7.2.2001

Unofficial translation by TVO

PRELIMINARY SAFETY ASSESSMENT OF THE NEW NUCLEAR POWER PLANT PROJECT

1 FOREWORD

Teollisuuden Voima Oy (TVO) has submitted an application for a decision in principle to the Council of State concerning the construction of a new nuclear power plant as prescribed in the Nuclear Energy Act (YEL) Section 11. The subject of the application is a nuclear power plant unit equipped with a light water reactor with a maximal thermal power of 4300 MW and net electricity output from 1000 to 1600 MW. The subject of the application also includes the on-site nuclear facilities for the storage of fresh nuclear fuel, interim storage of spent nuclear fuel and handling, storage and final disposal of low and intermediate level operational wastes.

As stated in the application the new power plant unit is to be located either at the power plant site in Loviisa or at Olkiluoto. TVO has made feasibility studies with plant suppliers on several light water reactor types, including both pressurized and boiling water reactors. According to the application, plant options other than those having been subject to the feasibility studies can also be considered.

When preparing the feasibility study TVO had discussions with the plant suppliers taking part in the study and with the Radiation and Nuclear Safety Authority (STUK) concerning the possibilities of the plant types available on the market to meet the safety requirements applied in Finland. The discussions have been preliminary, as prescribed in the YEL Section 55, and they have mostly taken place prior to submitting the application for a decision in principle to the Council of State. The plant suppliers have presented in general terms their plant concepts and emphasized especially their safety design, using the nuclear safety regulations valid in Finland (VNP 395/1991 and the YVL guides) as the standard for comparisons. STUK, for its part, has detailed some generally formulated requirements of the YVL guides.

This preliminary safety assessment is, as prescribed in the YEL Section 12, an assessment on the possibilities of the new nuclear power plant unit to meet the nuclear safety requirements in effect in Finland, and it has been prepared at the request of the Ministry of Trade and Industry (request of statement 4/330/2000, dated 27.11.2000). The preliminary safety assessment’s 2nd chapter presents the safety requirements applicable to the new plant especially for the parts the requirement level of which has essentially changed after the construction of the existing plants. Chapter 3 deals with the plant types, subject to the feasibility studies, and their possibilities of meeting the Finnish safety requirements. The plant types´ most essential safety features have been described by plant type in the Appendix of the preliminary safety assessment. Chapter 4 presents the safety issues independent of the plant type.

When preparing the preliminary safety assessment STUK has familiarized itself with the following appendices to TVO's application as prescribed in the Nuclear Energy Decree:

- study on the expertise available to the applicant (Appendix 3)

- general description on the technical operating principles of the intended nuclear facility (Appendix 7)

- study on the safety principles to be observed (Appendix 8)

- general study on the applicant’s plans and methods available for arranging nuclear waste management (Appendix 14).

STUK's opinion on the technical acceptability of the plant options is presented in general principal terms in this preliminary safety assessment. The acceptability of individual design solutions will be assessed in relation with the construction license process of the plant project, if it becomes actual.

2 REQUIREMENTS SET FOR THE NEW PLANT

The safety requirements for the new nuclear power plant are presented in the decision of the Council of State (VNP) 395/1991 as well as in the YVL guides published by STUK. The YVL guides especially concern new nuclear power plants; they are also applied to the existing operating plants, each time through a separate decision made on the execution/implementation when revising the guides. In addition to the requirements concerning safety design, the YVL guides present the guidelines to be followed when procuring e.g. the plant equipment. According to the basic principles of the YVL guides an alternative procedure presented by the license holder can be approved to replace the procedure of the guide, if the license holder proves that the safety level referred to in the guides will be met.

The requirements are based on the experience obtained from the existing plants, the results of safety studies as well as the general objective to prevent hazardous effects on people, property or the environment possibly resulting from the use of nuclear energy. The YVL guides and their interpretations have been developed according to the principle of continuous improvement of nuclear safety, which is presented in the VNP 395/1991 Section 27.

The requirements of the new nuclear power plant differ from those followed in the design of the existing plants. They take into account in a more systematic manner the possibilities of eliminating factors endangering safety based on the awareness of new knowledge obtained during 30 years. The most essential additional knowledge concerns the possibilities to prevent emissions of radioactive substances to the environment, even if the reactor itself would be seriously damaged. In this respect, the safety requirements of the new plant are clearly more stringent than those applied in the construction of the existing plants. Safety improving modifications designed to reach the safety level requirements of a new plant have been carried out in the existing Finnish nuclear power plants during their operating lifetime.

STUK has defined the safety requirements of the new nuclear power plant with the aim that no major modifications during its operating lifetime would need to be carried out in a plant constructed according to the requirements. The international development of the nuclear safety regulations suggests that the requirement level set by STUK endures comparison with other countries even in the long term. A continuous and consequent tightening of the requirements is not the general international practice. Therefore the safety requirements followed in many countries and recommendations outlined in international cooperation as grounds for national regulations are still based on the technical solutions established during the 1970´s.

When setting for the safety requirements, the intention has been to take the planned operational lifetime of a new plant, in principle 60 years, into account. The long operating lifetime requires preparedness to technical renovations/improvements and changes in key social community infrastructures. Preparations to meet changes shall be made, even if specified preparedness requirements cannot be presented. Changes at the international level comprise e.g. commercial reorganization of the plant, equipment and fuel suppliers. In Finland, the changes relate to training and research arrangements in the field as well as other societal aspects, which might affect the long term (tens of years) sustainability of know-how in the field. The social stability and alterations in existing values can change the technical and other services available on the market, which then might affect nuclear power operations.

Approximately some 30 light water reactors are at the moment under construction in the world. During the past years the construction of nuclear power plants has centralized to the Asian countries (Japan, Korea, China). In East Europe some temporarily suspended nuclear power plant projects have now been completed. During the past years, new nuclear power plants have not been constructed in the USA or Western Europe, except in France. The latest new plant in France was taken into use in 1999, after which no new or ongoing projects remain there.

During the past years increasing investments have been directed to initiate international projects to study new nuclear reactor concepts. A technological breakthrough, in which the focus of electricity production carried out by nuclear energy moves towards technologies essentially different from light water reactors, may occur during the operating life time of the possible new nuclear power plant. In this kind of a situation, the importance of maintaining sufficient domestic know-how of light water reactor technology will further be emphasized.

This chapter presents in general terms the safety design of nuclear power plants, management of severe accidents, and requirements concerning the use of probabilistic safety analyses for the parts where the requirement level has distinctively changed after the construction of the existing plants. The requirement level of the Finnish safety design is for some parts clearly more stringent than the so-called international level. The international level is not exactly defined, but the minimum requirement level is presented in a relatively general form in the regulations and recommendations of the International Atomic Energy Agency (IAEA). On the other hand some requirements, which are in effect in Finland, are less demanding than those generally used in Western Europe. These are presented as well.

2.1 Safety design

The design of a nuclear power plant to be safe is fundamentally a technical design undertaking. On one hand, all technical design is based on the fact that the causal (deterministic) relation between the cause and consequence is understood for practical purposes in a sufficiently accurate way and, on the other hand, that all available knowledge is limited and incomplete. The limited scope of knowledge becomes apparent when technical equipment malfunctions or is damaged in unexpected ways or when plant operators make unexpected errors. This will be taken into account in safety design of the nuclear power plant by the application of the defence-in-depth principle.

According to the defence-in-depth principle the release of radioactive substances hazardous to people, the environment and property is prevented by multiple independent barriers. Barriers are as independent of each other as possible, so that the failure of one barrier would not endanger the others. This provides security against the knowledge-related uncertainty and other imperfections related to the design and implementation of the barriers.

The barriers are dimensioned to maintain their integrity with the best possible certainty even if the worst imaginable barrier-specific threat were directed against them. If necessary, the threat is limited by design solutions and safety systems affecting the behaviour of the plant.

Threats to the barriers´ integrity are mainly limited by designing the nuclear reactor and the other main processes and systems self-regulating, slowly reacting to disturbances and by dimensioning safety margins related to physical phenomena large. Factors restricting the size of safety margins are technical-economic viewpoints and possible mutually contradicting different safety objectives - for example, the reactor core emergency cooling must not jeopardize the integrity of the reactor pressure vessel.

In addition to the safety margins, the barrier integrity is ensured by different protection and safety systems designed to limit the possibilities of transients to develop more serious and to mitigate the consequence of events. These systems are designed to perform their functions irrespective of various presumed failures or failure combinations. Thus the safety of the plant is maintained also in the case of failures and failure combinations, which in practice have never occurred at nuclear power plants. The key objective in managing transients and accidents is to uphold the integrity of the first barrier, the reactor fuel cladding, sufficiently well.

According to the defence-in-depth principle, also failures in mitigating transients and accidents are taken into account. Such failures result in large-scale loss of the reactor fuel cladding integrity. For this kind of an event, a severe reactor accident, the nuclear power plant will be equipped with a containment building dimensioned to withstand the loads resulting from the accident and to keep the hazardous radioactive substances inside the building.

2.2 System design

The possibility of equipment failures will be taken into account in the design of safety systems. As concerns the failure, it is required in Finland that the most crucial safety systems must be able to carry out their functions even if any single device of the system were out of order, and any other device of the system were simultaneously not in use due to maintenance or repair. This so-called N+2 failure criterion affects the structure of some systems implementing safety functions: the complete system comprises of several almost identical separate subsystems, i.e. redundancies. It is further required that for the so-called common cause failures, i.e. similar equipment gets failed from same reasons, the safety functions are secured by systems and/or devices functioning on different principles (diversity). For external threats (such as fire) systems ensuring each other and their redundant parts are physically separated from each other. Balanced operation between different systems is to be ensured with probabilistic methods, described below in more detail.

Systems taking part in the implementation of safety operations are classified, based on their safety significance, in safety classes 1, 2, 3 and 4 in declining order significance according to the Guide YVL 2.1. If the system has no nuclear safety significance, it is classified in the category EYT.

When dimensioning the safety functions, a larger group of events than that used for the original design of the existing power plants has to be taken into account as possible initiating events or other dimensioning factors. The requirements concerning this are presented in the Guide YVL 2.2. Some examples are presented below:

- operational transients during which the reactor shutdown with the control rods is assumed to fail completely (so-called ATWS) have to be included among the initiating events to be considered as postulated accidents

- a possible leak from the pressurized water reactor’s primary coolant circuit to the secondary coolant circuit must not lead to the coolant discharge into atmosphere

- also indirect threats are to be recognized in the initial events analyses, and precautions against them are to be taken into account when designing the systems and devices. An apparent direct threat to the barrier integrity is always connected to every initial event, for example, loss of coolant resulting from a pipe break always disturbs directly cooling of the fuel cladding, thus threatening the integrity. In addition to direct threats, indirect threats may be connected to initial events: for example, in connection with loss of coolant, materials damaged by the pipe break may clog up filter structures of the emergency cooling system cooling the fuel and thus disturb the emergency cooling. As another example, a natural process can be related to loss of coolant in pressurized water reactors, during which the boron dissolved in the coolant for reactor power control purposes concentrates to the reactor core while clean water pockets are formed elsewhere in the primary circuit. The clean water getting into the reactor core at a later stage can cause reactor re-criticality, which would be unsafe during the accident.

The acceptability concerning the design of safety functions, the reactor and the system dimensioning is proved with deterministic safety analyses. The so-called conservative (including disadvantageous assumptions in view of end result) and the so-called best-estimate computer programs can be used as analytical methods. Irrespective of the method, the safety analyses have to take always into account the uncertainties related to the analyses e.g. by carrying out a sufficient number of sensitivity studies. A sufficient safety margin must remain between the result of the analysis and the acceptance criterion to cover uncertainties. The acceptance criteria to be used in the analyses of every initiating event are determined in the Guide YVL 6.2 for the reactor fuel, in the Guide YVL 2.4 for overpressure protection and in the Guide YVL 2.2 for other parts. In addition, the mutual independence of barriers partly covers the uncertainties resulting from limited scope of knowledge and incompleteness of analytical methods.

Some countries apply the so-called leak-before-break principle (LBB) to the dimensioning of the emergency cooling systems. The idea of the principle is to ensure the integrity of the primary circuit pipelines by accurate and carefully controlled manufacture, inspections during operations and continuous monitoring of leakages. The objective is to discover possible leaks in the primary circuit at their initial stages and thus avoid the possibility of a large break. Based on this principle some countries have considered acceptable to simplify the plant structure e.g. by removing the break supports of main circulating pipelines, which improves the possibilities to inspect the pipelines during operation. The application of the leak-before-break principle to the Finnish nuclear power plants will be dealt with in the Guide YVL 3.5 to be published in the near future. STUK applies the principle in the following way: if its preconditions are fulfilled, the pipelines can be constructed without pipe whip restraints, however, without modifying the dimensioning of the emergency cooling system.

2.3 Passive safety systems

The passive safety systems refer to systems implementing safety functions without external source of power. The requirements defined in the YVL guides, originally developed mainly for active systems, are applied for the passive safety systems as they concern the safety objectives of systems and demonstration of their reliability. Passive systems must be based on experimentally well-founded proof of functionality and reliability especially where there is no comprehensive experimental knowledge from earlier corresponding technical solutions.

If the safety function of a passive system is ensured (applying the diversity principle) by an active system, which in the first place is designed as a system of normal operation, the active system in question shall be safety classified.

2.4 Management of severe reactor accidents

The design of the new nuclear power plant takes into account the possibility of extensive reactor core degradation, the so-called severe reactor accident. The requirement primarily concerns the design of the containment building, because a severe accident in itself signifies loosing of the integrity of the innermost barriers (fuel cladding, primary circuit).

Successful management of a severe reactor accident requires a strategy, which systematically takes into account the plant characteristics and the phenomena threatening the containment. The strategy must include well-founded methods to prevent or mitigate the energetic phenomena related to development of the accident (e.g. hydrogen burn, high pressure core melt discharge, energetic core melt coolant interaction). It has to guarantee also the coolability of core melt and the decay heat removal of pressure suppression containment in such a way that the containment remains leak tight during the accident and long after it.

In case of a severe reactor accident the designed systems shall perform their functions even if any single device of the system were out of order (N+1 failure criterion). The systems to be designed to mitigate the severe reactor accidents are to be independent of other safety systems.

The severe reactor accident shall be mitigated in all nuclear power plant operational states, i.e. not only during the power operation but also during the shutdown periods.

2.5 Use of probabilistic safety analysis

The probabilistic safety analysis as prescribed in the Guide YVL 2.8 has to be used as a tool in the nuclear power plant safety design. Its main purpose is to recognize the factors reducing the reliability of designed systems and thus ensure the reliability technical balance of the system design. The probabilistic safety analysis methods are also used for the classification of initiating events based on their estimated frequency.

The active safety systems are generally quite complicated and often the successful functioning of the system depends on many auxiliary or supporting systems. The reliability of various parts of the whole must be in correct proportion to the importance of that part in ensuring the success of the entire safety function. The passive safety systems do not as a main rule need auxiliary or supporting systems to function.

The basis of the probabilistic safety analysis is the physical analyses and studies concerning the plant behaviour with which the ability of systems to perform safety functions in different transient and accident situations are studied. The logic models concerning the implementation of safety functions and the reliability of systems are drawn up based on the physical analyses. The frequency of an (undesired) final event, for example a core meltdown, can be calculated with these logic models. Estimated frequencies of various initiating events and reliability data of components and the operator actions are needed for the calculations. The reliability data are mainly acquired from the existing plants´ operating experience. The calculation also gives some understanding of probabilities of chain events and failure combinations leading to end state. The worst failure combinations, which do not yet prevent the implementation of the safety function, are especially charted. Some various risk values can be further calculated on the basis of the results. The Guide YVL 2.8 presents numerical design objectives related to risk values.

The accuracy requirements of the physical analyses forming the basis of probabilistic studies vary from relatively coarse to very detailed. In addition to the physical analyses all initiating events and sequences have to be identified as accurately as possible in order to make the analysis comprehensive. The uncertainties included in the final results can be quantitatively estimated only for the uncertainties of the reliability data (known) used in the analyses. Other sources of uncertainty are the structure of models, some factors difficult to estimate, such as the frequencies of failure combinations or the reliability of operators´ function as well as the choices made at the beginning of the study concerning the scope of analysis, for example, to what extent the external events (flood, fire, meteorological conditions, seismic phenomena) are dealt with in addition to the internal initiating events of the plant. Due to these reasons substantial differences occur now and then in the calculated risks between identical plants. According to the Finnish requirements the analyses shall cover all the phenomena mentioned above.

2.6 Use of programmable automation technology

The nuclear power plants are equipped with control and protection systems. The automation technology used for safety critical purposes is in transition where the previous analog (wired) technology is being replaced by a totally different programmable digital technology. The requirement level of ensuring safety of programmable technology is being harmonized in Europe. This is also reflected by the fact that there are several suppliers on the market prepared to supply the safety critical systems with programmable technology. In the present competitive situation the expertise and ability of the client to define appropriate requirements for deliveries are the key prerequisites for a good final result.

Extensive and long-term experience of programmable automation is available from the process industry and other applications. Irrespective of positive experience, the use of programmable systems for applications important to safety is challenging, because the flawlessness of software included in them cannot even in theory be comprehensively proved by testing or other quantitative methods. The reliability of the software can however be affected through the production process of the software, e.g. thorough design, implementation and documentation. Therefore the software can be assessed from two points of view: the software production process and the competence of personnel taking part in it, and the software itself which is assessed with the help of tests, analyses and experience gained from their use.

The general principle of nuclear safety applications is that the nuclear power plants´ automation systems as a whole have to be designed so that the fulfilment of their reliability requirements can be proved. The process technology and automation design of the nuclear power plant are tightly connected with each other.

The most advantageous solution to meet the high safety level required for the nuclear power plant is obtained when all physical processes of the plant are smooth and self-regulating. In a well-designed plant the operating automation assisting the normal operation is entirely separated from the limit and protection functions, which simplifies the design concerning both operating automation and limit and protection systems. As the operating automation has no safety function, it can be designed and approved of by lighter procedures to function as widely as possible as a multi-sided operating support. The protection system can be designed to be as simple as possible, which makes the high reliability required of it more readily provable. For this reason it is required that the protection system structure also includes a sufficient amount of diversity. In order to reduce the need for actuation of the protection system function during transients, a separate limit function, which is more reliable than the operating automation, has to be designed between the operating automation and the actual protection system.

According to Stud’s preliminary opinion, the actuation systems of protection functions parallel to the protection system totally independent (naturally functioning, passive) of the automation technology can in principle be taken into account when determining the reliability level required of the protection automation system. In practice the advantage obtained from using the passive actuation equipment depends on their own reliability, which STUK will examine and assess separately in more detail, if the use of such equipment in Finland will become actual.

STUK applies the European requirements level when assessing the programmable automation important to safety.

2.7 Procurement of mechanical equipment for use in classified safety systems

Systems and components of the nuclear power plant are classified according to the Guide YVL 2.1 in safety classes 1, 2, 3 and 4 and EYT based on their safety significance as mentioned earlier. The safety class 1 comprises e.g. central structures of primary circuit, the safety classes 2 and 3 comprise e.g. the process, electrical and automation technical safety systems as well as their components. The scope of supervision concerning components and equipment is determined on the basis of the safety classification, as specified more precisely in the YVL guides.

According to the practice described in detail in the YVL guides, STUK supervises the manufacture of mechanical components in a relatively detailed way. The supervision comprises of e.g. the assessment of quality management systems of the manufacturers, the inspection of component structural designs prior to manufacturing and inspecting of the completed components. During the supervision the quality control results records from various manufacturing phases are examined. In the most demanding safety classes, the control gets more detailed as the component's safety significance increases and the manufacturing of main components is supervised during several phases. All these measures are focused to assure the component's mechanical quality; the process technical requirements at the component level are not specified in the YVL guides, because they can appropriately be determined only for each application separately.

According to the basic principles of the YVL guides, an alternative method or procedure presented by the license holder can be approved for the procedure presented in the guides, if the license holder proves that the safety requirements level referred to in the guides will be met.

2.8 Location of the plant

When choosing the location of the nuclear power plant the objective is to protect the plant from external threats and to keep the disadvantages and threats caused by the plant to the environment as insignificant as possible. The provisions concerning the restriction of radioactive emissions/releases from the nuclear power plant are presented in the decision made by the Council of State on the general provisions concerning the safety of nuclear power plants (VNP 395/1991, Section 3.). The Guide YVL 1.10 defines the requirements concerning the nuclear power plant site and its near-by vicinity. Matters to be taken into account are the impact on the land use, social and economic impact, traffic arrangements, transmission of electricity to the main grid and viewpoints related to security of basic national supply.

2.9 Physical protection and emergency preparedness arrangements and nuclear liability

The location of the plant also affects the possibilities to implement the physical protection arrangements related to the physical integrity of the plant in an appropriate way. Although the primary responsibility of the plant physical protection systems lies with the license holder, the legislation sets obligations also to the society for this part.

As concerns emergency preparedness, the most reasonable solution would be reached if the plant located in a sparsely populated area and far away from significant residential centres. Measures concerning preparedness to accidents would then be directed to a small group of people. The requirements, which concern the plant site itself, the exclusion zone extending to 5 kilometres and the emergency preparedness zone of 20 kilometres as prescribed by the Ministry of the Interior, are the essential ones in the Guide YVL 1.10. In case of accidental situations the surroundings of the plant, outside the plant site, must be equipped with a radiation measurement system.

The nuclear liability is prescribed in the Nuclear Liability Act. The Nuclear Liability Act takes into account the international treaties concerning Finland, which mainly set the minimum limits to the liabilities for nuclear damage. Raised liabilities can be enacted nationally, as is also done in some countries. In this connection STUK wants to state that the present liabilities for damage in effect in Finland are not sufficient to cover the costs of all imaginable severe reactor accidents. Negotiations to develop the international treaties in question are under way. It is presumable that in the near future the minimum amounts of liabilities for damage will significantly increase. The case is problematic because no upper limit in marks can reasonably be determined for the liabilities for damage.

STUK is not aware of impediments concerning the applicant's capability of fulfilling the obligations set by the existing Nuclear Liability Act for the part of nuclear liability.

2.10 Some requirements deviating from the West European practice

As concerns earthquakes and aircraft crash the safety requirements in effect are in Finland less strict than usual in West Europe. For fuel burnup, the discharge burnup is limited lower in Finland than in West Europe in general.

A small aircraft with a single engine is assumed as an aircraft crash in Finland while, for example, in Germany it is a jet fighter, and a big passenger plane in other countries if the plant is located near airport. This affects significantly the structure of the outer containment building. The difference in requirements results from differences in air traffic frequency and characteristics in various parts of Europe and from the intended location of the new plant in respect with airports and flight routes.

As concerns the earthquakes it is ensured during the construction license procedure that the structures and safety systems of the plant withstand the earthquakes estimated possible in Finland. The plant suppliers have in general carried out the basic design for a market area seismically more active than Finland, therefore no problems are expected to arise for this part.

The discharge burnup of nuclear fuel is limited in Finland to such a value that the fuel behaviour within the allowed burnup range is well predictable not only in normal operations but also in transient and accident situations. The predictions are based on experimental knowledge and calculations; for the present there is rather little experimental knowledge available on high-burnup fuel behaviour in accidents, and the extension of this data basis is a necessary prerequisite for significant increase of fuel burnup.

3 FULFILMENT OF REQUIREMENTS IN DIFFERENT PLANT OPTIONS

In Appendix 8 to the application for a decision in principle TVO presents the summarized opinion on meeting the VNP 395/1991 requirements concerning the new nuclear power plant project. TVO has also delivered more detailed descriptions of every plant option to STUK. This chapter presents at principal level the plant-specific key perceptions from the viewpoint of nuclear safety design. Perceptions jointly related to several or all plant types are presented in Chapter 4.

As concerns the safety design the plant types having been subject to the feasibility study can be divided into two groups: The "innovative" plants including passive safety systems and the evolutionary plants based on active systems. SWR 1000 and EP1000 as well as AP1000 (AP600) include passive safety systems; VVER 91/99, EPR, EABWR and BWR 90+ are based on active systems. The Appendix presents short general plant descriptions and their safety characteristics; the safety technical conclusions are presented further below. The Table below presents the main information of the plant options.

Plant	Supplier	Type, rating[1)]	Characterization
VVER 91/99	Atomstroyexport	PWR, 1000 MWe	Evolutionary
SWR 1000	Siemens[2)]	BWR, 1000 MWe	Innovative
EP1000 / AP1000	Westinghouse[3)]	PWR:s, 1000 MWe / 1000 MWe	Innovative
EPR	Nuclear Power International[2)]	PWR, 1500 MWe	Evolutionary
EABWR	General Electric	BWR, 1400 MWe	Evolutionary
BWR 90+	Westinghouse Atom[3)]	BWR, 1500 MWe	Evolutionary

Remarks to the Table:

1) PWR = pressurized water reactor, BWR = boiling water reactor. The capacities are approximate values of electrical output, e.g. the seawater temperature of the plant site affects the final electrical output.

2) After submitting the application Siemens and Framatome have incorporated their nuclear power business activities to a new joint company, Framatome ANP, in which Framatome has the majority of ownership. The previous joint company Nuclear Power International has been merged to this new company.

3) Westinghouse's nuclear power business activities are nowadays in the ownership of the British Nuclear Fuels Ltd (BNFL). BNFL has also bought the ABB corporation's nuclear power business activities (ABB Atom and ABB Combustion Engineering), and ABB Atom, the designer of BWR 90+, has become a part of Westinghouse.

In its application TVO states that the final choice may also be directed to another light water reactor type other than those presented in the application. STUK is continuously following the development of nuclear power technology and therefore also the plant options available on the market, but does not assess them in this connection. Their prerequisites in principle to meet the Finnish requirements have to be examined separately, if this kind of a choice becomes actual.

The reactor core design of all the scrutinized light water reactors is implemented so that the natural feedbacks, affecting the reactor power, mitigate the changes in power. The fuel and core design affects, in addition to feedbacks, also the reactor stability and the extent of a possible reactivity accident. The fuel life cycle in the reactor comprises some years, therefore the fuel and core design will evolve throughout the lifetime of the plant, and during that time these safety aspects have to be taken care of.

All boiling water reactors being under scrutiny are equipped with a pressure suppression containment, which is opened for the refuelling periods. The implementation of sufficient containment function during shutdowns still requires additional design in case of these options. All pressurized water reactors having been subject to discussions are equipped with the so-called large dry containment in which the containment function can rather easily be ensured also during the refuelling shutdowns.

The probabilistic methods have been utilized in the design of all plant types. Some differences between the suppliers can be found concerning the extent of use of methods and principles of application, but in all cases the probabilistic studies have supported the safety design. As a rule the preliminary risk estimates of the plant suppliers result in essentially smaller probability of severe reactor accidents than that of the existing plants.

The most important conclusions concerning the plant types, more precisely presented in Appendix, are the following:

VVER 91/99

In principle there are no safety technical obstacles, which would hinder the approval of this plant type. In practice several details of the design (e.g. some technical questions related to the decay heat removal from primary circuit, the recirculation arrangements of emergency cooling systems and the severe accident management strategy) require closer examination and continuation of design work. The manufacturing supervision according to the modern quality assurance principles requires special effort from the client. The plant supplier has expressed its readiness for acquiring entire systems or components from the subcontractors chosen by the client itself.

SWR 1000

In principle there are no safety technical obstacles, which would hinder the approval of this plant type in Finland. However, several design details require more detailed examination and continuation of design work, especially relating to the capacity of passive systems and possibly the recirculation arrangements of emergency cooling systems. Additional work is also required for ensuring sufficient containment function during shutdown periods, as with all boiling water reactors. The external cooling of the reactor pressure vessel, which is part of the severe accident management strategy, requires confirmatory research.

EP1000 and AP1000

In principle there are no safety technical obstacles, which would hinder the approval of this plant type in Finland. As concerns the safety characteristics expected from a passive plant type the core design of EP1000 seems to be more successful than that of the AP1000 plant. However, several details of the design require closer examination or continuation of design work, especially as concerns the capacity and the failure tolerance of the passive systems as well as possible shielding of the inner containment against missiles. The severe accident management strategy by cooling the core melt in the reactor pressure vessel may be difficult to prove safe in the case of the AP1000 concept, if the pressure vessel diameter remains the same, as originally planned, as in the AP600 which has less reactor power. The in-vessel melt retention strategy requires confirmatory research for the EP1000 concept as well.

EPR

In principle there are no insurmountable safety technical obstacles, which would hinder the approval of this plant type in Finland. However, several details of the design require additional work, especially concerning the reactor core design, the reactor emergency borating system, the containment liner, the capacity of the reactor emergency cooling systems and recirculation arrangements of emergency cooling systems as well as the severe accident management strategy. The cooling of the core melt related to the mitigation of severe accidents has been designed complicated in this plant type, and therefore its successful functioning is difficult to prove in a dependable way.

EABWR

In principle there are no insurmountable safety technical obstacles, which would hinder the approval of this plant type in Finland. Several details of the design, however, require additional work (e.g. some system design features, such as the capacity and failure tolerance of reactor borating system as well as the recirculation arrangements of emergency cooling systems). Significant amount of additional work is required for taking care of the severe accident management strategy as a whole in an acceptable manner and for ensuring sufficient containment function during shutdown periods.

BWR 90+

In principle there are no safety technical obstacles, which would hinder the approval of this plant type in Finland. Some details of the design, such as the recirculation arrangements of reactor emergency cooling systems, however, require additional design work. Additional work is required for ensuring sufficient containment function during shutdown periods. The core catcher concept, which is designed for cooling the core melt possibly discharging from the reactor during a severe accident, seems to be promising, but confirmatory research to prove its function is still needed.

4 FULLFILMENT OF THE REQUIREMENTS CONCERNING GENERAL QUESTIONS

4.1 Passive safety systems

The common feature of the innovative plant types is that their safety functions are primarily carried out with active systems designed for normal operation. These are not classified as safety systems in all cases. In such cases, when designing the systems the requirements referring to the safety function are, however, partly taken into account. In principle there is no technical reason, which would hinder the application of safety classification as prescribed in the Guide YVL 2.1, but at the feasibility study phase this could not be carried out due to the time schedule aspects for reissuing the guide.

The passive safety systems have been designed to carry out safety functions, if the active system, which is normally used, does not function. Although the safety objectives are the same, the inherent capacity of passive systems is lower in general compared to the active ones. The capacity requirements have in general been considered reliably achievable by using larger physical safety margins of the main processes when designing and dimensioning the plant. According to the experience obtained, large safety margins in the dimensioning of main processes are advantageous for safety in all cases. As concerns capacity of systems, large over dimensioning would be detrimental in the situations with several simultaneous safety objectives and they limit the dimensioning from opposite directions.

Research is still required in order to really prove the innovative technical solutions to be good, which for the traditional active systems has been carried out already for long. On the other hand experimental knowledge on the operation of the active systems of the nuclear power plant in real accident situations is so limited that essential differences in operational reliabilities between different systems can not be drawn on the basis of this; the success in operations of both types of systems has been justified by experimental studies carried out in smaller scale and by scaling the measurement results to the plant scale by using calculational methods.

4.2 The use of programmable automation technology

It is STUK's understanding that programmable automation technology, which meets the applicable European requirement level set for the safety related automation technology, is available on the market. The automation concepts offered by different plant suppliers have not in this connection been assessed to such extent that STUK could take a position on the acceptability of any individual alternative. The same applies also to the emergency control rooms of different plant options.

4.3 Procurement of mechanical equipment

TVO has suggested changes in the existing practice when dealing with mechanical equipment belonging to systems classified under the safety classes 2 and 3 (safety systems and their auxiliary systems). TVO considers this new approach necessary for the possible new plant project as well as for the procurements of equipment for the existing plants. The proposition is based on the opinion that the manufacturing industry has developed high level quality systems due to which the quality of series-produced products is uniformly good. TVO considers that taking into account the special requirements of the nuclear industry could lead to changes in the production process leading to reduced quality.

TVO considers the current supervision procedure concerning the manufacture of mechanical components, described in a relatively detailed way in the YVL guides, to be burdensome and presents procedure in which “the modern industrial quality control” of the component manufactures together with the equipment specific “Component Suitability Report” could replace the current practice.

When acting as TVO proposes, it could be possible to use essentially the same mechanical components in the safety systems as in other demanding industrial applications, since the component specific documents and inspections partly deviating from the “industrial standards” required according to the current practice would no more be needed to the same extent. At least in principle there would then be a lot of knowledge on operating experience from other applications, which can be considered to be positive in case this knowledge of operating experience is relevant from the viewpoint of nuclear power plant operations. Equipment specific suitability study would give the opportunity for a more precise documentation, already at the design phase, of the requirements set for the process; this can be regarded as a reasonable initiative from the viewpoint of safety.

Negative aspects of the proposal include the fact that the equipment specific knowledge of technical structures may partly remain with the manufacturer only, and its availability during decades may not then be guaranteed, and that the assessment of the real significance and/or sufficiency of the operating experiences can be difficult. According to the already available experience, the conventional industry does not have operating experience, which could completely correspond to the requirements of nuclear power plant safety systems, although there is plenty of other experience available.

If the intention is to appeal to the operating experience during the approval procedure concerning mechanical equipment, it must be proved that the operating experience is, in every respect, covering and relevant. In lack of sufficient operating experience the type specific test has to be planned and carried out in order to prove the acceptability.

The YVL guides allow to develop and use new practices provided that the licensee proves that the alternative procedure leads to at least the safety level referred to in the guides.

4.4 Location of the plant

According to the application for a decision in principle intended location of the new nuclear power plant is at the current nuclear power plant site at Hästholmen in Loviisa or at Olkiluoto in Eurajoki. TVO and Fortum Power and Heat Oy (FPH) have agreed that the location planned for the new plant unit at Hästholmen owned by FPH would be in TVO´s use if needed, as respectively the planned location at Olkiluoto owned by TVO. An environmental impact assessment report (EIA) has been drafted and handled for both plant sites already earlier.

The planned location at Hästholmen in Loviisa is situated south of the existing plant units. At Olkiluoto in Eurajoki two location alternatives are still presented at this phase. They have been presented in the EIA report concerning extension of the Olkiluoto nuclear power plant with a third unit. They are located at the plant site at the distance of less than half a kilometre from the existing plant units towards west and north.

STUK has presented the Ministry of Trade and Industry the statements concerning the EIA reports mentioned above stating that both EIA reports are extensive and they deal with the key questions from the viewpoint of environmental impact. In the statement especial attention was drawn to the EIA reports´ radiation and nuclear safety questions which in the environmental impact assessments were not yet studied very much in detail. The Ministry of Trade and Industry has already presented its final statements concerning the EIA reports of both plant sites.

The assessments of the EIA reports earlier presented by STUK did not bring up any matters concerning the environmental radiation safety, which might prevent the construction of the new nuclear power plant at Hästholmen in Loviisa or at Olkiluoto in Eurajoki. However, STUK paid attention to the impact of the cooling water used by the plant on the surrounding sea areas. The impact of the heat load and various alternatives related to water intake and outlet are to be studied in detail, if the project proceeds.

The application for a decision in principle does not present any new information, which would complete the descriptions presented in the EIA reports concerning the environmental radiation safety. In Finland the applicable requirements for the location are stricter than those applied in locating nuclear power plants in several other countries using nuclear power. In Sweden at the Forsmark, Oskarshamn and Ringhals nuclear power plant sites similar basic approach has been followed as in Finland. These sites comprise three or four nuclear power plant units. The releases of radioactive substances resulting from the new nuclear power plant are estimated to remain so small that the plant site-specific requirements applied to the releases will be met with certainty.

STUK has gained experience in supervising the operations of Finland’s existing nuclear power plants during more than two decades as well as significant information on the safety in the vicinity of Hästholmen site in Loviisa and Olkiluoto site in Eurajoki. According to the information available to STUK both plant sites are suitable for the location of the new nuclear power plant, this concerns also the bedrock. Observations presented in connection with the EIA report are taken into account in connection with the possible construction license procedure.

Both plant sites have already a radiation measurement system built in the surroundings as part of the emergency preparedness arrangements to accidents. As concerns the residential density in the vicinity there is no reason, which would hinder the construction of the new nuclear power plant.

4.5 Expertise of the applicant

A decision in principle has been applied by Teollisuuden Voima Oy (TVO) having experience from the construction period of the existing plants in Olkiluoto and after that over twenty years' experience in nuclear power plant operations. As a general rule, the operational experience from the plant has been very good which proves that TVO's organization and the hierarchical management style function well in this task. However, design of modifications in the nuclear power plant or the construction of a new nuclear power plant is a significantly different task than successful plant operations. The successful design of modifications, also concerning the new plant, requires high quality technical expertise in all central fields of technology and management processes, in order to employ the specific expertise correctly. Development of activities is needed also in order to ensure smooth change-of-generation arrangements the nuclear energy field is facing with in the future.

During TVO’s modernization project, TVO’s own expertise or the expertise available was not in all cases utilized sufficiently for the design of modifications of systems and equipment. STUK has noted the matter also in connection with the latest plant operating license renewal procedure.

TVO bought at turnkey basis the Olkiluoto plants from the supplier who, in problematic situations, was ready to offer its clients help even beyond the clients´ requests. The experience related to plant modifications and the modernization project proves that this kind of integrated service is no more available on the market. As a consequence of this, if the new plant will be constructed, already at the construction phase TVO has to familiarize itself with the construction and design basis of the plant thoroughly in a considerably more profound way than during the construction of its existing units. TVO's own familiarity is the prerequisite of successful plant operations and maintenance, which respectively require development of TVO's organization and lines of action.

According to the application TVO has planned to carry out the actual construction project with a separate organization comprising 60 persons, half of the personnel being employed outside TVO. Competent and experienced personnel are still available at present, therefore increasing the organization would not be problematic. Commencing the new nuclear power plant project might also help the change-of-generation arrangements which will take place in the near future with the personnel taking part in the current plant operations, safety research and supervision.

In its application for a decision in principle TVO refers to the use of external expert organizations and consultants as a support of its own organization and activities. Using the outside consultants is an appropriate alternative when due to large amount of work, particular specialties or other corresponding reasons it is not feasible to have adequate expertise or personnel inside the company. This procedure also requires of TVO deep and profound expertise in all fields of technology having effect on the plant safety in order to be able to utilize the outside expertise to carry out right tasks at the right moment. The one who needs outside special services has to attend to maintaining the needed services to a sufficient extent throughout the lifetime of the plant.

4.6 Nuclear fuel and nuclear waste management

Each candidate plant supplier also manufactures nuclear fuel. On the nuclear fuel market it is normal to make the suppliers compete; the choice of the plant type does not in itself limit the possibilities to purchase nuclear fuel. (A fuel rod bundle in cross direction larger than the present one is designed for SWR 1000, all suppliers might not have direct preparedness of its manufacturing.)

The life cycle of nuclear fuel in the reactor is quite short, from 3 to 4 years in Finland, and therefore the fuel technology develops quicker than the nuclear power plant technology. The development of nuclear fuels and taking new design solutions into use mean a continuous challenge throughout the plant's entire lifetime to understand the behaviour of fuel. This also requires maintenance and development of the tools used for the analyses of transients and accidents, because the fuel characteristics and the reactor core design affect essentially the reactor core safety limits and the natural behaviour of the reactor during transients and accidents and thus is crucial to the safety of the plant.

The procedures used for the handling of fresh fuel and the handling of low and intermediate level wastes represent the established technology. The new plant would utilize the infrastructure existing at the current plant sites. There are no technical obstacles to attend safely to the handling of fresh fuel and the handling of both low and intermediate level radioactive wastes. After the decommissioning of the new plant the waste resulting from the decommissioning can be handled as planned concerning the decommissioning waste from the existing plants, i.e. to finally dispose of the radioactive decommissioning wastes with other low and intermediate level radioactive wastes. The design of the new plant takes into account the viewpoints related to appropriate carrying out of decommissioning.

The intention is to carry out the similar procedures for the spent fuel from the new plant as for the existing plants, i.e. to finally dispose of the spent nuclear fuel in the bedrock in such a way that the migration of radioactive substances from the final disposal repository back to biosphere is reliably prevented for a sufficient period of time.

In practice the spent nuclear fuel is kept in the interim storages built on the ground to wait for the final disposal. The storage capacity of spent nuclear fuel from the current power plants is sufficient till the beginning of the 2010´s, by which time preparations to extend the capacity, due to prolonged plant operations, will be made. The need for storage capacity concerning the spent nuclear fuel accumulated from the intended new nuclear power plant can be considered in connection with the future extensions.

For the actual final disposal the spent fuel bundles are packaged in tightly sealed metal canisters, which will be placed in the bedrock at a depth of 400-700 meters in geologically intact rock areas. For this purpose a tunnel network, in which the canisters will be placed, is excavated in the bedrock. The canisters are isolated from the bedrock and the tunnels are filled with a clay compound protecting the waste canisters and acting as additional isolation of the radioactive substances. The technology needed for this kind of final disposal is already far developed. The construction of the final disposal repository for spent nuclear fuel from the existing plants would begin in 2010 according to the current time schedule.

Posiva Oy, jointly owned by Fortum and TVO, will carry out the final disposal. Posiva has applied for a decision in principle of the Council of State concerning the final disposal of the spent nuclear fuel from the existing power plants with a procedure presented above and concerning the construction of the final disposal repository in the Eurajoki municipality. In principle, the intended final disposal repository can be extended to hold also the spent nuclear fuel from the new nuclear power plant, but in this case it has to be ensured that the extension is carried out inside a geologically intact rock area. STUK has assessed the extended final underground repository facility in the preliminary safety assessment given by it, and has not identified any matters which would prevent the extension.

As its conclusion STUK states that such technical factors that would hinder the safe handling of spent nuclear fuel accumulated from the new nuclear power plant unit or the final disposal as planned for the spent fuel from the existing plants have not been disclosed.

5 CONCLUSION

Teollisuuden Voima Oy (TVO) has applied for a decision in principle of the Council of State that the construction of the new nuclear power plant is in line with the overall good of society. In its application TVO presents that the new nuclear power plant would be a light water reactor and presents seven available light water reactor types. Options include pressurized water reactors, the reactor type of Loviisa, and boiling water reactors, the reactor type of Olkiluoto. In this preliminary safety assessment the Radiation and Nuclear Safety Authority (STUK) assesses whether the principal prerequisites of the project presented in TVO's application meet the Finnish safety requirements.

The safety regulations concerning the technical solutions of the power plant and the procedures to be followed during its construction and operations are presented in general terms in the decision of the Council of State (VNP 395/1991) and in more detail in the YVL guides published by the Radiation and Nuclear Safety Authority (STUK). The starting point of the preliminary safety assessment made by the Radiation and Nuclear Safety Authority (STUK) is that meeting of these safety regulations signifies that the Nuclear Energy Act Section 6 ("The use of nuclear energy must be safe; it shall not cause danger to people, or damage to the environment or property.") is met as concerns the technical solutions of the nuclear power plant.

The safety regulations concerning the technical solutions of the new nuclear power plant are in many respects stricter than the regulations applied to the existing nuclear power plants during their construction. Modifications improving the safety have been made to the existing Finnish nuclear power plants, as the development of science and technology as well as the operating experience has given reason to it. This principle of continuous improvement of safety is also included in the nuclear safety provisions (VNP 395/1991) and it will be also followed for the intended new plant. The international development of nuclear safety regulations refers to the fact that the requirement level applied in Finland is stringent enough even in the long run.

TVO presents to build the new nuclear power plant at either site of the existing nuclear power plants. According to the assessment made by STUK there is no obstacle for this from the viewpoint of safety. The releases of radioactivity resulting from the operation of the new plant together with those from the existing power plants at the site remain clearly smaller than the limits set for the entire plant site. However, the viewpoints presented in the assessments given concerning the environmental impact assessment reports must be taken into account in order to ensure sufficient cooling water supply to the power plant.

According to STUK's opinion TVO has reason to develop its organization, lines of action and its own technical design expertise and the design expertise of safety related systems to ensure safe operation of the plant in the situation where integrated safety design services are not available on the market.

There is no obstacle for safe handling of low and intermediate level radioactive wastes accrued from the operation of the new nuclear power plant at the existing plant sites. In this respect the new plant unit can mostly utilize the existing infrastructure at the plant sites. The same applies to the nuclear fuel management and the handling of spent fuel of the new nuclear power plant, which are carried out in the similar way as in the existing plants. It is assessed that it is possible to carry out the extension of the facilities used for the final disposal of wastes so that the extension will not jeopardize the safety of the final disposal.

The preliminary safety assessment of STUK has not brought up matters, which would prove that the plant options, presented in the application for a decision in principle, could not be made to fulfil the Finnish safety regulations. None of the presented options does, however, meet all the requirements as such. The nature and/or extent of the necessary modifications vary considerably by plant types. Relatively small system technical modifications have to be carried out in some plant types, some plant types need more extensive structural modifications. In the application for a decision in principle TVO states that the final choice can be directed to some other light water reactor, other than those presented in the application. The Radiation and Nuclear Safety Authority has followed the development of the nuclear power technology and also familiarized itself with other light water reactor types, but their possibilities to meet the Finnish safety regulations have to be assessed separately, if such a choice becomes actual.

APPENDIX: GENERAL DESCRIPTION OF THE PLANT OPTIONS

Innovative plants

SWR 1000

The reactor in question is a German 1000 MWe boiling water reactor designed by Siemens. (Since the beginning of 2001 the nuclear power plant business activities of Siemens have been integrated into a joint company Framatome ANP, in which the French Framatome is the principal owner). SWR 1000 is a thoroughly renewed design based on German boiling water reactor technology; its safety design is based on natural characteristics and passive safety systems. The containment is a pressure suppression containment, typical of boiling water reactors, but its dimensioning standpoint is to withstand loads caused by severe reactor accidents as well. The planned operational lifetime is 60 years.

The reactor is a boiling water reactor equipped with internal reactor coolant pumps, and its operation parameters correspond to current boiling water reactors. The core design has, however, been totally renewed. The core height is approximately one meter (30%) lower than that of the current reactors with corresponding capacity, and it is situated lower inside the pressure vessel than the core of current boiling water reactors. Its power density is lower than that of the current large-scale boiling water reactors. These features improve the inherent safety compared to current plants in terms of both reactor stability and thermal margins. In addition the core is situated in an advantageous position (low) as concerns transients and accidents. Due to the large volume of the reactor pressure vessel, the pressure regulation transients develop more smoothly than in current boiling water plants while, however, still being fast.

The fuel rod bundles are larger by cross-section than the current ones. Measurement data on the fuel behaviour of a bundle with the designed dimensions are not yet available, but it has been possible to assess the effects of the change by calculations. When using large bundles the amount of control rods can be reduced, this also reduces the number of penetrations through the bottom of the pressure vessel. On the other hand the reactivity values of single control rods can simultaneously increase. If so, the significance of prevention and limitation of transients caused by the failure movements of control rods, including throw outs, will further increase. As concerns safety this kind of an increase in the bundle size is possible within the Finnish discharge burnup limit. According to Siemen's calculations the internal power distribution with the discharge burnup approved in Central Europe would be so uneven that safety margins would be essentially reduced. During the operating cycle the core reactivity is controlled with control rods (driven by electric motor) and solid burnable absorbers in the fuel.

It is probable that the turbine plant will be similar to those in current boiling water reactors, however, with the difference that the protection measures, which are an essential part of reactor safety, starting as a consequence of a disturbance in the turbine plant can be carried out more straightforwardly than in the existing Finnish boiling water reactors.

The safety functions have been implemented with active and passive systems, of which the active systems are primarily meant for normal use. They are, however, safety classified . The control of normal operation, transients and accidents can be fulfilled both with active systems, meant for the normal operation, and the separate passive systems or with the passive systems alone in the following way:

- reactivity: the power level is actively regulated with the control rods and reactor coolant pumps as in the existing boiling water reactors; the hydraulic scram with the control rods and the boron system feeding directly to the reactor core based on pressure accumulators functioning with steam pressure form the passive scram systems; the scram can also be carried out by actively driving the control rods to the core with an electrical motor drive

- cooling and decay heat removal:
  - A 2x100% decay heat removal system based on pumps and heat exchangers, which can cool down both the reactor and the containment, is functioning as the active system.
  - The passive system consists of so-called isolation condensers functioning with natural circulation cooling the reactor (4x50% - the natural circulation condensers´ ability to transfer heat is a function depending strongly on the reactor pressure, decreasing with decreasing pressure).
  - There are six relatively large safety/relief valves for the reactor pressure depressurisation, with two types of different operating principles.
  - In order to control possible pipe breaks of the primary circuit the reactor water supply is backed up with a gravity-driven flooding system (fourfold system) from containment water pools. Part of the reactor pipe connections is also equipped with flow limiters. The long-term decay heat removal after a pipe break would take place by natural circulation between the reactor flooded and the containment.

- the basic type of containment is the even today common pressure suppression containment built of prestressed concrete equipped with a condensation pool, as in the existing boiling water plants, inerted with nitrogen for power operation. Its decay heat removal is carried out either actively with the same active decay heat removal system as the reactor itself or passively with the natural circulation condensers (4x50%), which transfer heat to the water pools of the reactor hall situated above the primary containment.

The safety systems appear to meet the failure and diversity criteria of the YVL guides at least in general; the final certainty of this could be reached in a detailed examination of the system design.

All essential safety functions of the SWR 1000 starting from the reactor water level, start directly either due to the effect of process changes related to the transient or to the control and protection automation with the help of the passive actuation transmitters. The passive actuation transmitters do not require external energy in order to function, not even actuation power as they react directly to the lowered water level which is always a cause necessitating the start of safety functions (such as the reactor scram or emergency cooling).

The mitigation of a severe reactor accident beginning from power operation has been an integral part of the reactor and containment design in this concept.

The severe accident management strategy is based on arresting the core melt in the reactor pressure vessel by cooling it from outside. If this can be shown to work, energetic threats directed to the containment due to the core melt water interactions, such as steam explosion, or core melt concrete interaction would not occur. According to the preliminary assessment made by STUK, the successful external cooling can be proved for this plant type, but the verification of sufficient margins for this part still requires more work.

During operation there is a nitrogen atmosphere inside the containment, therefore hydrogen burns inside the building during a severe reactor accident are excluded. The pressure capability of the containment is dimensioned to withhold the hydrogen development resulting from a 100% oxidation of the core zirconium as required by the YVL regulations. The dimensioning is based on collecting the released hydrogen in the gas space of the condensation pool compartment, which has no return flow routes to other containment compartments. The decay heat removal from the containment during a severe accident is carried out passively with the same natural circulation condensers used during other accidents.

The aim is to prevent entirely severe accidents during shutdown by ensuring that there is a sufficient water amount in the refuelling pool to flood the reactor core. The principle is good, but in the presented concept, the existence of a personnel access leading directly out of the space under the reactor significantly reduces the reliability of its technical implementation.

In principle there are no safety technical obstacles, which would hinder the approval of this reactor concept in Finland. However, several details of the design require closer examination or continuation of design work, especially concerning the capacity of the passive systems and possibly the recirculation arrangements of the emergency cooling systems. Additional work is also required for ensuring sufficient containment functions during shutdown periods, the same relates to all boiling water reactors. The outside cooling of the reactor pressure vessel, which is part of the severe accident management strategy, requires confirmatory research.

EP1000 and AP1000

The reactors in question are innovative pressurized water reactors designed by Westinghouse Electric (nowadays in BNFL's ownership) with a safety design based on passive systems. AP1000 with two main circulation loops is a 1000 MWe version of the 600 MWe passive pressurized water reactor AP600 previously designed by Westinghouse; the detailed design of AP600 is practically complete. The United States nuclear safety authority USNRC has granted a design certificate for the AP 600 plant based on its own design license review. AP1000 deviates essentially from AP600 only by the fact that some central components (core, steam generators, reactor coolant pumps, containment) are larger than those of AP600. The diameter of the AP 1000 reactor pressure vessel is the same as that of AP600, but it is longer by length corresponding to the change in core length. EP1000 is based on the component technology of AP600, but it is a 1000 MWe passive pressurized water reactor type equipped with a reactor pressure vessel larger by diameter and with three main circulation loops. The European power companies have also taken part in the design of EP1000.

The starting point of the safety design in this plant family is to carry out all safety functions in all operating situations, including transients and accidents, either with active systems designed for normal operation or solely with passive, naturally functioning systems. The containment is a large, dry containment and similar in design to the existing pressurized water reactors. The planned lifetime is 60 years.

As concerns these plant types the fuel, reactor core and reactor design are mainly implemented as in the current pressurized water reactors. However, it has been possible to significantly improve the safety for some parts. For example, the core boron content of all these reactors can be lower by design resulting in approximately one half of the highest boron content currently in use. This can be achieved in these plants by using so-called grey control rods with the help of which reactivity control can be carried out for a longer period parallel with the “usual” so-called black control rods as well as by using solid burnable absorbers in the fuel. This is why possible transients in the reactivity control due to boron dilution are entirely prevented or essentially reduced in effect compared to the current plants. The power densities of the AP600 and EP1000 reactors are lower than those of the current pressurized water reactors, and the cores are of medium height, which has an advantageous effect on the thermal safety margin. The thermal safety margins of AP 1000 correspond to those of the current large-scale pressurized water reactors. Its core is high and the thickness of the radial reflector surrounding the core has partly been reduced in order to increase the number of fuel rod bundles. This may have a disadvantageous effect on the embrittlement of the AP 1000 reactor pressure vessel during its long operating lifetime. The plant supplier has designed the discharge burnup according to the US practice, which clearly exceeds the limit considered safe in Finland. The core loading design can, however, easily be modified to meet Finnish requirements.

The primary coolant circuit represents the familiar, proven technology of the current pressurized water reactors with the difference that the reactor coolant pumps are a fixed part of the steam generator cold plenum. Each primary circuit loop has a hot leg, steam generator, two reactor coolant pumps and two cold legs. The loop seals in the cold legs typical of the existing plants are not used in the primary circuit loops, which for several reasons is advantageous for the safety. The reactor coolant pumps are hermetic by structure and therefore do not contain possibly leaking seals. Flywheels have been constructed to the pumps to give rotational inertia in spite of the hermetic structure. The pressurizer is of considerable volume, which significantly reduces transients in terms of pressure control of the primary circuit.

The secondary circuit of the plant types is essentially similar to that of the existing pressurized water reactors. The steam generators (two in AP600 and AP1000, three in the EP1000 version) are vertical U-pipe steam generators and their technology corresponds to that of the newest steam generators in use in the current plants, in which the problems connected with the integrity of heat transfer pipes typical of vertical steam generators are to be eliminated by material selections and structural solutions.

The safety functions are implemented with both active and passive systems, while the active systems are mainly designed solely for normal operation, and therefore they are not classified by safety. The safety functions are carried out in the following way:

- control of reactivity: the control rods and the boric acid dissolved in the primary coolant can be used as an active system controlling the power level. The safety is improved by decreasing the boron content from the existing pressurized water plants´ level. This has been carried out by using control rods of various types (so-called grey and black rods) and burnable absorbers in order to control the reactivity. The scram functions passively by dropping the control rods into the core, as is also the case today in the pressurized water reactors.

- cooling and decay heat removal:
    - an active 2x100% reactor decay heat removal system based on the pumps and heat exchangers classified as an operating system which can also be used as an active emergency cooling system in pipe breaks.
    - the passive system consists of cooling the reactor with so-called isolation condensers (one 100% condenser in the AP series, two in EP1000) functioning with natural circulation, as well as an active 2x100% auxiliary feed water system classified as an operating system in the secondary cooling circuit.
    - a safety/relief valve system in four stages each with a capacity of 2x100% is designed for the reactor depressurisation. The ultimate relief phase is coupled with the hot legs, while the others are coupled to the pressurizer.

- in order to mitigate possible pipe breaks of the primary coolant circuit, the reactor water supply is backed up with a 2x100% passive emergency cooling system which floods the reactor with borated water from the water tanks discharging with both gravity and gas pressure. A pipe break always actuates the depressurisation of the reactor coolant circuit. The EP plant comprises of two flooding tanks functioning with nitrogen pressure, and the AP series has one; the flooding tank functioning with gravity as the high-pressure system is larger in the AP series than in the EP plant.
- the long term decay heat removal resulting after a pipe break will be implemented with natural circulation between the flooded reactor and the containment (relief valves of the reactor coolant circuit coupled with the hot leg are needed for secure natural circulation)

- The basic type of the primary containment is a steel shell, large dry containment, the ordinary type used in the existing pressurized water reactors. Its decay heat removal is carried out passively: by natural circulation to the steel shell, by heat conduction through it and from there with natural circulation further to the environment. In order to increase the heat transfer the steel shell can be watered by gravity from the water tank designed to be placed in the upper part of the secondary containment. The primary containment is surrounded by a secondary containment made of concrete. The space between the primary and secondary containments is open to the environment concerning the section taking part in the natural circulation cooling of the steel shell, which is a deviating feature from the double containment principle. The section of the steel shell taking part in the cooling is, however, integral, i.e. without penetrations or other possible leakages. It can therefore be considered as a sufficient barrier, if the generation of missiles and their impact against the steel shell can be reliably prevented. All penetrations of the containment are placed in the lower part of the secondary containment, which, as usual, is isolated from the environment.

As a general rule, the safety systems seem to meet the failure criteria of the YVL guides, provided that some improvements (e.g. duplication of some valves which originally were designed single) are carried out. A final guarantee could be reached during the detailed examination of the system planning.

In Westinghouse's plants the start up of all passive safety functions is always driven by the control and protection system. The start-up is always carried out with a one-way valve operation. External power supply is thus necessary for the maintenance of the automation system operations and the start-up of the safety functions. The driving force of the actual working of the safety functions is then gravity or pressure of stored gas.

The severe reactor accident management strategy is based on arresting the core melt in the reactor pressure vessel by cooling it from outside. If this can be proved to work, no energetic threats will be directed to the containment due to the core melt water interaction, such as steam explosion, or core melt concrete interactions. According to the preliminary assessment made by STUK, successful external cooling can be proved at least with AP600 and EP1000, but the verification of sufficient margins still requires confirmatory research for this part. As to the AP 1000 concept, it can be difficult to prove reliably, if the AP600 pressure vessel diameter is used as originally planned.

The hydrogen control is to be carried out passively with catalytic recombiners as is also planned for the current pressurized water reactors. When dimensioning the hydrogen control it is important to prevent hydrogen burns in locations where the burn or a missile resulting from it can be a threat to the integrity of the containment. The containment decay heat removal during a severe accident is to be carried out by passive natural circulation and conduction through the steel containment.

In principle there are no safety technical obstacles, which would hinder the plant type's approval in Finland. The core design of EP1000 seems to be more successful concerning the safety characteristics expected from a passive plant than those of AP1000. Several details in the design, however, require closer examination or continuation of design work, especially concerning the capacity of the passive systems and failure tolerance as well as possibly the shielding of the inner containment against missiles. The severe reactor accident management strategy by cooling the core melt in the pressure vessel may be difficult to prove reliably in the AP1000 concept, if the pressure vessel diameter of the smaller capacity AP600 concept is kept in the AP1000 concept as originally planned. A study confirming this is also necessary for EP1000.

Evolutionary plants

VVER 91/99

The reactor in question is a pressurized water reactor with a design power of 1000 MWe offered by the Russian Atomstroyexport (ASE). VVER 91/99 is based on the VVER 1000 plants´ basic technology, and by its basic design it is a so-called evolutionary plant, i.e. a plant type equipped with active safety systems, improved from its basic technology by small steps. The containment is the ordinary, so-called large dry double containment. The plant can be designed for a 60-year lifetime.

The reactor plant is essentially similar to the existing VVER 1000 plants. The fuel and core design of the plant functions with similar practices as in the current large-scale pressurized water reactors with the difference that the coolant's highest boron content has been reduced by using solid absorbers in the fuel. The amount of control rods has, however, been increased to improve safety. The discharge burnup level of the core design meets the Finnish safety criteria.

There are four reactor coolant loops, each comprising of a horizontal steam generator, reactor coolant pump and loop seal. Additional examination concerning a brittle failure risk of the reactor pressure vessel may be necessary in order to prove the 60-year lifetime. Changing the manufacturing technology has eliminated the flaws of manufacture detected in steam generators during the VVER 1000 plant’s initial phase. A shaft seal is developed for the reactor coolant pump to be able to withhold at least one day (24h) even without cooling. The secondary circuit is essentially the same as that of the current plants, the pressure relief valves of steam generators are said to have been verified not only for steam but also for two-phase and water relief.

The key safety systems are implemented in the same way as in the present plants:

- The reactivity control is actively implemented with control rods and the boric acid dissolved in the primary coolant as well as with solid burnable absorbers. Due to the decreased boron content possible reactivity transients are less severe than those in the current pressurized water plants; the reactor can be shut down with an active 4x50% emergency boron system in addition to the passive dropping of the control rods.

- The emergency cooling of the primary circuit and the decay heat removal of the secondary circuit are active 4x100% systems, in addition the primary circuit comprises of an active 4x50% decay heat removal system. The electricity supply necessary for systems is consistently backed up in a multiple way. Meeting the failure criteria of the YVL guides does not seem to be problematic;

- The primary containment is the so-called large dry containment made of prestressed concrete, equipped with a leak-tight liner. The secondary containment of concrete is designed outside it. A 4x50% active system can be utilized for cooling the containment.

The severe accident management strategy is based on a core catcher at the bottom of the reactor pit. The hydrogen control has not yet been discussed, but the plant layout does not prevent its implementation, and it can be carried out as has been planned in the existing pressurized water reactors.

Based on the preliminary studies, especially concerning the Russian supply, doubts could arise concerning the manufacturing supervision of mechanic components and its functioning according to modern quality assurance principles. This requires especial effort from the client.

In principle there are no safety technical obstacles for the approval of this plant type. In practice several details of the design (e.g. automation system, some technical questions related to decay heat removal of the primary circuit, recirculation arrangements of the emergency cooling systems and severe accident management strategy) require closer examination and continuation of design work. The supervision of the manufacturing following modern quality assurance principles requires special effort from the client. The plant supplier has expressed its readiness to procure entire systems or components from the subcontractors chosen by the client itself.

EPR

The reactor in question is a pressurized water reactor with a design 1500 MWe power offered by the German-French Nuclear Power International (NPI). (NPI, a mutually owned company by Siemens and Framatome has been merged to Framatome ANP as part of the mutual reorganizations between these suppliers). EPR is based on the basic technology of the German 1300 MWe Konvoi-series plant and the French 1450 MWe N4-series plant, and, by its basic design, it is a so-called evolutionary plant, i.e. equipped with active safety systems, streamlined by small steps from its basic technology. During the design period the plant suppliers have been in close cooperation with their domestic nuclear safety authorities. The containment is the ordinary, so-called large dry double containment. The planned lifetime is 60 years.

The reactor plant is essentially similar to that of the current pressurized water reactor plants. Similar practices to those, which are in use in the large-scale existing pressurized water reactors, are followed in the fuel and core designs. The control of reactivity during operating cycle is carried out with boric acid dissolved in the primary coolant with a content corresponding to the current plants. The discharge burnup designed for the Central European plants significantly exceeds Finnish safety criteria. The core loading design can, however, be modified to meet Finnish requirements.

There are four primary circuit loops, each comprising of a steam generator, reactor coolant pump and loop seal. The problems related to the integrity of heat transfer pipes typical of the vertical steam generators are eliminated by material selections. There are two optional reactor coolant pumps, one German, the other French, and the shaft seal of each is designed to be self-sealing during transients.

The central safety systems are carried out in the same way as in the existing power plants:

- The control of reactivity (power regulation) is implemented with control rods and boric acid dissolved in the primary coolant which makes the possible reactivity transients essentially similar to those of the current pressurized water reactor plants; the boron content is kept at the level of current plants. The reactor can be shut down with an active 2x50% boron system in addition to the passive drop down of control rods.

- The emergency cooling and decay heat removal systems are designed to be 4x50% active for both the primary and the secondary circuits. The electricity supply is backed up correspondingly. The dimensioning basis for emergency cooling is the break of a pipe (but not the main circulation pipe) connected to the largest main circulation circuit: when considering the main circulation pipe break, the emergency cooling capacity is in the class of 4x34%. The dimensioning of the emergency cooling system is based on the leak-before-break principle (LBB).

- The primary containment is the so-called large dry containment made of prestressed concrete and the secondary containment made of concrete is designed outside it. The intention of the supplier is to eliminate the steel liner of the primary containment, but according to French experience the leak tightness of steel concrete containments without liner is poor.

There is no problem to meet the failure criteria of the YVL guides except for the boron system and emergency cooling necessary for main circulation pipe breaks.

The severe accident management strategy is based on cooling the core melt in a specific area in the lower section of containment, adjacent to the reactor pit. The procedure is rather complicated and though the individual phases related to cooling by melt spreading are reasonably understood, it is not yet clear, if the timing of core melt movements can be controlled with the accuracy required of this procedure. The depressurisation of the primary circuit is necessary for severe accident management, but the basic concept is equipped with only one relief valve. The version offered to Finland comprises two pressure relief valves. The cooling of the spread area is carried out with a 2x50% active system, which means that the single failure criterion required of the systems designed for severe accidents is not met. The hydrogen control is passively carried out with catalytic recombiners as is also planned in the current pressurized water reactors - the cooling of the containment is actively carried out with a separate cooling system.

In principle there are no insurmountable safety technical obstacles, which would hinder the acceptance of this plant type in Finland. However, several details of the design require additional work, especially concerning reactor core design, reactor emergency boron system, containment liner, capacity of reactor emergency cooling systems and recirculation arrangements of emergency cooling systems as well as severe accident management strategy. The cooling of core melt related to the mitigation of severe accidents is complicated in this plant, and therefore its successful function is difficult to prove in a dependable way.

EABWR

The reactor in question is a boiling water reactor with a design power of 1400 MWe offered by the American General Electric (GE). EABWR is designed for the European market. It is a version of the ABWR plant type which is based on GE's previous boiling water reactor technology and, by its basic design, it is the so-called evolutionary plant, i.e. equipped with active safety systems, streamlined by small steps from the basic technology. The United States nuclear safety authority USNRC has granted a design certificate for the ABWR plant based on its own review. This plant type represents development compared to GE's previous products, and its basic solutions are very similar to the TVO Olkiluoto plants designed during the 1970´s (in which during their operation several modifications improving the safety have been carried out e.g. severe accident management measures). The containment is ordinary pressure suppression containment with the condensation pool, the severe reactor accident management strategy has been planned only after the containment layout has been decided.

The reactor is a boiling water reactor equipped with internal reactor coolant pumps, which, for the operating parameters and safety characteristics, corresponds to the current large-scale boiling water reactors. The same concerns the safety features of the fuel and reactor core designs, no improvement in the present large-scale boiling water reactors has been searched for. The control of reactivity during power operation is carried out with solid burnable absorbers in the fuel and the control rods. The discharge burn up designed for the US plants significantly exceeds Finnish safety regulations.

It is probable that the turbine plant will be similar to those of the current boiling water reactors, however, with the difference that the protection measures, which are an essential part of reactor safety, starting as a consequence of a disturbance in the turbine plant can be carried out more straightworthy than in the existing Finnish boiling water reactors.

The central safety systems have been carried out in the same way as in the current plants:

- reactivity: the power level is controlled with active systems, the control rods (electric motor drive) and reactor coolant pumps, as is also in the existing boiling water reactors; the scram is carried out hydraulically with control rods and backed up with an active electric motor drive. The single failure tolerant active boron system N+1 can be used as a parallel shutdown system.

- cooling and decay heat removal:
    - a 3x100% reactor decay heat removal system based on pumps and heat exchangers is functioning as an active system

- the passive system consists of so-called isolation condensers (3x approx 50%) functioning with natural circulation and cooling the reactor
- 18 safety/relief valves for the reactor depressurisation, some of which can function with principles different from the others
- in order to mitigate possible pipe breaks of the primary circuit the reactor is equipped with the 3x100% emergency cooling systems to some extent as now is the case in the boiling water reactor plants; one subsystem uses steam thus differing from the other (electric driven) subsystems.

- the basic type of containment is a pressure suppression containment equipped with condensation pool, the ordinary type used in the current boiling water reactors. Its decay heat removal is actively carried out using the same 3x100% decay heat removal system as the reactor.

The emergency systems appear to meet the failure and deviation criteria of the YVL guides with some exceptions (boron system); this could be finally proved by a detailed examination of system design.

The severe accident management strategy is based on cooling the core melt in the reactor pit. To promote the cooling there are passively opening flooding valves on the wall between the reactor pit and condensation pool. Compared to the large amount of core melt the reactor pit is narrow. Large-scale international research programs have tried to prove experimentally and theoretically the coolability of the core melt spread on a concrete floor, however, without success so far.

During operation there is a nitrogen atmosphere in the containment, which makes the hydrogen burns impossible during the accidents initiated during power operation. The primary containment is, however, dimensioned to be so small that, in order to control a 100% hydrogen generation, a separate “storage building” has been designed to which excessive gas pressure can be released. The cooling of containment during a severe accident is to be carried out with external active systems.

Precautions for accidents during shutdown have not been separately designed for this plant type.

In principle there are no insurmountable safety technical obstacles that would hinder the acceptance of this plant type in Finland. However, several details of the design require additional work (e.g. some features of system planning, such as the capacity of the reactor boron system and failure tolerance as well as recirculation arrangements of the emergency cooling systems). Significant amount of additional work is required for taking care of the severe accident management strategy as a whole in an acceptable manner and for ensuring sufficient containment function during shutdown periods.

BWR 90+

The reactor in question is a boiling water reactor with a design power of 1500 MWe offered by Westinghouse Atom (previously Asea Atom, then ABB Atom, today in BNFL's ownership). BWR 90+ is based on the previous boiling water technology of Asea Atom, and by its basic design it is the so-called active evolutionary plant, i.e. equipped with the active safety systems, improved from the basic technology. The containment is the ordinary pressure suppression containment with condensation pool, however, the requirements the severe accident mitigation have been taken into account from the very beginning when dimensioning the containment.

The reactor is a boiling water reactor equipped with internal reactor coolant pumps and, by the operating parameters and characteristics, it corresponds to the current large-scale boiling water reactors. The same concerns the safety features of the fuel and core design, no improvement in the current large-scale boiling water reactors have been searched. During operation the reactivity is controlled with solid burnable absorbers and control rods.
The turbine plant is probably the same as that of the current boiling water reactors, however, with the difference that the protection measures, essential to reactor safety, starting up from a disturbance in the turbine plant can be carried out more straightforwardly than those in the existing Finnish boiling water reactors.

The key safety systems are carried out in the same way as in the current plants in principle, however, emphasizing more diversity than redundancy:

- control of reactivity: the power level is actively controlled with control rods (electric motor drive) and reactor coolant pumps, as in the existing boiling water reactors; the scram is hydraulically carried out with control rods, actively backed up with an electric motor drive, and the boron feed system functioning with gas pressure can be used as a parallel cooling system.

- cooling and decay heat removal:
  - the passive system consists of cooling the reactor with so-called isolation condensers (2x50%) functioning with natural circulation
  - for the reactor depressurisation there are preliminary 24 safety/relief valves, three different types
  - in order to mitigate possible pipe breaks of the primary circuit the reactor is equipped with active emergency systems to some extent as in the existing boiling water reactors. The amount of pumps has, however, been reduced by increasing their capacity, redesigning their operational range and diversifying the power sources. There are two high pressure emergency cooling pumps, each with a capacity of 100% in full pressure and 50% in situations with quickly reduced reactor pressure, and the generator as power source driven by a gas turbine. The low-pressure pumps comprise of two 100% pumps the diesel generator as the power source. A separate "auxiliary feed water system"

operated by gas turbine can also be used for emergency cooling, its capacity corresponds to one high-pressure pump, but it is classified entirely as a system for normal operation.

- the basic type of containment is the usual pressure suppression containment equipped with condensation pool as in the existing boiling water reactors; its decay heat removal is actively carried out with its own 4x50% decay heat removal system using diversified power sources (diesel generators and gas turbine generators)

The emergency systems appear to meet the failure and diversity criteria of the YVL guides; this could be finally proved when examining the system design in detail. Emphasizing the diversity principle in addition to redundancy, especially concerning the energy sources of active systems, represents advanced safety thinking.

The severe accident management strategy is based to cooling the core melt with a core catcher placed at the bottom of the reactor pit. The catcher is cooled with the condensation pool water on the exterior surface. It can also be flooded from upwards. The principle is promising and the reliability of the catcher can be proved at least for the external cooling according to the preliminary assessment made by STUK.

During operation there is a nitrogen atmosphere in the containment, and therefore hydrogen burns there are impossible during the accidents initiating during power operation. The primary containment and its volume distribution are dimensioned so large that even with a 100% hydrogen generation the successful hydrogen control can be considered very likely. During a severe accident cooling the containment is carried out with separate active external systems.

In the design of containment precautions are made for accidental situations estimated possible during refuelling shutdowns. The volume under the reactor is dimensioned so that the reloading pool water is sufficient to flood the reactor core, and the leak paths out of the containment are entirely eliminated below the top of reactor core. Controlling the situation for a longer period after the flooding still requires continuation of design work.

In principle there are no safety technical obstacles, which would hinder the approval of this plant type in Finland. However, some details of the design, such as recirculation arrangements of the emergency cooling systems require continuation of design work. Additional work is required for ensuring the sufficient function of containment during shutdown. The core catcher concept designed for a long-term cooling of the core melt possibly being discharged from the reactor during a severe reactor accident seems promising in itself, but confirmatory research to prove its function is still required.