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RECENT REGULATORY ISSUES IN FINLAND

This paper presents general regulatory issues from Finland since the last VVER Regulators Forum meeting in Odessa 11-13 October 2000. More specific issues concerning Loviisa nuclear power plant are described in the Annex of this paper.

1. STUK completed its preliminary safety assessment for a Decision in Principle for the fifth nuclear unit in Finland

In November 2000, the Finnish utility Teollisuuden Voima Oy (TVO) submitted to the Council of State an application for a Decision in Principle for the fifth nuclear reactor unit to be constructed either in Olkiluoto or in Loviisa. The application has been reviewed by all stakeholders, and the respective statements have been submitted to the Ministry of Trade and Industry that prepares the issue for the Council of State decision making. One of the binding statements to the ministry is STUK's safety assessment. By the Atomic Energy Act, STUK has to check whether there are any shortcomings in the proposed designs that would render them unsafe or that would prevent them from meeting Finnish safety requirements, even if modified.

The plant concepts covered in the application were as follows:

Boiling water reactors:
- BWR 90+ (Westinghouse Atom)
- EABWR (GE Nuclear Energy)
- SWR 1000 (Siemens Nuclear Power)

Pressurized water reactors:
- AP 1000 (Westinghouse Electric Company)
- EP 1000 (Westinghouse Electric Company)
- EPR (Nuclear Power International)
- VVER 91/99 (Atomstroyexport)

STUK's preliminary safety assessment concluded that all the proposed designs could be made to meet Finnish safety requirements, but that all the designs would require some modification to do so. The nature and type of the required modifications varies a lot between the different designs. One general shortcoming is that all the designs need at least some improvement to adequately deal with severe accidents in which safety systems fail to perform to such an extent that the reactor core is extensively damaged. Two of the alternatives would require significant work to meet this requirement. Numerous additional, smaller design-related items were also identified, but these could be addressed in the design phase and be reviewed in connection with the detailed construction permit review. The preliminary safety assessment also covered numerous other related subjects, such as reactor fuel and spent fuel management questions, waste questions, the overall safety of the proposed sites, and the competence of the applicant.

The overall conclusion of the preliminary safety assessment for a fifth nuclear reactor is that there are no obstacles in principle that would prevent the project being completed safely. Major engineering work and organisational development will be needed in subsequent stages, however, to actually complete such a major project safely.

The preparation of the Council of State decision is pending until an appeal to the administrative court has been brought to a verdict. That appeal challenges the legal validity of the statement by one of the two municipalities where the plant would possibly be located. It is expected that the Council of State decision could be made in the fall, and a positive decision by the Council of State would still need to be confirmed by the Parliament.

2. Decision in Principle of the Parliament for the Final Repository of the Spent Fuel

The Finnish Parliament approved after voting (159 for, 3 against) in May 2001 a Decision in Principle for the final repository of the spent fuel. This approval was preceded by an approval of the host municipality in January 2000 and by the Council of State in December 2000. The next step will be a construction of an underground research facility that would later on become a part of the final repository. Further research and development work will be done in that research facility prior to the submittal of a construction license application. At the moment, the construction of the final repository is planned to be started in 2010 and the operation of the repository in 2020. The research, development and planning work for spent fuel disposal and its later implementation is carried out by Posiva Oy, a company owned by the Finnish nuclear power plant utilities.

3. Prescriptiveness of the STUK's Regulatory Guides studied by VTT

During the first review meeting of the Convention on Nuclear Safety in 1999, a question on possibly too prescriptive and detailed regulatory Guides (YVL Guides) in Finland was raised by some other countries. The concern behind the question was that if the system is too prescriptive and detailed the own initiatives of the licensees may be hampered in safety issues, with a negative impact on their safety culture. An independent review to study the prescriptiveness of YVL Guides was performed by the Technical Research Centre of Finland (VTT) in 2001. The study was mainly based on the results of interviews with the licensees. A draft report of the study was received at the end of April. The main conclusions and recommendations of the final report will be included to the Finnish Nuclear Safety Convention report for the second review meeting.

Based on the interviews and a general assessment YVL Guides were not considered to be too prescriptive and binding for the nuclear utilities in Finland. All persons interviewed had a clear positive view of the YVL Guides and they were seen as giving structure to the safety activities at the plant. This positive view has however to be qualified with respect to a few problematic YVL-guides. It was concluded that YVL Guides have a logical and covering structure, they are relatively well balanced with a reasonable level of detail, and the requirements put forward are reasonable reflecting the international practise. It was also said that Guides are understandable and quite easy to interpret. A considerable effort is put by STUK to keep the Guides up to date.

However, some recommendations were given in the report for how to further develop the YVL Guides. For example one general recommendation was that there is a need to address how the YVL Guides could be improved with the aim of making interpretations less dependent on the inspectors. Another more general recommendation was related to the safety requirements building on a combination between deterministic and probabilistic considerations. It was seen difficult to find a proper balance between the two principles. This has to do with an interpretation of the residual risk and an agreement when it is small enough. There are also some improvements to be implemented in a short-term perspective related to the few problematic YVL Guides. It was also concluded that STUK should initiate a discussion of a more long-term strategy for the development of the YVL Guides related to the possible needs for harmonised regulatory approaches.

The results of the study will be carefully analysed at STUK and a strategy for the development of Finnish regulatory Guides will be planned and implemented in the near future.

4. Common cause failure of motor-operated valves in the RHR system at Olkiluoto reactors

In August 2000 it was found that one of the motor-operated isolation valves of the core spray system at Olkiluoto 1 unit did not pass the periodic test (failed to open) specified in the Technical Specifications. It was found out that some cogs of the gear had ruptured. The actuator was replaced with a spare actuator. In December 2000 another isolation valve of the same system failed to close in a periodic test. In this case two cogs of the gear had ruptured. The gear material in these AUMA actuators (type SA 140-63) was bakelite. In the second event the bakelite gear was replaced with the brass gear as corrective action.

A detailed investigation of the rest of AUMA actuators in the core spray system of Olkiluoto 1 and 2 units revealed that all eight gears concerned had cracks that were found by penetrate testing. One of most damaged actuator was tested to demonstrate whether it would have been operable in accident situation. The actuator managed the test, but after the test it was found out that again one cog was ruptured. After evaluation of the test results all bakelite gears of the actuators in the core spray systems were replaced with actuators having brass gears.

According to material technological studies the probable reason for cog failures was fatigue. Root cause of the event is evaluated to be deficiencies in the operating experience feedback process. Similar actuators have broken in the past both at Olkiluoto and at Swedish nuclear power plants. Obviously these events were not adequately considered when maintenance and replacement of this kind of actuators were planned at Olkiluoto even though information was available. Corrective actions of the utility are focused to fix these deficiencies.

When assessing the safety significance of the event it has to be noted that the system has four redundancies and even one redundancy is enough to ensure the integrity of the fuel in all emergency core cooling situations. In case all redundancies fail at the same time, which is not expected taking into account the test results, the auxiliary feedwater system could take care of the cooling of the core in most accident scenarios, if it operates at full capacity. Because of the common cause failure the event was classified level INES 1.

5. Results of the investigation related to the delays and deficiencies in implementation of a risk reducing modification at Loviisa NPP

Loviisa nuclear power plant has planned to improve the reliability of primary circuit pump (PCP) seal cooling in order to assure their leaktightness. According to PSA results, a loss of cooling of PCP seals is a significant contributor to core damage frequency at Loviisa NPP. Currently the PCP mechanical seals are cooled by primary circuit water, which is cooled by an intermediate cooling system. The intermediate cooling system is cooled by service water system. The planned modification would allow the use of boron injection system for cooling of PCP seals independently of the service water system, which could be lost during certain abnormal weather conditions or internal floods. After installation of a diverse seal cooling system the calculated core damage frequency will decrease by about 20 %.

The modification was planned to be implemented during the outage of 2000. Because of delays and deficiencies in the modification project, both at the utility and at STUK, the implementation could not be carried out and had to be postponed to the outage in 2001. STUK decided to assign an investigation team to investigate the causes of the delays and deficiencies in the modification project. Investigation revealed shortcomings in the actions of both the utility and STUK.

One of the most important causes for the utility's belated accomplishment of plans was that implementation decisions were postponed because time and human resources allocated to design were insufficient and the number of necessary analyses was underestimated. In addition, the design criteria set for the seals were insufficient and the seals' actual endurance was not verified.

STUK failed to perform its regulatory process related to this modification in time due to a coincidence of several factors. Design documents were submitted to STUK for approval only one month before the planned implementation and these documents were reviewed simultaneously with many other modification documents. Priority was not put on this modification due to shortcomings in communication within STUK on the safety significance of this particular modification. A more generic factor contributing to the result is that STUK did not manage the entire modification project in a centralized manner.

Based on the results of the investigation STUK is now revising its own regulatory procedures related to modification projects to eliminate emerged shortcomings. More emphasis will be put on coordination and communication of large safety significant modifications in order to avoid for example gaps in the information flow within STUK. This means more responsibility on those in charge of modification projects to take care of the timeliness and comprehensiveness of STUK's regulatory activities and decisions.

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General operating experience from Loviisa npp from 2000-2001

In 2000 Loviisa 1 produced 3798 GWh (gross), the capacity factor was 84,8 % and the refuelling and maintenance outage lasted 44 days. Loviisa 1 had an inspection outage, which is performed every fourth year. In 2000 the gross production of Loviisa 2 was 4075 GWh, the capacity factor was 91,0 %, and the refuelling outage lasted 19 days. The collective radiation doses in 2000 were 1,73 manSv for Loviisa 1 and 0,54 manSv for Loviisa 2.

Seven events in 2000 were classified on the International Nuclear Event Scale (INES). There was one level 1 event and the classification of the other events were 0. The level 1 event occurred at Loviisa 1 during the refuelling outage in 2000 when recurrent line-up errors (in filling the suction line of the draining pump in the fuel pond cooling system) caused reactor pool water leakages onto the floor of the steam generator compartment. In the first four months of 2001 three events have been classified on the INES scale. These events have been below the INES scale (level 0).

The first stage of the final repository for medium and low level radioactive waste was licensed in 1999. The first stage includes underground tunnels for solid low active waste. In 2001 STUK approved the Preliminary Safety Analysis Report of the solidification plant and detailed planning of the plant is going on. The civil construction works will be started later in 2001.

Recent and planned plant modifications at Loviisa

Following plant modifications have been implemented at Loviisa NPP during 2000 and the first half of 2001.

- Motors of the Service Water System pumps (2 out of 4 at the moment) of Loviisa unit 2 (VF) have been recoiled in order to strengthen the reliability of the system. The cause for the recoiling is that based on the analysis results the power needed in certain undervoltage situations is more than originally estimated. Based on the analysis of Loviisa unit 1, recoiling of pump motors is not needed.
- New fire detection system has been installed at both plant units. This has included for example new detectors and computerised indication system in the main control room.

Following plant modifications will be implemented at Loviisa NPP during next years

- Based on the updated risk analysis results, containment bypass leakages (VLOCA) are a significant risk contributor. The risk builds up of several different factors, and one factor that is being tackled in the outage 2001 is the reliability of the isolation of primary circuit drainage and went system (TY). To improve the reliability of the isolation of TY system, two additional motor operated isolation valves will be installed into two different TY system lines.
- Connecting lines between drainage lines in the hot legs and cold legs of primary loops will be removed at both plant units (removed from Loviisa 1 in 2000). These were installed for the management of specific small break LOCA situations. Lines

are found to be very sensitive for cracking because of thermal stratification. According to the safety analysis such lines are not necessary.

- Strengthening of the ultimate heat sink and residual heat removal. The purpose is to ensure the cooling of the unit to the cold shutdown state in a loss of turbine hall or normal heat sink. The modifications include additional possibility to get water from the outlet cooling channel to the inlet, and an additional residual heat removal exchanger with pumps outside the turbine hall.
- Improving the cooling of the primary pump seal water. Modification includes changes to the automation that provides cooling water to the seals in case of a temperature increase of the normal seal water.

A partial isolation of the containment followed by manual reactor scram and non-compliance with Technical Specifications during start-up at Loviisa 1

Loviisa unit 1 was at full power on 9.2.2001, when a partial isolation of the containment occurred and was followed by manual reactor scram. Before the isolation, maintenance personnel were removing unnecessary electrical cables in the electrical room. Cables that were going to be removed by cutting were replaced in 1986 by new ones due to modification. However, cables were left in the cabinets in case they would have been needed afterwards. The need never realised and cables were decided to be totally removed from the cabinets. When cables were cut, failure signals were induced because some of the cut cables were still connected from the other end to the instrumentation cabinets of plant protection system. These failure signals triggered mini circuit breakers of the cabinets and cabinets lost electricity and consequently actuated partial containment isolation. due to isolation some valves closed.

One of the closed valves prevented flow outside containment of the leakage control system of seal water system of PCPs and thus a relief valve in the steam generator compartment opened and caused a small leak. Leak was soon detected by the operators who quickly scrammed the reactor and stopped PCPs and isolated the leak. Operator actions during the event were seen appropriate. Causes for the event were that cables were not removed as instructed in the working documents. However, working documents did not specify that work is related to the plant protection system. If it would have been in the documents, documents would have been reviewed by the safety unit and work would have been postponed to the outage. In the future, cables connected to the plant protection system will be removed only during outage and working document deficiencies will be included in a ongoing root cause analysis related to the deficiencies in the administrative control of maintenance work at Loviisa.

When the cause of the isolation was found and direct corrective actions implemented, the power of the reactor was increased about 14 hours after scram. Reactor was stabilised according to operating procedures at 53 % power level for one hour. During the stabilisation phase, the position of the controlling group control rods settled under the limit specified in the Technical Specifications (175 cm) for this power level. Operators managed to get the reactor in compliance with Technical Specification limits within 4 hours.

At Loviisa reactors the peak of Xenon poisoning is about 9 hours after scram after which it starts to decrease. In this case, operators started to increase reactor power about 14 hours after scram, which accelerated also loss of Xenon in the reactor. In this kind of situation, if inadequate amount of boron is injected into to reactor, control rods start to sink even though power of the reactor is increased at allowed gradient.

During the start-up, operators injected boron into the reactor but too late and too little. This caused the before mentioned non compliance with Technical Specifications limits. This resulted in alarms related to the exceeding of linear power and temperature of coolant in some fuel assembly positions. The safety significance of these deviations were assessed small because exceeding of the limits were small.

The cause of the event was misjudgement of the operators, who understood the situation as a transient during which it is allowed to deviate few hours from control rod positions specified in the Technical Specifications. However, the operators did not know precisely the amount of boron which needed to be injected into the reactor and what is the effect of power increase to the Xenon poisoning and related time constants in this situation. Tools for the analysis of Xenon transient and needed boron injection are available at Loviisa, but they were not used beforehand because of several tasks of operators related to the start-up without extra manpower in the control room.

This event will result in training of operators and revised operating procedures related to the use of the analysis tools before start-up.

Development of plant modification management

Analysis of reported events often reveal that deficiencies of modification management have been a contributing factor. Such deficiencies include late planning, lack of co-ordination with other works, last moment changes, documentation defects, unfinished disassembling works and delayed updating of the documentation.

Proper planning and scheduling are the key factors in modification management. Loviisa Power Plant has completed an extensive project training course in 2000 for those in the operating organization who will be involved in future modification projects. Successful projects like the plant modernisation and power upgrading have been used as good examples.

The scheduling of the modification planning for the next outage is fixed in order to get enough time for preparations. Minor modifications are concentrated to every second annual maintenance outage and major works are carried out every fourth year. This is accomplished by starting from a long term investment planning which converts into a long term modification plan.

During the maintenance outage the scheduling office is directing their efforts from the previous control of the overall schedule to controlling the individual work packages including also the modification works. In the main schedule more time is allocated to tests related to start-up. New arrangements for handling the work orders in the main control room have been introduced for the next annual outages. The idea is to even up the work load in the main control room and decrease the disturbance of the operators.

Quality procedures for executing modifications have recently been updated. The authority to make decisions on last moment changes in the scope or schedule of the modification works has been clarified.

Development of the quality system

After Fortum Corporation was formed a need for an updated quality policy was obvious. In 1999 a quality statement "Fortum's Policy Commitment to Quality in the Nuclear Power Operations" was issued by the president of Fortum Power and Heat Oy.

Recent development of the plant quality management system is based on the principle of continuous improvement in accordance with the observations and remarks made in quality audits. Loviisa Power Plant adopted in 2001 a newly formulated management procedure which defines an annual planning process from strategic planning to annual reports. A first 10-year strategic plan for the power plant was developed in 2000.

A second important and new procedure describes those review processes (e.g. management reviews, self assessments), which are needed in an effective quality management system.

In the internal quality audits new efforts are directed to the evaluation of recurrence of events. This have considerably increased necessary background work both in the preparation and in the reporting phase of an internal audit.

An evaluation on the plant quality management system compared to the ISO/DIS 9001, 9004:2000 standards was made in 2000 by Fortum Nuclear Services. The work continues in 2001 and a similar comparison with IAEA Safety Series No. 50-C/SG-Q has already been ordered.

Preparation of the environmental management system according to the ISO 14001 standard is included in the quality management system. Preparations at the procedure level have introduced a new chapter in the Quality Manual and updating of numerous quality procedures. A novel environmental aspect shall be considered in internal audits and new part-time auditors have been trained for environmental evaluations. The readiness for certification of the environmental management system should be achieved by the end of 2001.

The present tracking system for quality and safety decisions, obligations and actions has capacity limitations and a new tailored application will be delivered in 2001.

Summary of the modernisation and power upgrading project

The project for the modernisation and power upgrading of the Loviisa NPPs gave an excellent possibility to take advantage of the latest development in the nuclear power plant technology. The key aspects were to verify the plant safety, to improve production capacity and to give a good basis for the extension of the plant's lifetime to at least 45 years.

Feasibility study and project objectives

In the first phase, before starting the project, a feasibility study for uprating of the reactor thermal power was carried out. The main result was in short that no technical or licensing issues which would prevent the raising of the reactor thermal output up to 1,500 MW from the original level of 1,375 MW could be found.

The carefully prepared feasibility study gave a good picture of the necessary plant modifications as well as essential areas in the analysis work, which was of use in planning the critical works and the time schedule of the project. The feasibility study focused on the following tasks:

- The optimisation of the power level and definition of the new parameters of the main process
- Reactor core and fuel studies, including RPV irradiation embrittlement
- Safety analyses and licensing
- The main components and systems
- Project planning and risk assessment

The main objectives for the project were based on the feasibility study:

1. Plant safety level as a whole will be checked and, if needed, improvements will be made,
2. Plant units will be licensed for 1,500 MW reactor thermal output,
3. Gross electric output of the plant units will be raised to about 500 MW,
4. Assistance to the life time extension of the plant units,
5. The long-term availability of the plant is not impaired
6. Increase in the expert knowledge of staff.

Time schedule and project organisation

The feasibility study concerning the reactor power upgrading and improvements of the turbine efficiency was started in spring 1994. After good results from the study, the preparation of the project plan began in summer 1995. Critical works in the time schedule, such as the revision of the Final Safety Analysis Report and the preparation of certain plant modifications, were started immediately.

The first step of the trial run by 103% reactor power could be started in January 1997. Test runs continued step by step during the year, and the last transient test by final reactor power was completed successfully in December 1997.

The Council of State awarded a new operating license for the Loviisa NPP in April 1998. The license is valid until the end of 2007 for 1,500 MW reactor thermal power, which is 9.1% more than the previous power level of 1375 MW. Measures to improve the efficiency of the steam turbines will continue in the annual maintenance outages until the year 2002.

The implementation of the project was carried out in co-operation between the Loviisa NPP and Fortum Engineering. In addition, many other organisations such as the Technical Research Centre of Finland (VTT) participated in the work. Special attention was paid to the QA routines in the project as well as to the co-ordination of the work in several organisations. One example of this was the particular subject-specific specialist groups which were established to overview essential sections such as nuclear safety and commissioning.

The work was divided into the following ten sub-projects each having a responsible person from the organisations of both the Loviisa NPP and Fortum Engineering:

1. Operating licenses
2. Other licenses
3. Safety analyses and basic data management
4. FSAR revision and comparison of the plant with regulatory body guidelines
5. PSA (including level 2 PSA)
6. Modification of the turbines
7. Electricity systems
8. Reactor and fuel
9. Process systems and automation
10. Commissioning and revision of instructions

Technical implementation and experience of the trial operation

Increasing the electrical output by about 50 MW at each unit was part of the Loviisa modernisation programme. After completing the upgrading of the reactor thermal output in April 1998, more than 80% of the total increase in the electrical output was fulfilled. The rest of the power increase is available when the measures to improve the steam turbines are completed in the year 2002.

The reactor power upgrading from 1375 MW to 1500 MW was planned on the basis of optimising the need for heavy plant modifications. In the primary side and the sea water cooling system, the mass flow rates were not affected, but the temperature difference has been increased in proportion to the power upgrading. In the turbine side, the live steam and the feedwater flow rate were increased by about 10%; the live steam pressure was not changed.

The reactor fuel loading was considered on the basis of the previous limits set for the maximum fuel linear power and fuel burn-up. The increase in the reactor thermal output was carried out by optimising the power distribution in the core and the power of any single fuel bundle was not increased above the maximum level before power upgrading. In parallel with this work, more advanced options related to the mixing rate of the cooling water in the fuel subchannels and the increasing of fuel enrichment were investigated. The dummy elements installed on the periphery of the core in Loviisa 1 and 2 were preserved to minimise irradiation embrittlement of the reactor pressure vessel.

The VVER 440 design margins in the primary side are rather large and the hardware modifications needed there were quite limited. Replacement of the pressuriser safety valves was indicated already during the feasibility study as a necessary measure because of the power upgrading. Most of the other substantial measures in the primary side were carried out on the basis of the continuing effort to maintain and raise the safety level of the plant, and they were not directly included in the power upgrading.

It was necessary to carry out more extensive measures in the turbine plant and to the electrical components. Steam turbines were modified to a higher steam flow rate. Because of these measures, also the efficiency and operation reliability have improved. Certain modifications were carried out in the electrical generators and the main transformers to ensure reliability in continuous operation with the upgraded power output.

The last step in the process to upgrade the reactor thermal power was the long-term trial run to verify the main process parameters as well as plant operation in both steady state and transient situations. The trial run was carried out in gradually upgraded reactor power with a power level of 103%, 105%, 107% and finally 109%. Transient tests defined in the test programme were performed with a reactor thermal power of 105% and 109%. The test results correspond very well with all analyses and calculations. All the acceptance criteria for the tests were fulfilled.

Licensing procedure and safety analyses

The modernisation programme as a whole was started from the basis of the positive safety progress. This was applied by taking advantage of the latest development in calculation codes and technology as well as feedback of the operating experience, expertise in the ageing processes and safety reassessment coupled with the evolution of safety standards.

STUK was closely involved at every stage of the project, from the early planning of the concept to the evaluation of the results from the test runs. STUK examined all the modification plans that might be expected to have an impact on plant safety. Individual permits were granted on stage by stage, based on the successful implementation of previous work.

The renewal of the operating license for the increased reactor power was carried out in the following steps:

- permission from the Ministry of Trade and Industry to make plant modifications and test runs with upgraded reactor power under the existing operating license and under the control of STUK
- assessment of the environmental impact (EIA-procedure) of the project
- STUK's approval of the Final Safety Analyses Report (FSAR), the safety-related plant modifications, test programmes and results.
- the Ministry of Trade and Industry, the responsible authority for the NPP operating licenses, received a statement from several local and national organisations
- the operating license was prepared by Ministry of Trade and Industry, and the Council of State awarded the license in their session on 2 April 1998. The license is awarded to 1,500 MW nominal reactor thermal power until the end of the year 2007.

The environmental impact has been assessed in the EIA Report, which was completed in December 1996. This was the first time in Finland (parallel with TVO plant having a corresponding modernisation programme) the EIA Procedure has been applied to a nuclear power plant. The law and the decree set certain procedures, including the public hearing for screening, scoping and the EIA statement, which are the stages of this procedure.

The result was that the reactor thermal power upgrading has no other considerable environmental impact than a slight increase in the outlet temperature of the cooling water. This means that the maximum temperature increase of the cooling water in the main condenser, before releasing back to the sea, is about 1°C higher than the previous temperature increase, which was typically close to 10°C.

An extensive safety review and comparison of the plant with the latest national regulatory body guidelines (YVL guides) have been carried out. This work was performed taking into account many international standards, such as the IAEA standard "A Common Basis for Judging the Safety of Nuclear Power Plants Built to the Earlier Standards INSAG-8". As a result of the work, a particular safety review report has been completed.

A part of the safety review and the licensing process of the reactor power upgrading was the renewal of the Final Safety Analyses Report. New accident analyses have been made concerning the containment pressure, LOCA and MSLB, for example. In addition to the accident analyses, there is a large number of transient situations that have also been analysed. The risk for radioactive release to the environment was considered probabilistically (PSA level 2) also for the first time for the Loviisa NPP.

Summary of the severe accident management implementation

The Loviisa severe accident program, which includes plant modifications and severe accident management procedures, was initiated in order to meet the requirements by the Finnish regulatory authority, STUK.

Fortum's approach for severe accident assessment and management for Loviisa is based on four successive levels. The first level of the approach is to ensure that severe accidents can be prevented with high probability. The quantitative targets for the overall core damage frequency (CDF) obtained from PSA level 1, are 10^{-4} /reactor year for existing plants.

The second level is to show a very low fraction of overall CDF for those classes of accident sequences which can be assumed to directly lead to a large release. Such sequences are the ones with impaired containment system function, high pressure core melt sequences and reactivity accidents leading to core damage. The class called sequences with impaired containment function consists of containment by-pass sequences (expecially, primary to secondary leakage accidents), sequences with pre-existing openings, containment isolation failures, containment pressure suppression system by-passes and sequences with induced leakage outside the containment.

On the third level of the approach, the focus is on physical phenomena capable of threatening the containment integrity. The challenge to the containment integrity due to any physical phenomena should be excluded either by excluding the phenomenon itself as physically unreasonable or by showing that the loads caused by the phenomenon are tolerable. The phenomena considered include in-vessel and ex-vessel steam explosions, hydrogen burns, direct containment heating, missiles, slow overpressurization due to steaming and generation of noncondensable gases, core-concrete interaction, recriticality of the degraded core and core debris, and temperature loadings of the containment. It is obvious that plant specific studies are needed for proper treatment of the individual phenomena. Instead of traditional PSA level 2 type of approach, in case of Loviisa, Fortum has treated the main phenomenological, Loviisa specific questions along the lines of the ROAAM (Risk Oriented Accident Analysis Methodology) approach.

After successful exclusion of the containment system and structural failures, the fourth and final level of the approach is to define the radioactive releases through containment leakages. The releases during the managed accident sequences should stay below the acceptable criteria concerning acute health effects and land contamination.

For Loviisa, the approach translates to ensuring the following top level safety functions:

- depressurization of the primary circuit,
- absence of energetic events, i.e. hydrogen burns,
- coolability and retention of molten core in the reactor vessel,
- long term containment cooling,
- ensuring subcriticality, and
- ensuring containment isolation.

The cornerstone of the SAM strategy proposed for Loviisa is the coolability of corium inside the reactor pressure vessel (RPV) through external cooling of the vessel. Since the RPV is not penetrated, all the ex-vessel phenomena such as ex-vessel steam explosions, direct containment heating and core-concrete interactions can be excluded. The only energetic phenomena remaining which could have potential to threaten the containment integrity are hydrogen burns.

In-vessel retention of corium

Some of the design features of the Loviisa Plant make it most amenable for using the concept in-vessel retention (IVR) of corium by external cooling of the RPV as the principle means of arresting the progress of a core melt accident. Such features include

- the low power density of the core,
- large water volumes both in the primary and in the secondary side,
- no penetrations in the lower head of the RPV and finally,
- ice condensers ensure a flooded cavity in most severe accident scenarios.

On the other hand, if in-vessel retention was not attempted, showing resistance to energetic steam generation and coolability of corium in the reactor cavity could be laborous for Loviisa, because of the small, water filled cavity with small floor area and tight venting paths for the steam out of the cavity.

The main focus of the thermal studies for IVR is therefore on finding out 1) the actual heat flux from the molten corium pool and 2) the critical heat fluxes at the corresponding locations on the RPV wall. Because of the relatively thick RPV wall, and because of the crust, which creates isothermal boundary conditions for the molten pool, the in-vessel and ex-vessel heat transfer phenomena can be effectively decoupled from each other.

An extensive research program was carried out by Fortum. The work included both experimental and analytical studies on heat transfer in a molten pool with volumetric heat generation and on heat transfer and flow behavior at the RPV outer surface.

Based on experiments, the IVR concept for Loviisa was finalized in April 1994. The concept includes plant modifications at four locations. The most laborous one is the modification of the lower neutron and thermal shield such that it can be lowered down in case of an accident to allow free passage of water in contact with the RPV bottom. Other two modifications include slight changes of thermal insulations and ventilation channels in order to ensure effective natural circulation of water in the channel surrounding the RPV. Finally a strainer facility will be constructed in the reactor cavity in order to screen out possible impurities from the coolant flow and thereby prevent clogging of the narrow flow paths around the RPV.

The conceptual design was submitted to STUK for approval and approval in principle was received in December 1995.

Absence of energetic events

Based on plant-specific features, the only real concern regarding potential energetic phenomena is due to hydrogen combustion events. The Loviisa reactors are equipped with ice-conderser containments, which are relatively large in size (comparable to the volume of typical large dry containments) but have a low design pressure of 0.17 MPa. The ultimate failure pressure has been estimated to be well above 0.3 MPa. An intermediate deck divides the containment in the upper (UC) and lower compartments (LC). All the nuclear steam supply system (NSSS) components are located in the lower compartment and, therefore, any release of hydrogen will be directed into the lower compartment. In order to reach the upper compartment, which is significantly larger in volume, the hydrogen and steam have to pass through the ice-condensers.

Because of the relatively low design pressure of the containment, the hydrogen burns that can create a potential threat include not only detonations, but all large-scale combustion events that are rapid enough to yield an essentially adiabatic behavior. An additional concern which is caused by the type of the containment occurs when the steam and hydrogen mixture passes through the ice-condenser. The steam will be condensed in the ice beds, which could potentially lead to very high local hydrogen concentrations.

In the early 1990's an extensive research program was initiated at Fortum to assess the reliability and adequacy of the existing igniter system. One of the focus areas in the studies was to determine the prerequisites for creating and maintaining a global convective flow loop around the containment for ensuring well mixed conditions. The global flow loop which passes from the lower compartment through an ice-condenser to the upper compartment and back to the LC through the other ice-condenser is necessary in order to bring air into the LC and thus to be able to recombine or burn hydrogen in a controlled way already in the LC. The experiments and the related numerical calculations demonstrated that the global convective loop will be created and maintained reliably provided that the ice-condenser doors will stay open.

The studies have been completed and the new hydrogen management strategy for Loviisa has been formulated. The new hydrogen management scheme concentrates on two functions: ensuring air recirculation flow paths to establish well-mixed atmosphere (opening of ice condenser doors) and effective recombination and/or controlled ignition of hydrogen. Plant modifications which are necessary include the new hydrogen recombination devices and a dedicated system for opening the ice-condenser doors.

Prevention of long term overpressurization

The studies on prevention of long term overpressurization at Loviisa started by considering the concept of filtered venting, as was done for many European NPPs after the Chernobyl accident. However, the capability of the steel shell containment to resist subatmospheric pressures is poor. If using filtered venting, it is possible that the amount of noncondensable gases after the venting is significantly less than originally, which later - after cooldown of the containment atmosphere - may lead to subatmospheric pressures and possibly collapse of the containment. Therefore, alternative solutions were sought for.

Since the concrete used in the reactor cavity of Loviisa does not contain any CO_2, the amount of noncondensable gases (except for hydrogen) generated during core-concrete interaction would be practically zero. Therefore, the overpressure protection of containment could be limited to condensing the steam produced. An obvious way of doing this is to spray the exterior of the containment steel shell. Later on, the concept of in-vessel retention was introduced to Loviisa (as discussed above), which excludes core-concrete interactions altogether and thus finally ensures that no noncondensable gases apart from hydrogen need to be considered.

The system is designed to remove the heat from the containment in a severe accident when other means of decay heat removal from the containment are not operable. Due to the ice condenser containment, the time delay from the onset of the accident to the start of the external spray system is long (18 – 36 hours). Thus the required heat removal capacity is also low, only 3 MW (fraction of decay power is still absorbed by thick concrete walls). The system is started manually when the containment pressure reaches the design pressure 1.7 bar. Autonomous operation of the system independently from plant emergency diesels is ensured with dedicated local diesel generators. The single failure criterion is applied. The active parts of the system are independent from all other containment decay heat removal systems. There are no active parts of the system inside the containment.

The both units Loviisa 1 and 2 have their own external spraying circuits and spray water storage tanks. The cooling circuit of the spraying system and the dedicated diesel generators are common for both units. The ultimate heat sink is sea water.

The design calculations were carried out with Fortum's own simplified containment thermal-hydraulic code PREDEC. The PREDEC calculations were supported by experiments carried out at the HDR containment (tests E11.2 and E11.4) in Germany. These experiments were aimed at studying the hydrogen distribution during stratified conditions inside the containment. The main result from the HDR experiments was that the PREDEC code could be used for the design calculations of the external spray system.

The influence of the external spray system was further studied experimentally using the VICTORIA facility.

Primary circuit depressurization

The primary depressurization is an interface action between the preventive and mitigative parts of SAM. If the primary feed function is operable, the depressurization may prevent the core melt. If not, it sets in motion the mitigative actions and measures to protect the containment integrity and mitigate large releases.

Manual depressurization capability has been designed and implemented through motor-operated relief valves. Depressurization capacity will be sufficient for bleed&feed operation with high-pressure pumps, and for reducing the primary pressure before the molten corium degrades the reactor vessel strength. Depressurization is to be initiated from indications of superheated temperatures at core exit thermocouples. The depressurization valves were installed at the same time with the replacement of the existing pressurizer safety valves in 1996.

Implementation

The SAM-strategy described in the previous chapters has lead to a number of hardware changes at the plant as well as to new severe accident guidelines and procedures.

The containment external spray was implemented at the two units in 1990 and 1991. Primary system depressurization capability was installed at both units in 1996. The major backfittings related to external coolability of the reactor pressure vessel and to opening the ice-condenser doors are, for the most part, implemented at Loviisa 1 in 2000 and at Loviisa 2 in 2002. Test samples of the new hydrogen recombination devices have been aged and tested in plant conditions and the devices will be installed in 2002. In addition to the mechanical equipment, the implementation includes also a new, dedicated, limited scope instrumentation and control system for the SAM-systems, a dedicated AC-power system and a separate SAM control room which is common to both units.

The severe accidents guidance for the operating crew consists of SAM-procedures for the operators and of a so-called Severe Accident Handbook for the Technical Support Team. The SAM procedures are entered after a prolonged uncovery of the reactor core indicated by highly superheated core exit temperatures. The procedures are symptom oriented and their main objective is the protection of containment integrity through ensuring the top level severe accident safety functions. The most important operator actions after the core uncovery are the ensuring of containment isolation, primary circuit depressurization, opening of ice-condenser doors in order to ensure mixing of hydrogen, lowering of the neutron shield of the lower part of the RPV and, in the long term, starting of the containment external spray. The Severe Accident Handbook contains background material for the procedures and it should facilitate the support team in gaining understanding of the progress of the accident and of potential means of recovery.