landslide	G6.	Above-water landslide
A7.	Extreme snow	W7.	Surface ice	G7.	External fire
A8.	Extreme hail	W8.	Frazil ice		
A9.	Mist	W9.	Ice barriers		
A10.	White frost	W10.	Organic material in water		
A11.	Drought	W11.	Corrosion (from salt water)		
A12.	Salt storm				
A13.	Sand storm				
A14.	Lightning				
A15.	Meteorite				

The results of remaining external natural hazards that are not treated in sections above are presented in the PSA document. All these external events have been screened out of from the detailed analysis.

C.5 Loss of electrical power and loss of the ultimate heat sink

Units in operation (OL1 and OL2)

The availability of electrical power and means of dealing with SBO or loss of main heat sink will be described in detail in the final report.

With the present plant configuration, loss of AC power (Station blackout) or long term loss of ultimate heat sink would lead to a severe reactor accident. The progression in time as well as management of such accidents is described in Section C.6.

Plant modifications are being planned in order to provide better possibilities for coping with these events (see Section 0).

As to the spent fuel pools at the plant units and at the KPA-storage, a total loss of pool cooling would jeopardize fuel cooling after several days. However, there are a couple of issues that could endanger successful accident management: the pools have no proper water level measurements, and the addition of make-up water to the pools is based on temporary arrangements. Improvements are being planned to address these issues (see section 0).

Unit under construction (OL3)

The OL3 has two station blackout diesel generators installed in two physically separated divisions.

The task of the station blackout diesel generators is to feed the 690 V busbars of the corresponding divisions in case of station blackout (LOOP and CCF of EDGs). One station blackout diesel generator is sufficient to meet the power demand of the loads required for station blackout operation. The equipment is designed to provide emergency power during and after the postulated accident. To minimize the consequence of external hazards the two station blackout diesel generator sets with their auxiliaries are installed in two geographically separated buildings. Each station blackout diesel generator is functionally independent and physically separated from the other such that the consequences of any single failure in one room will only affect one division.

The SBO-diesel has no automatic start. Starting is possible manually only and has to be decided by the operators according to procedures. The start and operation of the SBO-diesel can be done with and without I&C power supply (in case 2h batteries are empty).

The OL3 autonomy time in case of total loss of AC power is two hours without resulting in fuel cladding overheating and cladding failure. The situation would lead to opening of the pressurizer safety valves and loss of primary coolant, but fuel integrity would be ensured.

Total loss of the ultimate heat sink, i.e. loss of sea water, is taken into account in the OL3 design and all safety functions are ensured in case of loss of the ultimate heat sink. Ambient air is the diverse heat sink for decay heat removal via the secondary side, component cooling and the station black out diesels.

In case of loss of ultimate heat sink, the fuel assemblies in the spent fuel pools in the fuel building are cooled by evaporation and steam is released to the vent stack ensuring the decay heat removal. Similarly in case of unavailability of the fuel pool cooling system the heat removal from the fuel assemblies is provided by evaporation and make-up. It is shown that the capacity of the fire extinguishing water system SG as make up system is sufficient to compensate the evaporated water as well as to raise the water level in the spent fuel pools, if needed. Subcriticality is ensured by the boron steel fuel racks.

A controlled release of the generated steam is provided in order to restrict the effects of harsh ambient conditions. Steam is released to a restricted area inside the fuel building which needs not to be accessed for actions necessary for recovery of the fuel pool cooling system. The design of the system enables a restart of operation at 100°C water temperature.

The radiological limit in case of evaporation cooling is met.

The spent fuel pools are equipped with pool level and temperature measurements in Safety Automation System and also with separate hardwired measurements in Safety Information and Control System.

C.6 Severe accident management

Units in operation (OL1, OL2)

The symptom based Emergency Operation Procedures (EOPs) provide guidance for the prevention and management of accidents as well as for the mitigation of the consequences of accidents. A Safety Parameter Display System is available for supporting the application of the EOPs. The EOPs cover all types of accident scenarios up to severe accidents, that is event sequences with extensive fuel damages and simultaneous threat to containment integrity.

The responsibility for accident management lies with the operating organisation in the short term and (in case of more severe scenarios) with the emergency preparedness organisation in the longer term.

The description of the structure and contents of the procedures and instructions as well as the related organisational matters will be based on the FSAR for OL1 and OL2.

Hardware modifications needed for the management and mitigation of severe accidents (up to core meltdown) were implemented in the plant units in the late 80's in the aftermath of the Chernobyl accident. The main goal for the accident mitigation is preservation of containment integrity and leak tightness, so that if releases from the containment become necessary, they can be performed in a controlled manner and the releases can be filtered. To this end, the following new systems were implemented:

- Protection of the penetrations in the lower drywell against direct contact with the molten corium.
- Containment filtered venting system.
- System for filling the containment with water from an external source (the fire water reservoir).
- A dedicated instrumentation system for monitoring the conditions inside the reactor containment in connection with severe accidents.

Even though the management and mitigation of severe accidents was not included in the original design basis of the unit, several original plant systems also play an important part in the accident management schemes, such as:

- Reactor depressurization system (to prevent pressure vessel melt-through under high pressure)
- Devices for gravity driven flooding of the lower drywell (provision for core melt relocation into the compartment).
- The fire fighting water systems (provides water for filling the containment in order to reach a safe stable state.

The hardware available for the management of severe accidents as well as the related procedures will be described in more detail, as presented in the FSAR.

A description of the expected progress of a severe accident sequence in time will be included, based on the analyses presented in the FSAR.

Unit under construction (OL3)

Severe accident management is taken into account in the EPR and OL3 design from the beginning.

Consequently, the EPR and OL3 design incorporates the following features for core melt mitigation and the prevention of large releases:

- Prevention of high-pressure core melt by high reliability of residual heat removal systems, complemented by dedicated severe accident depressurization valves ;
- Prevention of hydrogen combustion by reducing the hydrogen concentration in the containment at an early stage using catalytic recombiners;
- Limitation of molten core concrete interaction by spreading the corium in a dedicated spreading compartment;
- Control of the containment pressure increase by a dedicated containment heat removal system JMQ which consists of a small-capacity spray system and allows recirculation through the cooling structure of the core catcher;
- A filtered containment venting system JMA30 to finally depressurize the containment at long term by purging the non-condensable gases. The system can also be used for decay heat removal by releasing steam from the containment.
- Collection of all leaks and prevention of any confinement bypass is achieved by a double-wall containment.

By these means, the external source terms are limited in a way that emergency response measures such as relocation or evacuation of the population would be restricted to the immediate vicinity of the plant, and restrictions on the use of foodstuffs would be limited to the first year harvest.

A description of the expected progress of a severe accident sequence in time will be included, based on the analyses presented in the PSAR and FSAR. FSAR is under preparation for the OL3 operating license application.

In case of a severe accident, responsibility for plant control is transferred to the emergency organization management in the technical support center. In the main control room the operators will stop using the ongoing event- or symptom-based emergency operating procedures and receive required operating instructions from the technical support center.

A separate severe accident management guidance document called “Operating Strategies for Severe Accidents (OSSA)” will be provided for the emergency organization management team to help assess the accident conditions and determine what coping strategies need to be implemented. Such strategies will be implemented by the operators in the main control room either using appropriate existing procedures (or parts thereof), probably from the set of symptom-based emergency operating procedures, or by “ad hoc operation" without predefined operating procedures according to the instructions of the emergency organization management.

OSSA is under preparation. The plant conditions will be monitored by the severe accident instrumentation system and the radiological conditions with the Central Radiological Computer System.

C.7 Preconception on issues which have not been discussed in previous analyses

As the final conclusions from Fukushima Daiichi are not yet ready some issues might still become relevant. E.g. one issue is hydrogen management during severe accidents. Present analysis of TVO show that hydrogen management is adequate in the containment but depending on Fukushima aftermath some revision of accident management might become relevant. This issue concerns mainly the possible leak of hydrogen into the surrounding buildings and the associated hydrogen combustion phenomena. The risk has been identified in the OL1/OL2 level 2 PSA.

Concerning all units (OL1, OL2, OL3 and KPA-storage) consequences of severe fuel damage occurring in fuel pools has not been analyzed comprehensively in the FSAR or PSA. The present analysis deals mainly to estimate time from the loss of cooling systems to pool water reaching boiling temperature and time until cooling water level reaches top of the fuel elements.

C.8 Preliminary plans for safety enhancements

Based on evaluations carried out after the Fukushima Daiichi accident, it has been concluded that such threats or deficiencies demanding immediate plant modifications do not exist.

However, the analysis done has shown some areas which could be further analyzed. Also, some modifications which further improve safe operation of the plant have been envisaged.

<u>Units in operation (OL1 and OL2)</u>

Earthquakes were not included in the original plant design basis. Presently the plant fulfills design basis of 0.1 g PGA. Considering the margins plant walk down during the spring 2011 revealed some possibilities for improvements. These include adding support for some batteries and strengthening some fire fighting equipment.

The climate change calls for reanalysis of harsh weather conditions and the design basis of the plant. Cold or warm weather exceeding current design basis calls for a study on their effects. E.g. new automation equipment may demand enhanced or more reliable HVAC systems.

Safety significant instrumentation and control systems as well as valve operations have a battery back-up power source, and there is mobile equipment at the plant site, which can be used for re-charging the batteries. However, if a long-lasting accident at several units is confronted more devices might be needed. TVO is currently doing an inventory on possible electrical consumers and the needs of more mobile electric generators. This study will be accompanied with an estimation of the need for fuel reservoirs of the possible new generator equipment.

Suitability of plant modifications are investigated in order to allow direct injection of fire fighting water into the reactor pressure vessel using the diesel motor driven pumps of the fire fighting water system. Due to the relatively low head of these pumps, a booster pump powered by a mobile diesel generator may also be needed. This modification would improve the possibilities for coping with station blackout and loss-of-main-heat-sink scenarios.

Possibilities to decrease the level of dependability on seawater will be studied. E.g. modifications of auxiliary feed water system have been considered.

The spent fuel pools at the units as well as those at the KPA-storage will be equipped with water level measurements and permanent systems for make-up water delivery, which can be manually operated from a safe location.

Evaporation of the pool water is acceptable from plant safety as well as from environmental safety point of view.

<u>Unit under construction (OL3)</u>

No design deficiencies have been identified regarding provisions against natural hazards and disturbances in power supply.

However, the following items are under investigation

- water supply to the steam generator secondary side by independent equipment
- water supply to the primary system by independent equipment during refuelling outage when the primary system is open and heat transfer via the secondary side is not possible
- water supply to the spent fuel pools by independent equipment
- the leak tightness of the nuclear island outside doors and walls and the internal penetrations at sea water level exceeding N60+3.5m.

D. References

1 ENSREG – EU "Stress tests" specifications, Annex I of the letter from G. Oettinger, Brussels 24 May 2011.

2 WENRA Chairman, Laaksonen, Jukka. Post-Fukushima "stress tests" of European nuclear power plants – Contents and format of complementary safety assessment report. Draft, 17 June 2011.