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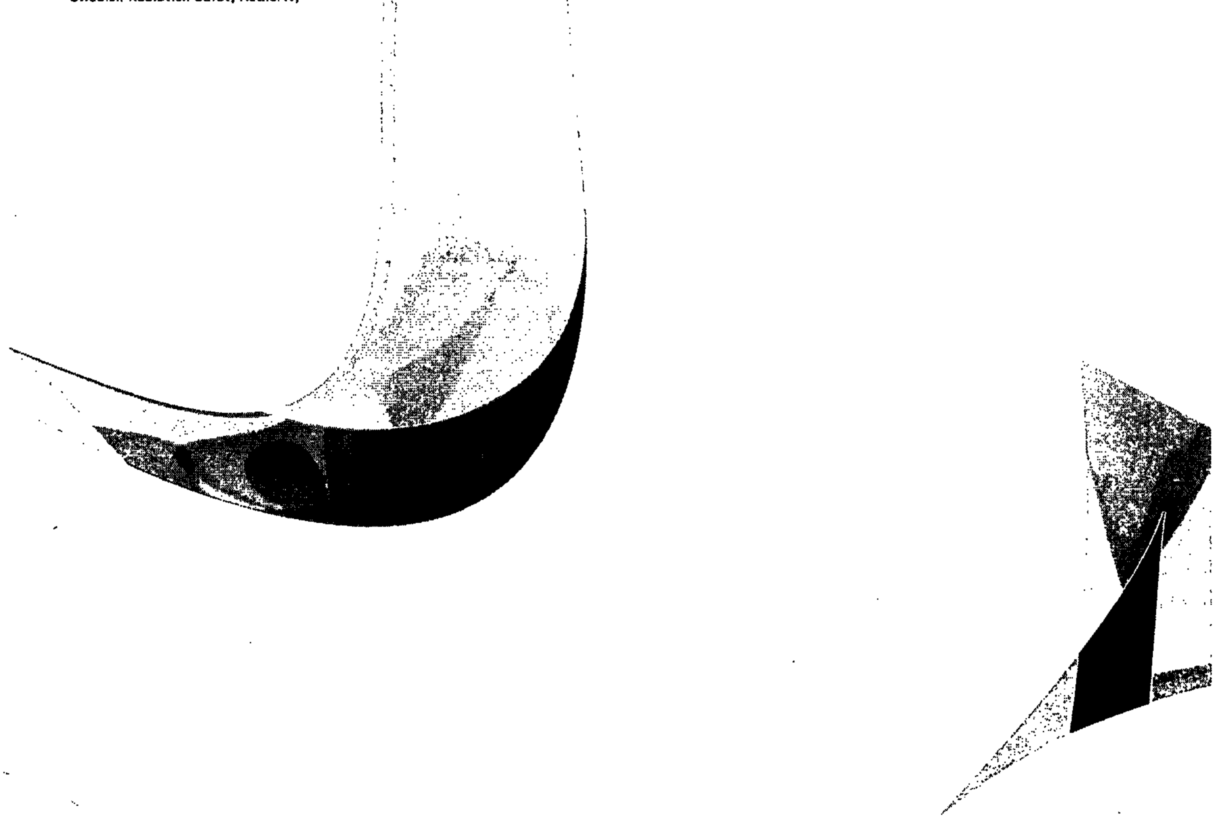

European stress tests
for nuclear power plants

The Swedish National Report

December 29, 2011

ABSTRACT

On 11 March 2011, the Tohoku region in north Honshu, Japan, suffered a severe earthquake with an ensuing tsunami and an accident at the Fukushima Dai-ichi nuclear power plant. Due to the accident the Council of the European Union declared in late March that Member States were prepared to begin reviewing safety at nuclear facilities in the European Union by means of a comprehensive assessment of risk and safety ('stress testing').

On 25 May, SSM ordered the licensees of the nuclear power plants to conduct renewed analyses of the facilities' resilience against different kinds of natural phenomena. They were also to analyse how the facilities would be capable of dealing with a prolonged loss of lectrical power, regardless of cause.

On 31 October, the licensees reported on their stress tests to SSM. After reviewing these reports, SSM produced a summary stress test report, which was submitted to the Government on the 15 December. The present report is the national report on Swedish stress tests of nuclear power plants. The report will be submit to the European Commission no later than 31 December.

Based on the review SSM has drawn the conclusion that the stress tests carried out by Swedish licensees are largely performed in accordance with the specification resolved within the European Union. The scope and depth of these analyses and assessments are essentially in accordance with ENSREG's definition of "a comprehensive assessment of risk and safety". The stress tests show that Swedish facilities are robust, but the tests also identify a number of opportunities to further strengthen the facilities' robustness.

SSM will order the respective licensees to present an action plan for dealing with the results from the stress tests. The Authority will then examine the plans and adopt a standpoint on proposed measures as well as check that the necessary safety improvements are made.

In a number of cases, the stress tests indicate deficiencies in relation to, or alternatively, deviations from applicable requirements imposed on safety analysis and on design and construction. In some of these cases, SSM may need to precede a licensee's presentation of an action plan and order this licensee to take measures so that the facilities fulfill the requirements.

The Authority nevertheless assesses that none of the deficiencies currently identified or the measures needed are of such a nature that the continued operation of the facilities needs to be put into question.

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1.1. Background

On 11 March 2011, the Tohoku region in north Honshu, Japan, suffered a severe earthquake with an ensuing tsunami. The accident at the Fukushima Dai-ichi nuclear power plant has led to many countries around the world launching investigations or taking various measures to review the level of safety at nuclear power plants. On 22 March 2011, SSM pointed out in a written communication to the licensees the importance of immediately launching work to identify lessons learned from the situation with the aim of assessing any further radiation safety measures that might be necessary at Swedish nuclear power plants as well as at the facility for storage of spent nuclear fuel.

Following an extraordinary meeting, in late March the Council of the European Union declared that Member States were prepared to begin reviewing safety at nuclear facilities in the European Union by means of a comprehensive assessment of risk and safety ('stress testing'). The Council was of the view that the criteria should be defined on the basis of lessons learnd from the situation in Japan so that the assessments could be conducted as soon as possible. The Council urged the European Nuclear Safety Regulatory Group (ENSREG) and the Commission to clarify these criteria through the participation of Member States and expert organisations such as the Western European Nuclear Regulators' Association (WENRA).

Since then, WENRA has drawn up a proposed specification for the scope and orientation of risk and safety assessment ENSREG processed this proposal and produced a specification for the stress tests.

On 12 May, the Swedish Government decided on an assignment for SSM that included submitting a comprehensive report on the stress tests of the relevant Swedish nuclear facilities to be conducted on the basis of the European Union-wide requirements. The material must be presented to the Swedish Government no later than 31 October 2012, compiled together with other information as part of the Government assignment on long-term nuclear safety dated 8 April 2010 (M2010/2046/Mk).

On 25 May, SSM ordered the licensees of the nuclear power plants, as well as the interim storage facility for spent nuclear fuel (Clab), to conduct renewed analyses of the facilities' resilience against different kinds of natural phenomena. They were also to analyse how the facilities would be capable of dealing with a prolonged loss of electrical power, regardless of cause. It was stated in the motivation for the decision that the specific details concerning the scope and performance of these renewed analyses and safety evaluations were stipulated by the specification for the 'stress tests' as agreed between European nuclear safety regulatory authorities and the European Commission within the framework of ENSREG. ENSREG's specification does not include analyses of fuel ponds outside the facilities, and tha analysis of Clab is therefore not part of present report.

On 15 August, each respective licensee submitted an interim report on its analysis work to SSM, and on 15 September, SSM submitted a national progress report entitled "European Stress Tests for Nuclear Power Plants, National Progress Report" (SSM 11-1089) to the European Commission. On 31 October, the licensees reported on their stress tests to SSM. After reviewing these reports, SSM produced a summary stress test report, which were submitted to the Government the 15 December. The present report is the national report on Swedish stress tests of nuclear power plants, and will be submit to the European Commission no later than 31 December.

1.2. Method

The Swedish stress test has followed the ENSREG specification. Each licensee has performed an assessment of risk and safety of their nuclear power plants and SSM has reviewed these assessments. The present report, issued by SSM, is based on the licensee's assessments and the results of the SSM review.

The stress tests relate to an assessment of whether the facilities' safety analyses and design bases are still valid in light of the lessons learnd from the accident in Fukushima or whether different kinds of measures are needed so that safety requirements are fulfilled, and relate to an assessment of the facilities' safety margins beyond the design bases. The assumptions for assessing the safety margins outside the design assumptions constitute more unlikely events where the safety functions have degraded to a greater degree than in the existing safety analysis report (SAR). This is to enable 'stressing' of the analysis assumptions to the level where serious core damage occurs. This method gives opportunities for identifying measures that could further increase the level of safety and resilience on the part of the facilities.

According to the specification from ENSREG, the analyses are to encompass three types of initiating events: flooding, earthquakes and extreme weather conditions. Besides these, two postulated events are to be studied: loss of electrical power, a loss of ultimate heat sink in addition to a combination of these. Finally, this also includes analysing the potential for emergency response management in a situation where several reactors are affected simultaneously and the circumstances are complicated by extreme weather conditions.

Due to the fact that the structure of the documentation of the three licensees assessments are not the same, the structure of the present report are not consistent in all parts.

1.3. The SSM regulatory approach

The SSM regulatory approach is rather process oriented. This means that the regulations are rather general with a focus on the required license processes and the outcome of these processes. There are no details on how these processes have to be performed. Even the regulations on design of nuclear power reactors are rather general, focusing on the principles of the design and what the safety functions have to achieve.

In addition to the regulation, general advice on the interpretation of most of the safety regulations is issued. The general advice is not legally binding per se, but cannot be ignored by the licensee without risking sanctions by the regulatory body. Measures should be taken according to the general advice or, alternatively, methods justified to be equal from the safety point of view should be implemented.

The SSM regulations concerning Safety in Nuclear Facilities, SSMFS 2008:1, are applicable, in a graded way, on all licensed nuclear facilities. The regulations aim at specifying measures needed for preventing and mitigating radiological accidents, preventing illegal handling of nuclear material and nuclear waste and for conducting an efficient supervision. The regulations are dealing with:

- Application of multiple barriers and defence-in-depth
- Handling of detected deficiencies in barriers and the defence-in-depth
- Organisation, management and control of safety significant activities
- Actions and resources for maintaining and development of safety
- Physical protection and emergency preparedness
- Basic design principles
- Assessment, review and reporting of safety
- Operations of the facility
- On-site management of nuclear materials and waste

- Reporting to SSM of deficiencies, incidents and accidents
- Documentation and archiving of safety documentation
- Final closure and decommissioning

The SSM regulations concerning Design and Construction of Nuclear Power Plants, SSMFS 2017:17, was issued the 1 of January 2005. The transitional rules to the regulations stipulate that measures to comply with certain paragraphs shall be implemented at the latest at time points decided by SSM. The reason for this is that the licensees must be given time to investigate in depth, specify, procure, install, test, and safety review the back fitting measures needed to comply with the regulations. A description of the measures implemented and the measures which still have to be implemented is found in section 1.9. The main part of the measures are implemented, but there are still some big modernization programmes to be performed. The modernization of Oskarshamn 2 is planned to be performed 2012 and there are still quite a lot of measures to be implemented in the Ringhals PWR units.

These requirements, with general advice, contain specific requirements for nuclear power reactors on design principles and the implementation of the defence-in-depth concept, withstanding of failures and other internal and external events, withstanding of environmental conditions, requirements on the main and the emergency control room, safety classification, event classification, requirements on the design and operation of the reactor.

The application of the 14 §, SSMFS 2008:17, is of special interest in the stress tests. The paragraph reads:

> "The nuclear reactor shall be dimensioned to withstand natural phenomena and other events that arise outside or inside the facility and which can lead to a nuclear accident. In case of such natural phenomena and events, dimensioning values shall be established. Natural phenomena and events with such rapid sequences that there is no time to implement protective measures when they occur, shall also be assigned to an event class. For each type of natural phenomenon that can lead to a nuclear accident, an established action plan shall exist for the situations where the dimensioning values run the risk of being exceeded."

Parts of the general advice to the 14 § reads:

> "Examples of natural phenomena that should be taken into account are:
>
> - extreme winds,
> - extreme precipitation ,
> - extreme icing,
> - extreme temperature,
> - extreme sea waves,
> - extreme seaweed growth or other biological conditions that can effect the cooling water intake,
> - extreme water level,
> - earthquake"

Besides the regulation SSMFS 2008:1 and SSMFS 2008:17, there are also regulations on preassure vessels, mechanical equipments, competence and education for operators, security and radiation protection.

1.4. Brief description of the Swedish sites characteristics

1.4.A. Forsmark

On the 15 of January 1973, Statens Vattenfallsverk (SV) (now Vattenfall AB) and Mellansvensk Kraftgrupp Aktiebolag (MKG) established Forsmarks Kraftgrupp Aktiebolag (Forsmark) to jointly build and operate a nuclear power plant in Forsmark, Östhammar. The license holder is Forsmark Kraftgrupp AB which is a subsidiary of Vattenfall AB.

The Forsmark power plant is situated on the Swedish east coast about 4 km north of Forsmarks Bruk in Östhammar Municipality in Uppsala County.

The distance from the capital, Stockholm, is 138 km. Other larger cities in the vicinity are Uppsala (73 km) and Gävle (75 km). In general, the immediate surroundings are considered sparsely populated, but the distance to large consumers of electricity is relatively short. This together with good bedrock, access to cooling water and labour-market constituted the main motives for choosing this location.

The Forsmark plant has three BWR reactors designed by the former Swedish company ASEA-ATOM (now Westinghouse Electric). The power plant also has a gas turbine unit with a 40 MW capacity.

Forsmark is connected to the 400 kV national grid (three lines) and to the 70 kV regional grid (two lines). Forsmark 1 and 2 have a joint 400 kV switch yard, while Forsmark 3 has its own switch yard. The 70 kV switch yard is shared between all three reactors. The connections to the 400 kV grid and 70 kV grid are independent of each other.

The design basis sea level is based on actual measurements of the sea level between 1895 and 1975 at Forsmark. To this, a margin was added. Additionally, a small land rise has increased the margins.

The seismicity is defined by the average Fennoscandian seismicity function. The seismic activity in the region is low. In addition site specific conditions are taken under consideration. The hardness variation in the rock below the plant is significantly less than the assumed variation for a typical hard rock site. Hence the site specific spectra can be obtained by multiplying the general Swedish hard rock spectra by 0.85.

Significant tsunamis have not occurred in the Baltic Sea. Furthermore, the site is in most directions protected by an archipelago. Globally, tsunamis have only once been registered in an inland sea like the Baltic Sea. This was after a major earthquake in Turkey 1999 with a magnitude of 7.4 on the Richter scale. The level of the tide in the Black Sea was then 2.5 m. Thus, tsunamis are regarded to be within the envelope of the design limitation set by the sea water.

Cooling water is brought in from the archipelago southeast of the power station. It is led through natural ponds and excavated channels to the intake buildings. There are no rivers or dams in the vicinity that can cause flooding.

1.4.B. Oskarshamn

The Oskarshamns site is situated on the Simpevarp peninsula in the north part of Kalmar sound of the Baltic Sea, about 30 km north of Oskarshamn in Oskarshamn's Municipality in Kalmar County.

The distance from the capital, Stockholm, is north 330 km. Malmoe and Gothenburg are located about the same distance south and east respectively from the site. Other large cities in the vicinity are Kalmar (95 km), Linköping and Norrköping (170 km respectively). The nearest major airport is Kalmar Airport.

The area around the nuclear power plant is sparsely populated. Only about 150 people live closer to the plant than 5 kilometers. Within a radius of 20 km, the corresponding figure is just under 1400 people.

The site has three BWR reactors. The site is operated by OKG Aktiebolag, who is the license holder.

Like all other Swedish nuclear power plants, the Oskarshamn plant site is located on stable bedrock and along the coastline in order to be able to use the sea as a heat sink.

The location along the western shore of the Baltic Sea causes the following effects of weather, wind and other external environmental conditions.

The wind situation is affected by the area's coastal location where the sea breeze and other effects can cause local wind variations. The average wind speed is 4.2 m/s. The estimated extreme wind speed, with a return period of 100 years, is 24.7 m/s at 10 meters above the ground (and gusts to 39. m/s).

The average temperature per month (average for the period from 1961 to 1990) is 17.0 °C for the warmest month (July) and -0.2 °C for the coldest (January and February). The highest temperatures recorded in Kalmar and Västervik during the period from 1876 to 1998 are 35.2 °C and 34.0 °C, respectively. The lowest temperatures recorded at those locations are -31.0 °C and -33.1 °C, respectively.

Annual precipitation is low (about 550 mm), because the landscape is in the rain shadow behind the highlands of Småland. Approximately 18% falls as snow. Most precipitation falls on average during the months of July through September. Minimum quantities generally fall during February and March.

Maximum rainfall intensity varies strongly with the extent of weather conditions at the time. In connection with short thunderstorms, the rainfall intensity can be very high, estimated to be at least 5 mm/min, whereas during a day-long rain the intensity is rarely higher than 5 mm/hour.

The maximum daily precipitation measured in Sweden by SMHI is 198 mm (Norrbotten 1997).

The following table provides a summary of the results of an analysis of extreme precipitation days for various return periods for the sites of nuclear power plants in Sweden (source: SMHI).

Table 1.1 - Extreme precipitation days for various return periods

Site Location	Extreme daily precipitation (mm)		
	100 year return period	1000 year return period	10 000 year return period
Ringhals	70-100	90-105	115-130
Simpevarp	90-110	120-140	150-180
Forsmark	100-115	145-155	190-200

The variation of the water level in the Baltic Sea is controlled by the inflow and outflow through the Öresund and Danish straits as well as the inflow from the rivers. The volume of flows through straits is controlled by air pressure variations and the associated wind conditions. Strong winds from the west to the northwest cause the strongest inflows while winds from the east to the northeast cause corresponding outflows. Persistent high pressure over the Baltic Sea generally causes low sea water levels. The sea water level locally is also affected by the wind. In the Baltic, the influence of tides is very small, at most about 0.1 m.

SMHI has registered the water level in Oskarshamn since 1975, and it represents quite well the water level outside the Oskarshamn plant site. The highest water level was measured in

January 1983 at +100 cm above mean sea water level. The lowest water level was measured in November 1979 at 75 cm below mean sea water level.

In the coastal area at Simpevarp, icing normally occurs in early February and not before the beginning of January. Ice usually occurs in late March and by the end of April. Ice thickness is normally between 15 and 20 cm and a maximum of 45 cm. Hamnefjärden is always ice-free thanks to the cooling water discharge.

Good knowledge of the bedrock in Simpevarp area has been obtained in connection with the expansion of various mountain sites such as cooling tunnels for the reactors, various caverns (Clab, BFA) and the 3600 m long Äspö tunnel down to 460 m depth. The bedrock in the plant area is dominated by two categories of rocks, granite and volcanic rocks.

Scandinavia is characterized as a region of low seismic activity. Southeast Sweden does not deviate from this. An abundant fracturing of the bedrock in southern Sweden (which includes the Oskarshamn area) causes the displacements in the bedrock to occur mainly by creep, and not by sudden shifts that can cause earthquakes.

The phenomenon tsunami is not relevant because of the site location at the west coast of the Baltic Sea. The area is seismically calm and the Baltic Sea is too shallow for a tsunami to be created. Therefore no tsunami is assumed when evaluating margins.

About 100 m outside the sea water intake the small islands Tallskär and Gloholmen together with the harbor pier constitute a protective reef against sea waves. Further the buildings are situated with distance from the shoreline and are erected on rock with a ground level of several meters above the normal average sea water level in the Baltic Sea. Thereby the plant is insensitive to wave surge and extreme sea waves are therefore not evaluated.

1.4.C. Ringhals

The Ringhals plant is situated along the west coast of Sweden on the Värö peninsula within the Varberg municipality in the province of Halland. Contiguous cities are Varberg 25 km south of and Gothenburg 60 km north of Ringhals.

The site has four reactors, one BWR and three PWR's. The site is operated by Ringhals AB, who is the license holder.

The site is situated by the sea, and is consequently exposed to events originating from the sea. In the vicinity of the site, there are no dams or rivers constituting potential sources of flooding. The single source of flooding with the potential to cause a sequence of events resulting in fuel damage is a sea-level rise.

The assumed highest flooding level is a sea level rise of 2,65 m.

In the seas surrounding Sweden no tsunami has been registered. Even if tsunami initiated events should occur, the impact would be small since the depth of the surrounding sea is limited. The average depth in the Baltic is 60 m and is assumed to be similar in the Kattegatt, resulting in a wave velocity of 25 m/s. This concludes that the waves will not generate any larger waves with the potential to cause any major damage.

Water level variations due to tide are usually insignificant, but may add up to an extra 30 cm to the water level variations due to atmospheric pressure and wind load.

In Scandinavia seismic activity is generally low. The region is generally considered as being seismically stable. Only a few incidents have been registered in historic time, which might have damaged an industrial plant of today. The risk of a nuclear accident in Sweden, caused by an Earthquake, may thus be considered to be low.

The design basis earthquake is an earthquake with the occurrence frequency of 10^{-5}/year.

The probability of an airplane incident in the vicinity of the Ringhals site is estimated to be very low, $2*10^{-8}$/year.

1.5. Main characteristics of the units

1.5.A. Forsmark

Forsmark 1 and 2 are light water reactors of boiling water type BWR69. They are of Swedish design by the former ASEA-ATOM (now Westinghouse Electric). The reactor produces saturated steam with a pressure of 7 MPa for direct use in the steam turbine.

A general description of the main characteristic for Forsmark 1 and 2 is given in Table 1.2.

Table 1.2 - General description of the main characteristic for Forsmark 1 and 2.

Unit	Date of first criticality	Reactor Original power level		Reactor Current power level	
	DD-MM-YYYY	Thermal	Electrical	Thermal	Electrical
F 1	23-04-1980	2711	900	2928	984
F 2	06-11-1980	2711	900	2928	996

The reactor vessel is made of low-alloy steel with a cladding of stainless steel on the inside. The design pressure is 8.5 MPa and the design temperature is 300°C. Pressure and temperature during operation are 7 MPa and 286°C. The reactor core consists of 676 vertical fuel assemblies. Groups of four fuel assemblies surround a cross-shaped control rod and constitute a super cell. The fuel assemblies consist of approximately 100 fuel rods in a 10x10 array and with internal water channels, surrounded by a fuel box, which constitutes a cooling channel.

Forsmark 1 and 2 have two turbines each. The turbines were designed and manufactured by the former Swedish company ASEA-STAL (now Alstom). Each turbine consists of a high-pressure turbine and three low-pressure turbines, integrated into a shaft assembly together with the two-pole generator. Fresh steam travels first through the high-pressure turbine and then on via the moisture separators and re-heaters in parallel through the low-pressure turbines. All turbines are axial dual flow turbines, where steam is taken in at the centre and goes out through the ends. The shaft assembly rotates at 3000 rpm.

Forsmark 3 is also a light water reactor of boiling water type BWR75 designed by ASEA-ATOM (now Westinghouse Electric). In principle, Forsmark 3 is similar to Forsmark 1 and 2, and is slightly newer and has a larger thermal power of 3300 MW. The reactor produces saturated steam with a pressure of 7 MPa for direct use in the steam turbine. Since the reactor has internal circulation pumps and fine motion control rods it is considered to be an advanced boiling water design of generation III.

A general description of the main characteristic for Forsmark 3 is given in Table 1.3.

Table 1.3 - General description of the main characteristic for Forsmark 3

Unit	Date of first criticality	Reactor Original power level		Reactor Current power level	
	DD-MM-YYYY	Thermal	Electrical	Thermal	Electrical
F 3	28-10-1984	3020	1100	3300	1170

The reactor core consists of 700 vertical fuel assemblies. Design pressure main reactor design parameters are essentially the same as Forsmark 1 and 2.

Forsmark 3 has just one turbine. The turbine was designed and manufactured by the former Swedish company ASEA-STAL (now Alstom). The turbine consists of a high-pressure turbine and three low-pressure turbines, integrated into a shaft assembly together with the four-pole generator. The shaft assembly rotates at 1500 rpm.

1.5.B. Oskarshamn

On the Oskarshamn site there are three nuclear power reactors with data as shown in Table 1.4.

Table 1.4 – Data for the reactors at the Oskarshamn NPPs.

Unit	Type of reactor	Thermal power MWt	Gross electrical output MWe	First criticality	Maximum number of fuel assemblies in storage at unit
O1	BWR	1375	492	12 Dec 1970	923
O2	BWR	1800	661	06 March 1974	1018
O3	BWR	3900	1450	29 Dec 1984	1038

All three reactors are BWR's of Asea-Atom design with pressure suppression type containments (pre-stressed concrete containments with steel liners).

The reactors have all been designed and built by Asea Atom, but belong to three different design generations. Recently, reactors Oskarshamn 1 and Oskarshamn 3 have been modernized and a major modernization project is in progress at Oskarshamn 2. The common denominator is that all reactors have safety measures implemented gradually as new knowledge and requirements to come.

On the site there is also a central interim storage for spent fuel for the whole Swedish nuclear programme. The storage is owned and operated by Svensk Kärnbränslehantering AB, SKB. It is an underground storage facility with fuel pools in granite vaults about 30 m underground.

The OKG reactor 1 and 2 are built together at the southern part of the Simpevarp peninsula. The height of the reactor building of reactor 1 is 57 m above the ground and the height of the ventilation stack of reactor 1 is 76 m above ground and for reactor 2 it is 116 m above ground. The ventilation stacks are connected to each reactor building. The ground level for reactor 1 and 2 is 6 m above normal sea water level.

The reactor 3 is situated separately from reactor 1 and 2 on the north eastern part of the peninsula. The height of the reactor building rises 60 m above ground and the ventilation stack is 103 m above ground level and connected to the reactor building. The ground level for reactor 3 is 3 m above normal sea water level.

On the site, within the fenced off area close to Oskarshamn 2, there are two gas turbine units with 40 MWe capacity each, diesel oil storage tanks (max capacity 12 000 m^3) and workshop buildings necessary for the most common repair and maintenance work. In the fenced off area there are also interim storage for radioactive parts in an underground vault.

The gas turbines serve primarily reactor 2 but also reactor 1 is automatically connected. Reactor 3 could also be connected, but manual operations are needed. Each gas turbine unit has two gas generators and two turbines connected to a common generator with the capacity of 40 MWe.

Outside the fenced area there are housing areas for employees working at the power plant temporarily, the old Simpevarp village with information expo and a small hotel, hydrogen plant, outer switchyard, fire station, meteorology mast, sanitary water plant and a harbour.

Raw water is supplied from the lake Götemaren or as a back up from a water pond (Söråmagasinet) fed from the creek Sörå with outlet into the pond.

Power is supplied to the national grid by four 400 kV transmission lines (towards Kimstad, Glan, Nybro and Alvesta) and four 130 kV transmission lines (towards Fårhult, Mariannelund and two towards Oskarshamn south).

Cooling water is taken from the Baltic Sea. The cooling water to reactor 1 is taken from a surface intake while reactor 2 and 3 are cooled by deep water intakes through two rock tunnels from about 500 m from the shore line. One tunnel to reactor 2 and one tunnel to reactor 3. In the future (estimated 2013) the reactor 1 will also be connected to the reactor 2 tunnel. The reactor 2 can also take water from the same surface intake as reactor 1 when needed.

Hydrogen explosions cannot occur in the containment due to the fact that the containment is nitrogen-filled. Any oxygen that is present or produced is consumed in the oxidation process. If the containment must be vented, this is done through the filtered venting system, se section 1.8.

1.5.C. Ringhals

The protective functions included in the plant design in order to maintain the barrier integrity are characterized of design according to the single failure criterion, i.e. one single failure in the protective function does not jeopardize the purpose of the function.

The plants were originally designed with two main trains of safety functions. The supporting power system was divided into four sub trains, each one with a dedicated emergency diesel generator. Two sub trains are supporting each main train.

The two train design has during the years been reinforced. Additional reactor trip systems have been installed on all reactors. On Ringhals 1 an entire diverse plant section (DPS) has been installed in addition to the original plant section (OPS). Measures to strengthen the physical and functional separation of the trains have been taken.

The parallel trains may in some cases perform the intended functions by different methods such as steam driven and motor driven pumps, diversification.

The plant is designed to protect the barriers with automatically taken measures or by permitting reasonable time for consideration of manual action. The latter is often called "grace time" or "time for reflection".

All reactors are equipped with steam driven systems to provide core cooling capabilities, either directly to the reactor vessel (BWR) or via the steam generators (PWR).

The power supply to the reactor during normal operation consists of two alternative sources; the national 400 kV grid and the national 130 kV grid. At accident conditions, the power is supplied by four dedicated diesel generators and gas turbine plant common for the site. Ringhals 1 has in addition to this another two air cooled diesel generators installed within the diversified plant section. The reactor is also equipped with house load operation feature, according to the national grid requirements. Furthermore, an additional full capacity mobile diesel generator is available at the site. The power supply to the filtered venting system has independent battery capacity and a dedicated mobile generator set.

The spent fuel is stored in the spent fuel pools at the reactor for an average time of one year, prior to transportation to the national spent fuel storage facility.

Ringhals 1 is an ASEA-Atom design BWR with a rated thermal power of 2540 MW. The first criticality was achieved by august 20, 1973.

Ringhals 2 is a Westinghouse design three loop PWR with a rated thermal power of 2660 MW. The first criticality was achieved by June 19, 1974.

Ringhals 3 is a Westinghouse design three loop PWR with a rated thermal power of 3151 MW. The first criticality was achieved by July 29, 1980.

Ringhals 4 is a Westinghouse design three loop PWR with a rated thermal power of 2783 MW. The first criticality was achieved by May 19, 1982.

1.6. Description of the main safety systems

This section describes some of the safety systems. Other systems and details, like the hook-up points, are given in chapter 2-6 as part of the analyses and description of the assessments. The information is in some parts limited due to the security confidentiality.

1.6.A. Forsmark

1.6.A.1. Basic design and safety principles

In order to fulfil the single failure criterion without jeopardising safety or operation, all three reactors are divided into four trains (4 x 50 %). Safety functions can be maintained if at least two trains are available. The four-train design enables the reactor to fulfil the single failure criterion even if one division is out of service.

The reactor containment is designed according to the pressure suppression principle. In the event of leakage in the primary system, steam is led down into the condensation pool located in the containment. Here, it is cooled and condensed. The pressure between the dry-well and the condensation pool is equalised. In addition, the sprinkler system for the reactor containment sprinkles and condensate the steam in the dry-well and cools the condensation pool through a heat removal system. In case of a beyond design event, the same functions are used to fill water in the dry-well below the reactor vessel in the containment (wet-cavity solution). The containment is a pre-stressed concrete containment.

The reactors are also protected through physical separation between vital components and positioning of safety related components in different fire cells, which limits the consequences of a fault or a fire. For example, there are four separated areas for safety equipment in the reactor building. Each of these is its own fire cell and houses vital components, such as valves, pumps and heat exchangers for the emergency core cooling systems and for the reactor containment sprinkler system. The physical separation is further improved in reactor 3, which is of a later design. The emergency diesel generators are placed in separated buildings on each side of the reactor building.

The safety functions Reactivity Control, Emergency Core Cooling, Reactor Coolant Pressure Boundary (RCPB) over-pressurization protection and Residual Heat Removal and Containment Isolation include functional diversification to protect against the effects of Common Cause Failure (CCF) events. This also includes diversified instrumentation and control (I&C) but not electrical power supply.

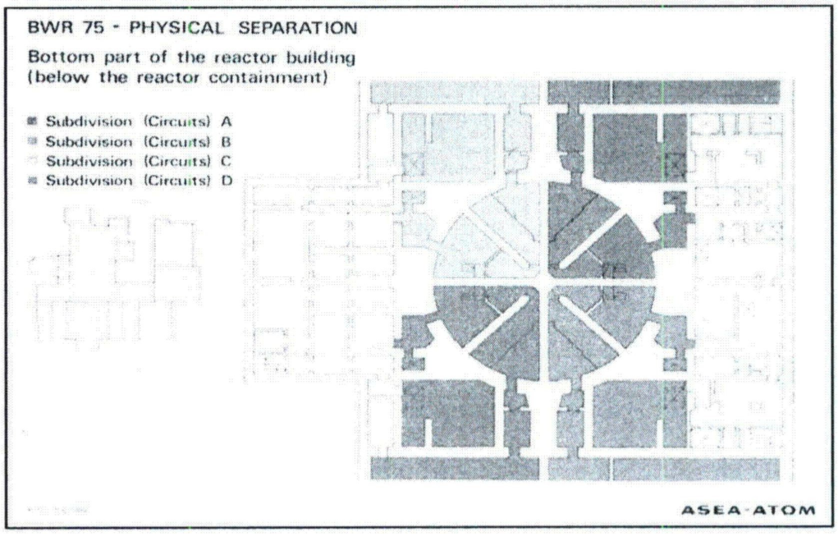

Figure 1.1 - General plan lay-out showing the physical separation of Forsmark 3

The safety function reactivity control is fail-safe. Emergency Core Cooling and Residual Heat Removal are both dependent of electrical power. The over pressure protection function is independent of electrical power and I&C. The Containment Isolation function is not dependent of electrical power.

1.6.A.2. General system description

The following system descriptions are specific to Forsmark 1 and 2. There may be minor deviations for Forsmark 3.

The main recirculation system consists of eight internal main recirculation pumps. Forced circulation and control rods are used as two different provisions to control reactivity and subsequently the thermal power. The pump motors are connected to the pump impellers. The pump shafts penetrate the bottom of the reactor vessel.

The design with internal pumps eliminates the need for large diameter piping connected to the reactor vessel below the top of the core. This eliminates all large pipe breaks below top of core.

The Pressure Relief Function protects the reactor vessel from high pressure to ensure the vessel integrity. For Forsmark 1 and 2 the system consist of 10 solenoid and pilot operated safety pressure relief valves (SRV), two pilot operated SRV, two operating control valves in series with two pilot operated SRV and four motor operated safety valves qualified for discharging both steam and water. All valves discharge steam to the condensation pool. The difference for Forsmark 3 is that there are 16 SRV which are both pilot and solenoid operated.

The discharge capacity is sufficient to protect the primary system from exceeding design pressure during design base events. The system is originally designed for 10 % overpressure at Anticipated Transient Without Scram (ATWS) and complete steam blockage with delayed scram signal.

The purpose of the four steam and water qualified valves is in addition to discharging steam also to enable circulation of water between the condensation pool and the reactor vessel (feed and bleed). The valves also serves as a diverse alternative for pressure relief and to eliminate the risk of high pressure melt through during a severe accident.

The Emergency Core Cooling Function (ECCF) has a high pressure and a low pressure system, each with four separated trains. The systems either utilises water from the condensation pool or water from an external source. The function is capable of cooling the reactor at any pipe brake. Neither the concept of Break Preclusion (BP) nor Leak-Before-Break (LBB) is used.

The Residual Heat Removal Function (RHRF) consists of four separated trains that cools the containment. In the event of pipe break or severe accidents a sprinkler part of the system is used.

Two trains of heat removal and reactor cleaning system is a diverse alternative to the primary function. During refueling the system is also used for cooling the fuel pools together with the fuel in the reactor vessel.

The Hydraulic Shut-down System consists of 18 control rod groups. Each group consists of 8, 9 or 10 control rods. Each group also includes of a nitrogen gas tank, a water tank and an actuating valve. In case of a reactor scram, the actuating valve opens and all control rods are automatically injected to the reactor. If the hydraulic scram should fail, the control rods are automatically inserted using the motor driven part of the Fine Motions Control Rod Drive (FMCRD).

The Automatic Borating System (ABS) is a diversified alternative for the Reactivity Control function in case of ATWS, with stuck control rods. The system consists of two redundant trains equipped by piston pumps to inject Enriched Boron Poison to the reactor at high pressure.

The Intermediate Cooling System (ICS) is designed to eliminate the need to use sea water in the reactor building. The purpose for this is twofold: To limit the consequence (leakage of active substances) in case of heat exchanger tube break. The other purpose is to prevent corrosion effects from salt water leakage. The systems have four independent cooling trains and the same degree of redundancy as the primary cooling systems.

The Filtered Containment Venting Function (FCVF) protects the containment from overpressure in case of severe accidents and pipe breaks with loss of the pressure suppression function.

The lower dry-well is automatically flooded in case of a severe accident to ensure coolability of the core if a reactor vessel melt-through should occur. At a later stage of the scenario, the containment is water filled by an external source to prevent high temperatures.

The function is monitored by battery backed up I&C (E- and F- division) which is separated from the primary safety systems (A-, B-, C- and D-division).

All the isolation valves in the inlet piping systems are pneumatically operated. These valves can be operated from the control room. If manual pressure relief should be required the valves can be operated locally. For further information, se section 1.8.

1.6.B. Oskarshamn

In addition to the description of the main safety systems below, there are also relevant information about the reactors in Oskarshamn in section 1.2.

1.6.B.1.Oskarshamn 1

The reactor core consists of 448 fuel assemblies and about 20% of them are replaced at the yearly outage. The power is controlled by 112 fine motion control rods and by the main recirculation flow (four speed controlled external circulation pumps).

The containment has a design pressure of 450 kPa and is inerted by nitrogen. Located to the upper part of the containment is the connection to the Multi Venturi Scrubber System (MVSS).

The turbine plant consists of a radial high pressure turbine with two counter rotating shafts. On each shaft there are one simple and two double axial low pressure turbines. On each shaft there is a generator with a water cooled stator and a hydrogen cooled rotor. The generator is connected to the 130 kV grid.

The electrical power system is divided in two separated parts. When the power plant is out of operation the plant is connected to the 130 kV grid. The safety classified electrical systems is divided in four subs A-D and each sub has its own emergency diesel generator. Oskarshamn 1 can also be feed from two gas turbines common with Oskarshamn 2.

As part of the modernization project O1 MOD (1994-2002), among others things, the following safety measures were performed: new Emergency Control building with two new diesels and new electrical installation. In addition, the control rooms and control systems were modernized. This was done in order to achieve higher safety and to obtain modern equipment and a modern design. Seismic qualification of safety classified equipment was performed in connection with the project.

1.6.B.2. Oskarshamn 2

Oskarshamn 2 went into commercial operation in 1975. It is a boiling water reactor of Asea-Atom design.

The reactor core consists of 444 fuel assemblies and about 20% of them are replaced at the yearly outage. The power is controlled by 109 fine motion control rods and by the main recirculation flow (four speed controlled external circulation pumps).

The containment has a design pressure of 500 kPa and is inerted by nitrogen. Located to the upper part of the containment is the connection to the Multi Venturi Scrubber System (MVSS), se section 1.8.

The turbine plant consists of a double axial high pressure turbine and three double axial low pressure turbines. On the same shaft a generator is coupled, with a water cooled stator and a hydrogen cooled rotor. The generator is connected to the national 400 kV grid.

The electrical power system is divided in two separated parts. When the power plant is not in operation the plant is connected to the 400 and the 130 kV grid. The safety classified electrical systems consists of two emergency diesel generators and two gas turbines. The gas turbines can also feed Oskarshamn 1 and Oskarshan 3.

The modernization of Oskarshamn 2 is not yet fully implemented (2006-2012). One of the objectives of Project PLEX is to verify that the plant can be taken to a safe condition during an earthquake. Within the scope of PLEX, among other things the following safety measures are implemented: two new control buildings with new electrical installation, four new diesels, new cooling chain and modernized control room.

1.6.B.3. Oskarshamn 3

Oskarshan 3 went into commercial operation in 1985. It is a boiling water reactor in Sweden from Asea-Atom and a twin to Forsmark 3.

The reactor core consists of 700 fuel assemblies and about 25% of them are replaced at the yearly outage. The power is controlled by 169 fine motion control rods and the main recirculation (eight speed controlled reactor internal circulation pumps).

The containment has a design pressure of 600 kPa and is inerted by nitrogen. Located to the upper part of the containment is the connection to the Multi Venturi Scrubber System (MVSS).

The turbine plant consists of a double axial high pressure turbine and three double axial low pressure turbines. On the same shaft a generator is coupled, with a water cooled stator and a hydrogen cooled rotor. The generator is connected to the national 400 kV grid.

The electrical power system is divided in four separated parts. When the power plant is not in operation the plant is connected to the 400 and the 130 kV grid. The safety classified electrical systems are divided in four physically and functional subs A-D and each sub has its own emergency diesel generator. Manual connection is possible to the gas turbine at Oskarshamn 2, all necessary cabling is in place.

Within the scope of Project PULS (2004-2009) among other things a new cooling chain was installed.

1.6.C. Ringhals

1.6.C.1. Reactivity control

Ringhals 1

The reactivity control is performed by varying the neutron flux in the core. Depending on the situation there are different ways to use the safety systems and safety functions. In case of a reactor scram, all control rods are automatically injected into the core. The average time for injection is about 4 seconds.

The control rods are used even during a planned shutdown, but then they are inserted by using the electric motors. The average time for insertion is then about 4 minutes. The main circulation pumps are used both at scram and at planned shutdown to reduce the thermal effect. This is done by reducing the pump flow. However, the use of the main circulation pumps is in general not credited in the safety analyses due to the fact that they are not safety classified.

The reactor can also be shut down by using the boron acid system, but this process is much slower than using the control rods.

Ringhals 2, 3 and 4

The reactivity control is performed by control rods that drop into the core at a reactor scram signal. The control rods absorb neutrons and the reactor is brought to subcritical shutdown state. The control rods are held in position by a control rod drive mechanism that is completely encapsulated by RCPB and operated by an electro-mechanical control system. At an interruption of power supply to the control rod drive mechanism, the rods are released and will hence fall into the core. The boron acid system and its function are not formally included in the safety-related part of the reactivity control function. Injection of boron is done by pumping borated water from the boron acid tank, using the boron acid pumps, to the suction side of the charging pumps, and then on to the reactor coolant system.

1.6.C.2. Primary system integrity

Ringhals 1

Pressure relief of the reactor is critical for the integrity of the primary system. Pressure relief is performed using safety valves located on steamlines inside the containment. The system contains valves that are electrically operated and valves mechanically controlled directly by the pressure.

Ringhals 2, 3 and 4

The purpose of the pressure relief function of the primary and secondary system is to provide enough pressure relief capacity so that the integrity of the primary and secondary system can never be put into question. The function is provided by the use of spring-loaded safety valves on both the primary systems and secondary systems. On the primary side, the function is executed by a pressure relief system that connects to the pressurizer and consists of three spring-loaded safety valves. On the secondary side, the function is performed by the use of the 'safety-valve headers'. On each safety-valve header, six safety valves are fitted. The safety valves on the secondary side help to mitigate the increase of temperature on the secondary side, which is obtained when the heat sink is lost.

1.6.C.3. Emergency core cooling function

Ringhals 1

The emergency core cooling function consists of several different systems that can pump water into the reactor to keep the core covered, or when this is not possible, provide spray cooling of the core. These systems take water from the condensation pool, the demineralized water tank, condensers, auxiliary feed water tank, spent fuel pool, industrial water system and the buffer tank. Ultimately, it is also possible to pump seawater. Injection of cooling water into the reactor can be done either by high-pressure emergency core cooling systems or by lowering the reactor pressure using the pressure relief system to enable the use of the low- pressure emergency core cooling system

Ringhals 2, 3 and 4

The emergency core cooling function consists of several systems that can pump water into the reactor in order to keep the core covered. Injection of borated water is done using three different systems: the accumulator, the low pressure injection system and the high pressure injection system. Both the low pressure injection system and the high pressure injection system take water from the refuelling water storage tank. The residual heat removal system is a subset of the emergency core cooling function, although the most frequent use of the system is for removal of residual heat. The pumps in the chemical and volume control system can also be used for high-pressure injection, although their normal operational use is for chemical and volume control.

1.6.C.4. Residual heat removal

Ringhals 1

The purpose of the residual heat removal function is to remove residual heat to a final heat sink after the chain reaction has been interrupted. The function consists of several different systems: cooling water systems, the pressure relief system, the cooling system for the reactor at cold shutdown and the containment spray system.

Ringhals 2, 3 and4

The purpose of the residual heat removal function is to remove residual heat to a final heat sink after the chain reaction has been interrupted. The residual heat removal function is performed by the auxiliary feed water system (a steam turbine-driven auxiliary feed water pump and two electric motor-driven auxiliary feed water pumps) to provide the steam generators with water when normal feed water is not available, e.g. at startup, cooling, reactor scram and safety injection. The auxiliary feed water system has three pumps; consequently, the system is fully redundant. The driving force for two of the pumps in the system is electrical power, while one of the three pumps is driven by steam from the steam generators.

1.6.C.5. Containment Function

Ringhals 1

The design of the containment, in order to reduce the size of it, is based on the PS principle (Pressure Suppression), which means that the steam released by a pipe-break is forced to condense in a water volume. This forced condensation means that the maximum pressure will be lower for a given reactor containment volume than if the steam would flow be released in a reactor containment where condensation would not be possible.

To get the released vapour to condense in water, the containment consists of two distinct areas: primary and secondary compartments. There are vacuum breakers between the secondary and primary compartments in order to prevent the pressure in the secondary compartment from exceeding the pressure in the primary compartment.

Ringhals 2, 3 and 4

The large volume of the containment is credited as heat sink in the initial stage of the design basis events. The containment's structure consists of large amounts of steel, concrete and other construction materials, while the containment exposes large areas in which heat transfer can take place.In order to achieve the overall target for the containment barrier (intact barrier and acceptable emissions), three different targets are defined:

- minimizing leakage paths/closure of isolation valves,
- limiting the pressure and temperature in the containment
- verifying atmospheric composition in the containment.

Isolation of the containment is performed by closure of the isolation valves. Containment spray controls the pressure and the atmospheric composition in the containment.

1.7. Significant differences between units

1.7.A. Forsmark

The significant differences between the reactors are described in section 1.5.A.

1.7.B. Oskarshamn

The Oskarshamn 1 has a safety classified 2-train emergency auxiliary condenser connected to the main steam line and to the return of the condensate to the main recirculation loop. The heat sink is the atmosphere. The condenser can also be used during normal operation when the plant is shut down. The auxiliary condenser is automatically taken into operation at scram. The two emergency diesel generators supplying the auxiliary feed water and the auxiliary condenser are air cooled.

Oskarshamn 1 has surface water cooling water intake. Oskarshamn 2 and 3 have deep water cooling water intake. Oskarshamn 2 can alternate between surface water intake and deep water intake when needed.

The Oskarshamn 1 and 2 has a common storage of diesel oil to supply the day tanks to the gas turbines and emergency diesel generators. The storage tanks are placed inside the fenced off area close to the main guard building. Oskarshamn 3 has separate underground storage tanks for each emergency diesel generator which supply the day tanks.

Oskarshan 1 and 2 have external main recirculation loops. Oskarshamn 3 has internal main recirculation pumps as integral part of the reactor pressure vessel and fly wheels for prolonged roll outs. There are no external pipe nozzles below the core level in the Oskarshamn 3 reactor pressure vessel.

Tables 1.5 and 1.6 describe the main features of Oskarshamn 1, 2 and 3.

Table 1.5 – Main features of Oskarshamn 1, 2 and 3

Plant	Reactor	Containment	Electrical power
O1	BWR/G1/ A-A External MCP	A-A 0,45 MPa, inerted. Similar to Mark II	492 MWe
O2	BWR/G2/ A-A External MCP	A-A 0.5 MPa, inerted. Similar to Mark II	661 MWe
O3	BWR/75 A-A Internal MCP	A-A 0,6 MPa, inerted. Similar to an improved Mark II	1450 MWe

Table 1.6 – Main features of Oskarshamn 1, 2 and 3

Plant	HPCI	RPV heat removal/ Conf. Cooling	LPCS	Aux condenser	Emergency power	Misc
O1	2x100%	0/ 2x100%	2x100%	2x100%	4x100% DG+ 1x100% GT	
O2	2x100%	2x100 %/ 2x100 %	2x100%	No	2x100% DG + 2x100% GT	Main feed water powered from gas turbines with automatic start at scram and when core cooling is needed
O3	4x100%	2x100%/ 4x100%	4x100%	No	4x100% DG	

1.7.C. Ringhals

The safety significant properties and differences regarding the areas of the stress test scope (earthquake, flooding, LOOP/SBO, UHS, severe accident management) are discussed below.

All reactors are finalising programs to improve the safety of the plants as requested in regulations that have been implemented for that purpose (SSMFS 2008:17). These requirements encompass the ability of the reactors to cope with external events.

With regard to the safety properties during earthquake conditions, there are no significant structural differences between the reactors.

The four reactors at the site are exposed to the same seismic conditions. Originally, the reactors were not designed with seismic considerations since Sweden is a zone with low seismic activity.

The modern national regulations require seismic upgrading of the reactors, which is an ongoing work.

The BWR unit is divided into two parts regarding the safety features, OPS (Original Plant Section) and DPS (Diverse Plant Section), where the DPS is seismically qualified.

The I&C-system on Ringhals 2 was recently replaced and seismically qualified.

The essential safety feature with regard to radioactive releases, the filtered vent of the containments, was originally designed with seismic considerations.

With regard to the safety properties at flooding conditions, there are no significant differences between the reactors.

The four reactors are erected at the same ground level, 3,0 meters above sea level, and are consequently equally exposed to flooding. Other potential flooding sources such as dams or rivers do not exist in the vicinity of the site.

With regard to the safety properties at LOOP/SBO conditions, there are differences between the reactors.

All reactors are equipped with four diesel generators, supplying redundant equipment. The durability of the diesel generators is further discussed in the LOOP section of the report.

Ringhals 1 is further equipped with two additional diesel generators, supplying the DPS. (At Ringhals 1, a physically and functionally separated and diversified reactor protection system called DPS (Diversified Plant Section) has been installed. The DPS works in parallel with the Original Plant Section, OPS.)

All reactors have steam driven systems supplying cooling capacity at loss of power situations.

All reactors are supplied with battery capacity for the SBO case. The durability of the batteries is further discussed in the SBO section of the report.

With regard to the safety properties of the UHS, there are no significant differences between the reactors.

The UHS is the sea. The seawater canal system is similar for all reactors.

The properties of the seawater canal system are further discussed in the UHS section of the report.

With regard to severe accident management, there is no significant difference between the reactors.

The severe accident management is based on the presence of the filtered vent feature of the reactors. The severe accident scenario resulting in a reactor vessel melt-through trained in similar ways for the BWR and the PWR's.

The accident management organisation is common for the site.

1.8. Mitigation systems implemented after the TMI accident

After the Three Mile Island accident in the United States in 1979, the Swedish government decided that all Swedish NPPs should be capable of withstanding a core melt accident without any casualties or ground contamination of importance to the population.

This resulted in an extensive backfitting for all Swedish NPPs, including:

- Filtered containment venting through an inerted multi-venturi scrubber system (MVSS) with a decontamination factor of at least 500
- Independent drywell sprays
- All mitigating systems designed to withstand an earthquake
- A comprehensive set of severe accident management guidelines

It was assumed during design that the environmental protection requirements can be met if containment integrity is maintained during the accident sequence (core melt-scenario) and that the releases and leakage from the containment can be controlled and treated.

Several potential threats to containment integrity occur during the core melt process. Briefly, these can be categorized into the following groups: pressure loads due to gas and steam generation, temperature loads due to the high temperature of the molten core, impulse loads due to the interaction between the molten core and water, concrete removal due to contact between the corium and concrete as well as high temperatures and aggressive materials.

The design requirements for the filtered venting function include the following:

- In the case of core damage, prepared plant specific strategies shall exist in order to protect the reactor containment function and to reach a stable condition where the core is cooled and covered by water.
- It is vital that the reactor containment function remains intact during the first 10 to 15 hours after a core damage.
- In order to protect the reactor containment against damage from overpressure during severe accident events, a controlled depressurization of the reactor containment shall be feasible. The pressure relief devices shall be independent of operator actions and independent of other safety systems. The pressure relief devices shall also be possible to operate by the operators.
- The releases shall be limited to maximum 0.1 % of the reactor core content of cesium-134 and cesium-137 in a reactor core of 1800 MW thermal power, provided that other nuclides of significance, from the use of land viewpoint, are limited to the same extent as cesium.

The chosen design scenario is station blackout (SBO, loss of all AC) and loss of steam-driven pumps, with no manual actions credited during the first 8 hours. Without actions or mitigating systems such a scenario will typically give serious core degradation within 1-2 hours, vessel failure within 2-4 hours followed by containment overpressurization and somewhat later gross failure of containment with large releases of fission products, unless mitigating measures are taken.

The requirement is to mitigate the design scenario so that the release of ^{134}Cs and ^{137}Cs will be no more than corresponding to 0.1% of the core inventory of a 1800 MW_{th} core.

After 8 hours the Independent Containment Spray (ICS) is assumed to be available, which will reduce the containment pressure and temporarily terminate the filtered release (or delay the initiation).

In the design scenario for BWRs the pressure in the containment will not reach the design pressure within 8 hours, why the actuation of independent containment spray (ICS) at this time will significantly delay the overpressurization of the containment until more than 24 hours. Somewhat later at a certain pressure the Containment Filtered Vent (CFV) is assumed to be actuated manually, but in case of no manual actions the CFV will be automatically actuated through the bursting of a rupture disk.

In the design scenario for PWR, the pressure in the containment typically will reach the design pressure after 4-6 hours. Since no manual actions are credited during the first 8 hours the CFV will be automatically actuated through the bursting of a rupture disk and will reduce the containment pressure through a filtered release. After 8 hours the ICS is available which will reduce containment pressure and terminate the current CFV-release.

However, (for both BWR and PWR) if no other means of cooling of the containment becomes available the ICS injection cannot continue forever since it will fill up the containment and has to be terminated at a certain level after approximately 30 hours. The pressure will then rise and there will be repeated activations of the CFV with an energy balance established with "feed-and- boil" with the ICS intermittently injecting water and the CFV dissipating energy through a filtered release of steam.

The possibility to passively (no operator decision required) prevent a containment over-pressurization is an important feature when answering the ENSREG questions about protecting containment integrity.

All plants have chosen the FILTRA/MVSS concept to fulfill the requirements.

FILTRA/MVSS stands for Filtered Containment Venting – Multi Venturi Scrubber System. The major component is a scrubber system comprising a large number of small venturi scrubbers submerged in a water pool. A venturi scrubber is a gas cleaning device that relies on the passage of the gas through a fine mist of water droplets. The design of the venturis is based upon the suppliers wide experience in this area gained in designing venturis for cleaning polluted gases from various industrial plants.

Separation of aerosols and gaseous iodine takes place in the venturis and the radioactive matter is transferred to the pool water. Since the venturis are located on different levels and are sealed off by the internal water level, the number of engaged venturis is proportional to the pressure in the containment and thereby the total relief flow. Consequently, each venturi always operate close to the optimal flow and high decontamination factors are therefore achieved over a wide flow range. The activation of venturis is inherently automatic and no external flow control is required. A moisture separator collects the entrained pool water droplets efficiently.

The basic design requirement in Sweden was that depressurization of the reactor containment shall be possible without relying on operator action. This required activation by means of a rupture disc and efficient separation over a very wide flow range. Further, a study design operating at low pressure was preferred, in order not to have the filter function jeopardized by dynamic effects such as hydrogen explosions.

Figures 1.2 – 1.4 of the Filtered venting function system are shown below.

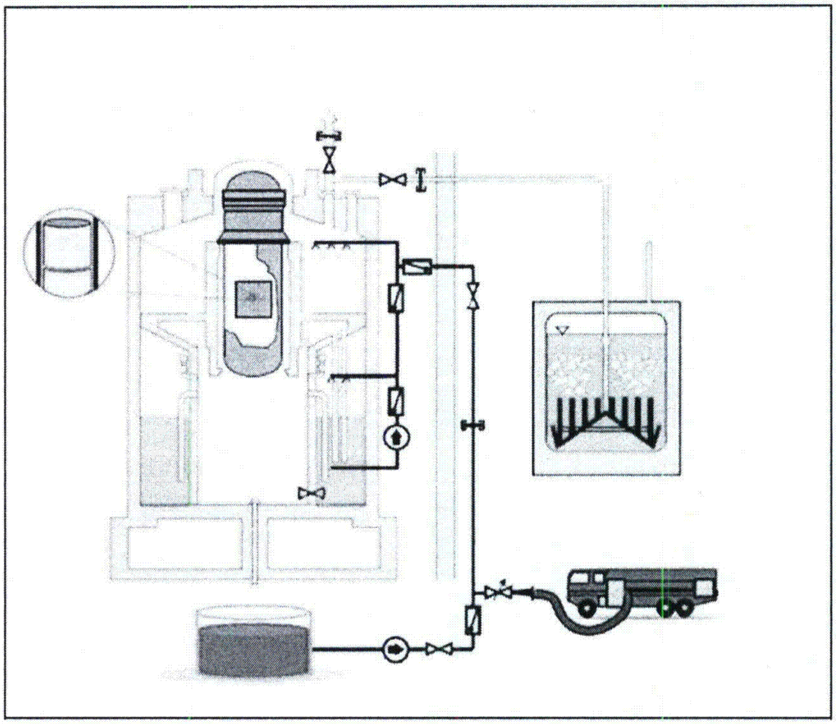

Figure 1.2 - General view of the Filtered Containment Venting Function

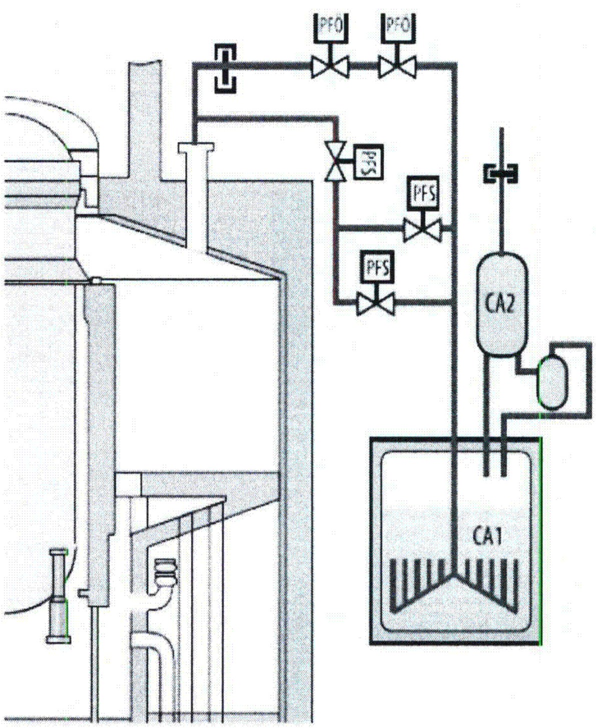

Figure 1.3 - FILTRA/MVSS connection to the containment

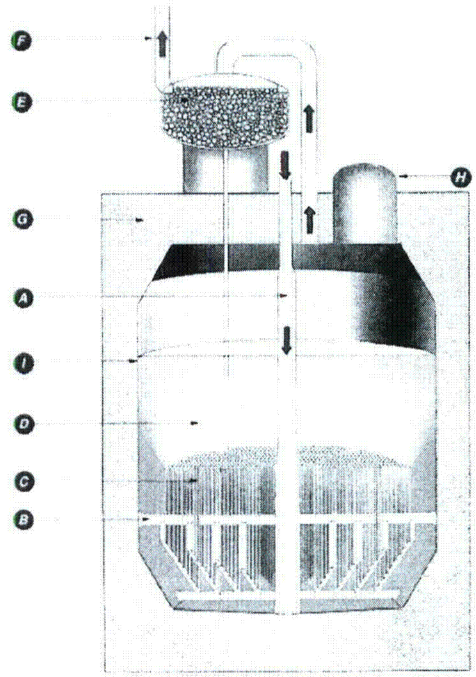

Figure 1.4 - Containment filtered vent principal design.

A = Pressure relief line from the containment B = Venturi distribution

C = Venturis including riser pipe D = Pool E = Moisture

F = Release to atmosphere G = Preassure vessel H = Man hole I = Liner

1.9. Safety improvement programmes of the nuclear power reactors

Safety improvements of the Swedish nuclear power plants have traditionally been conducted through successive plant modifications and special projects as a result of events and problems identified in the plants. These successive modifications have been based on newer reactor designs, which have indicated possible safety improvements, and new insights gained through safety analyses and research. This process has to some extent been driven and confirmed by the periodic safety reviews.

Examples of problems that have led to this type of facility modification include the "strainer incident" at Barsebäck in 1992 when it was found that the emergency core cooling systems in the BWRs with external reactor recirculation pumps did not perform as postulated in the safety analysis reports. The event led to re-evaluations of previous analyses as well as modifications of the affected systems in all Swedish reactors. The problem has also been recognised internationally as a major generic safety issue.

Due to the background in these events the Swedish Authority decided to issue general regulations on design and construction of nuclear power reactors. These regulations, now SSMFS 2008:17, and general advice on their interpretations came into force 1 January 2005 with transitional provisions.

The regulations are based on Swedish and international operating experience, recent safety analyses, results from research and development projects and the development of IAEA safety standards and industrial standards that were applied in the construction of the facilities.

Since the 10 operating power reactors in Sweden have different prerequisites to comply with general regulations on design and construction, an assessment of the consequences was made for each reactor. This assessment includes the issue whether further analyses and back fitting were needed in relation to each paragraph of the regulations.

The following is an overview of the implemented safety related measures in Swedish reactors during the period 1995-2006.

A indication of year is the scheduled time for implementation for the different reactor. The abbreviations used below are:

- F1, F2 and F3 are the three reactors in Forsmark
- O1, O2 and O3 are the three reactors in Oskarshamn
- R1, R2, R3 and R4 are the four reactors in Ringhals

Oskarshamn 1-3

- exchange of moderator vessel vessel head and steam separators (O1)
- safety improvements in the core cooling systems, electric power system (two additional trains), introducing digitalised systems for neutron flux monitoring and the reactor protection system, modernization of the control room (O1)
- a new safety concept based on the safety requirements for modern nuclear power plants (O1)
- new and modernized systems for performing safety functions (O1)
- a modified concept for the reactor protection system and safety I&C including a new emergency control room (O1)
- a modified concept for electrical power supply (O1)
- a new emergency control building, as well as some modifications to existing buildings (O1)
- installation of two diesel generators including auxiliary systems and fuel tanks, completely physically separated (O1)
- installation of two secondary cooling water pumps and heat exchangers for safety systems (O1)
- insatallation of two auxiliary feed-water booster pumps (O1)
- installation of a pump for supplying demineralised water to the auxiliary condenser basin (O1)
- installation of switch gears, batteries and bus bars for the redundant safety trains (O1)
- installation of a physically separated four-train reactor protection system and other I&C equipment (O1)
- installation of a redundant ventilation system (O1)
- replacement of piping, penetrations and valves in the primary systems within the reactor containment (O2)
- replacement of reactor internals, i.e. steam separators, and core spray nozzles and piping (O2)
- changes in the reactor protection system including addition of a new condition for reactor scram (O2)
- improvements of some fire protection systems (O2)
- improvements to reduce risks for hydrogen explosions in piping systems (O2)
- upgrading of feed water control system to programmable I&C equipment (O2)

- separation of safety and non-safety related equipment in some I&C systems (O2)
- upgrading of battery-backed electrical distribution system and change-over of power supply to certain main steam valves (O3)

Forsmark 1–3

- core grids and other reactor internals have been replaced in units (F1–2)
- replacement of equipment in the main circulation pumps to reduce transients on the fuel at loss of external power
- prevention of oxy-hydrogen in steam systems
- diversified reactor vessel level measurement
- new equipment for physical protection
- improved fire safety and security systems
- alteration of the reactor's auxiliary cooling circuits, separation of power supplies and increase in capacity (F1)
- replacement of electrical control boards in the main control room (F2)
- replacement of 6 kV switchboards (F1, F2)
- modification of the reactor pressure vessel head sprinkler (F2)
- modernization of the power measurement system (F2)
- modification of the cooling chain for increased capacity and separation of power supply connections (F2)
- new automatic stop of reactor building ventilation in case of loss of heating system for the building (F3)

Ringhals 1–4

- replacement of primary system piping (R1)
- verification and improvement of piping supports (R1)
- exchange of control rod indication and manoeuvring system (R1)
- introduction of alarm for core instability (R1)
- separation of electric power supply of core cooling systems (R1)
- improvements in fire protection systems (R1, R2, R3, R4)
- improvements of the safety valves of the pressurizer (R2, R3, R4)
- replacements and improvement in the electrical supply systems for improved separation and safety (R2)
- modernization of the radiation monitoring system (R2, R3, R4)
- modernization of the safety injection pumps including vibration monitoring (R3, R4)
- upgrading with redundant cooling of the charging pumps at shut-down (R3, R4)
- modernization of vibration measurement/monitoring of the reactor coolant pumps (R3, R4)
- introduction of cavitation alarms on the residual heat removal pumps (R3, R4)
- fire system modernizations (R1, R2, R3, R4)
- measures to cope with containment sump blockage during design basis accidents (R2, R3, R4)
- improved battery capacity during station black-out (R2, R3, R4)
- securing of piping for the pressurizer. (R2, R3, R4)
- a new main fire water ring installed for the site of Ringhals 1 and 2
- pressurizer relief valves replaced/modified (R2)
- replacement of toroid plates (R2)
- modernization of 110 V DC systems with new switchboards (R2)
- a fourth level measurement channel installed in the steam generators (R2)

- reactor pressure vessel heads replaced (R3, R4)
- pressurizer relief valves replaced/modified (R3, R4)
- new emergency core cooling strainers fitted in the bottom of the containments (R3, R4)

The following is an overview of recently implemented safety related measures in Swedish reactors.

Forsmark 1:

- Modernization of instrumentation for activity measurement in the off-gas system. These modifications comprise detectors as well as electronics.
- Measures to handle slow decreasing voltage in the outside grid. Relay protection modification to disconnect the outside grid if the voltage decreases to less than 85% for 10 sec.
- Capacity and physical separation of cooling chain to the condensation pool improved. These cooling chains are now divided in four sub divisions.
- Partial scram upgraded. Modification comprises design as well as conditions for the activation of partial scram.
- Installation of cyclone filters in the feed water system inside containment. The purpose of these filters is to collect debris which could cause fuel damage
- Reconstruction of the sequence for control rod screw activation in order to fulfil requirements on diversity
- Replacement of the power range monitor system. The new system contains protection against power oscillations
- Improved fire protection of safety functions by additional spray nozzles in culverts containing power and I&C cables
- New high voltage switch gear for connection of Forsmark 1 to the 400kV grid

Forsmark 2:

- Partial scram upgraded. Modification comprises design as well as conditions for the activation of partial scram.
- Replacement of the power range monitor system. The new system contains protection against power oscillations
- Modernization of instrumentation for activity measurement in the off-gas system. These modifications comprise detectors as well as electronics.
- Measures to handle slow decreasing voltage in the outside grid. Relay protection modification to disconnect the outside grid if the voltage decreases to less than 85% for 10 sec.
- Improved fire protection of safety functions by additional spray nozzles in culverts containing power and I&C cables
- New RPV-internals. Moderator vessel head, steam and moisture separators installed.
- Diversified reactivity control implemented. Automatization of the initiation of the boron injection system
- New main steam inboard isolation valves installed
- Reconstruction of the sequence for control rod screw activation in order to fulfil requirements on diversity
- New high voltage switch gear for connection of Forsmark 2 to the 400kV grid

Forsmark 3:

- Measures to handle slow decreasing voltage in outside grid. Relay protection modification to disconnect the outside grid if the voltage decreases to less than 85% for 10 sec.
- Diversified source for emergency feed water to the RPV
- Partial scram upgraded. Modification comprises design as well as conditions for theactivation of partial scram
- New nuclide specific on-line measurement equipment in the stack
- Separation of operation and safety functions in the power system with battery back-up

Oskarshamn 1:

- Monitoring system installed to detect core instability/power oscillations
- Recombiners installed in the turbine off gas system to reduce radioactive discharge to the environment
- Ventilation valves installed on top of the reactor to evacuate non-condensable gases following a loss of coolant accident

Oskarshamn 2:

- Modernization of the feed water system inside the containment involving the exchange of inboard isolation valves, installation of pipe break valves and cyclone filters
- Environmental qualification of components outside the containment
- Recombiners installed in the turbine off-gas system to reduce radioactive discharges to the environment
- Ventilation valves installed on top of the reactor to evacuate non-condensable gases following a loss of coolant accident

Oskarshamn 3:

- Nuclide specific on-line measurement installed in the turbine offgas system with thepurpose to achieve early detection of fuel failures
- Reconstruction of the auto-switching automatics for the diesel bus bars at voltage less than 85%
- Increased capacity of Cooling Cystems -
- Replacement of Reactor Recirculation Pumps, Reactor Internals, Main Steam Line Isolation Valves, Auxiliary Power Transformers, 400kV Transformer and Diversified Cooling Systems

Ringhals 1:

- Part two of fire protection modernization programme completed
- Diversified source for feed water to the core spray system installed
- Modernization project RPS/SP2 completed. The main purpose of these modifications is.
- Major modifications of the reactor protection system and improvement of the residual heat removal systems, to increase the level of separation in order to strengthen defence against fire and to mitigate failures with common cause.

Ringhals 2:

- Passive autocatalytic recombiners installed in the containment

- I&C equipment exchanged to new technology. Modifications include new main control room (MCR), all I&C and cables connected to MCR together with sensors and measuring apparatus in the plant.

Ringhals 3 & 4:

- Diesel back up power supply to the spent fuel pool cooling systems installed
- Passive autocatalytic recombiners installed in the containment
- Upgraded capacity in the heat exchangers to the fuel building cooling systems
- Power operated relief valves at the pressurizer qualified to withstand water blowing
- Fire protection in the relay and cable spreading rooms improved
- Environmental qualification of components in the turbine and auxillary building.

The following is an overview of safety related measures that shall be or have just recently been implemented in Swedish reactors to fulfill the requirements in SSMFS 2008:17.

Improvement of physical and functional separation

- Physical separation within the 220 V systems (F1: 2011, F2: 2012)
- Separation of operation and safety systems within the switchgear (R1: 2013)
- Analysis of the possibility for physical separation in rooms for relays, including measures if necessary (F1: 2011 F2: 2012)
- Modernization of reactor protection system to strengthen the separation of operation and safety systems (O2: 2012)
- Analysis of dependencies between the hydraulic scram system and the pressure relief system, including measures if necessary (O1–2: 2012, O3: 2010)
- Installation of a new pipe for safety injection, due to secondary effects of pipe break (R2: 2012)
- Measures to make the auxiliary feed-water system independent, including a new water supply (R2: 2011)
- Physical separation within the ventilation system in the auxiliary systems building (R2: 2011)
- Analysis of the physical separation within the power system in the auxiliary systems building and the containment, including measures if necessary (R2: 2011)
- Separation within component cooling system (R2: 2012)
- Physical separation to reduce the consequences of steam in connection with a pipe break (R2: 2011)

Diversification of safety functions

- Automation of the boron system for reactor shut down (R1 & O1–3: 2012, F1–3: 2010)
- Analysis of the requirement for two different parameters to identify the need of initiation of the reactor
- Protection system, including measures if necessary (F1: 2011, F2: 2012, F3: 2013, R3–4: 2012, O1–3: 2012)
- Analysis of the requirement for diversified measurement of the reactor pressure vessel level, including measures if necessary (F3: 2010)
- Installation of an external water supply for emergency core cooling (O3: 2010)
- Installation of a new digital reactor protection system and control room modernization (O2: 2012)
- Installation of two phase flow relief valves (O2: 2012)
- Installation of new logic for the pressure relief system (O3: 2010)

Accident management measures

- Additional assessment of the containment integrity in the event of a severe accident, including measures if necessary (all reactors: 2012)
- Strategy for long term cooling of a severely damaged core, including physical measures if necessary (all reactors: 2012, some measures before 2012)
- Change to two phase flow relief valves (R1: 2011)
- Measures to vent incondensable gases from the reactor vessel (R1: 2012)
- Analysis of the emergency control post, including measures if necessary (O3: 2012, R3–4: 2012)
- Installation of a new emergency control post (F1: 2011, F2: 2012, O2: 2012)

Withstanding local dynamic effects from pipe breaks

- Analysis of local loads (F1–3 2010, O1–3 2010), including measures if necessary (F1–3 2011, O1 2012, O2 2007– 2012, O3 2010, R1 2010, R2–4 2011)
- Supports for several containment isolation valves (R2 2011)

Withstanding external events

- Analysis of natural phenomena, including measures if necessary (O1–2: 2012, O3: 2010, R3–4: 2013)
- Analysis of earthquake (R1: 2011), including measures if necessary (R1–2: 2013)
- Measures to the I&C system due to earthquake (O2: 2012)
- Reinforcement of the control room ceiling to survive an earthquake (O1–2: 2012)
- Fire hazards analysis (O3: 2010), including measures if necessary (R2: 2013)
- Improvement of the fire protection (F1: 2010, F2: 2011, O2: 2012)
- Analysis of strong wind, including measures if necessary (O2: 2012)
- Measures to withstand the consequences of strong wind (F2: 2010)
- Reinforcement of the reactor building to withstand flooding (O2: 2012)
- Update of the PSA of flooding caused by pipe break in the salt water system, including measures if necessary (R2: 2012)
- Measures due to risk for turbine missiles (O2: 2012)

Operational aids

- Improvement of the back panels in the control room (R1: 2011)
- Detection of, and automatic protective measures against local core instability (F3: 2010)

Environmental qualification and surveillance

- Update of the environmental qualification (F3: 2010), including measures if necessary
- Update of the environmental qualification outside the containment (O1: 2012, O3: 2010), including measures if necessary (O1: 2012, O3: 2010)

1.10. Scope and main results of Probabilistic Safety Assessments

Probabilistic Safety Assessments (PSAs) are used in Sweden to systematically identify, evaluate and rank different combinations of occurrences that can lead to core damage and/or radioactive releases to the environment. Identification and thus also possibilities to improve risk-dominating events in the NPPs comprise one of the main goals of the probabilistic study.

According to the regulation SSMFS 2008:1, probabilistic safety assessment (PSA) shall be a part of the Safety Analysis Reports (SAR) for the Swedish nuclear power plants. All operating NPPs are expected to perform complete plant-specific Level-1 and Level-2 PSAs, including all operating modes (power operation, shutting down, start up and cold shutdown) and all initiating events that may have an effect on the nuclear safety. Level 1 PSA is an analysis describing the probability of fuel damage. PSA Level 2 is an analysis describing the probability of radioactivity releases and the amount of released fission products.

PSAs are expected to be evaluated annually taking into account plant modifications and operation which have an impact on the PSA models.

1.10.A. Forsmark

In PSA level 1, sequences from initiating events to potential core damage is studied and quantified (core damage frequency per year). The end state of the PSA level 1 is the starting point of PSA level 2. In PSA level 2, sequences leading to core damage are grouped into plant damage states, which via event trees for the containment lead to a number of release categories, which in turn are grouped into release groups: acceptable releases, non-acceptable releases, large releases and large early releases (LERF). The frequency (per year) for each release group is quantified. Of main interest in PSA level 2 is the integrity of the containment and the reactor's ability to limit or stop the release of radioactive material. PSA level 2 is yet only performed for power operation.

There are two PSA studies at Forsmark Nuclear Power Plant, one for the twin reactors 1 and 2, and one for reactor 3. The PSA studies are updated with an interval of approximately two years.

1.10.B. Oskarshamn

Probabilistic safety analysis is used to systematically identify, evaluate and rank different combinations of occurrences that can lead to core damage or/and radioactive release to the environment. Identification and thus also possibilities to improve risk dominating events in the nuclear power plant is one of the main goals of the probabilistic study. The analysis is probabilistic, meaning it is based on probability and reliability calculations and the result is an estimate of the frequency for identified course of events.

The probabilistic safety analysis is complementary to the deterministic safety analysis performed in the rest of the Safety Analysis Report General part (SAR A1 chapter 6).

PSA Level 1 and 2 is performed for all operational modes.

Each PSA model is updated after the yearly outage period if the changes made in the plant affect the PSA model. Changes in SAR are reported to the regulatory SSM.

1.10.C. Ringhals

The PSA models for Ringhals 1, 2, 3 and 4 are plant specific. The focus is on the frequency of core damage, PSA level 1, and the frequency of release of radioactive substances, PSA level 2. The PSA models constitute an important tool for many risk-informed applications and the results are described in the Safety Analysis Report (SAR) for each plant. However it is important to bear in mind that the results of a PSA are valid only given certain conditions and assumptions. This also implies that the numerical values for different plants (including the four reactors at Ringhals) are not directly comparable due to differences in the modelling.

2 Earthquakes

2.1 Introduction

In Sweden, only the two newest reactors, Oskarshamn 3 and Forsmark 3, were originally designed to withstand earthquakes. The other Swedish reactors became subject to general requirements imposed on resilience against earthquakes when the Swedish Nuclear Power Inspectorate's regulations concerning the design and construction of nuclear power reactors, SKIFS 2004:2, entered into force in 2005. In order to allow licensees sufficient time to take the measures and fulfil the requirements, separate decisions were taken giving the licensees a certain period of time to plan and take the requisite measures to fully comply with the mentioned regulations, now designated as SSMFS 2008:17. The deadline for taking measures under these 'transitional decisions' is the year 2013. However, it should be noted in this context that the licensees also previously took resilience against earthquakes into consideration, primarily in terms of mechanical equipment in connection with modernisation work and plant modifications.

Scandinavia is considered to have seismically stabile bedrock and the risk of earthquake caused damaging of buildings has traditionally been considered negligible. Nuclear power plants and other industrial facilities therefore were designed without special consideration of earthquake loads. However, within the safety modernization to current requirements of the older units, extensive modifications have been carried out to ensure that the plants could withstand a seismic event. In addition to hardware modifications, extensive requalification analyses of buildings, systems and components have been performed.

Since the mid 1990s, earthquake induced loads have been considered during plant modifications. In their design and other analyses, the licensees apply a dimensioning earthquake within a radius of twenty kilometres of a strength corresponding to a magnitude of approximately 6.0 on the Richter scale and with a probability of once per 100,000 years (10^{-5}). As far as concerns consequence-mitigating systems, a dimensioning earthquake has been applied of a magnitude approximately four times more powerful and having a probability of once per 10 million years (10^{-7}).

The comprehensive work to develop the envelope ground response spectra for Sweden was performed in the late 1980s and early 1990s in a joint venture project with the Swedish Nuclear Power Inspectorate, Vattenfall, Sydkraft and OKG. The envelope ground response spectra are valid for a typical Swedish hard rock site. The seismicity is defined by the average Fennoscandian seismicity function. The transmission of seismic waves from the source to the surface of the ground is through hard rock with the average properties of Swedish bedrock in respect of its effects on the wave propagation. The "Swedish earthquake" was presented in a report, SKI 92:3, as envelop ground motion spectra with a yearly exceedance probability of 10^{-5}, 10^{-6} and 10^{-7}.

Site specific investigations have subsequently been performed for the Swedish sites by the licensees. The hardness variation in the rock below the plant have been found to be significantly less than the assumed variation for the typical Swedish hard rock site. Hence site specific spectra at the sites are obtained by multiplying the general Swedish hard rock spectra by 0.85.

The mitigation systems, installed during the 1980s in accordance with a government decision, were designed according to US NRC Regulatory Guide 1.60 scaled to peak ground acceleration (PGA) values of 0.15 g horizontal and 0.1 g vertical. This is the same response spectra as the original design response spectra for Oskarshamn 3 and Forsmark 3.

Swedish earthquake risk is dominated by near-field earthquake events and is characterized by high acceleration responses at high frequencies.

Concerning the safety significance of this kind of input motion there is a consensus that it is very low. The Swedish 10^{-5} earthquake corresponds to intensity VI of the modified Mercalli scale (MMI VI). MMI VI is the intensity at which slight damage begins to appear in poor buildings. Therefore there are good reasons to assume that the Swedish 10^{-5} earthquake will not affect the infrastructure, such as bridges, roads and harbours.

2.A. Forsmark

2.A.1 Design basis

2.A.1.1 Earthquake against which the plants are designed

2.A.1.1.1 Characteristics of the design basis earthquake (DBE)

Forsmark 1 and 2:

In the early 1990s Forsmark decided that Forsmark 1 and 2 should apply the "Swedish Earthquake" with an estimated probability of 10^{-5} per year for verification of safe shutdown after an earthquake. This corresponds to horizontal Peak Ground Acceleration (PGA) of 0.09g - 0.11g.

For earthquakes with an expected frequency greater than 10^{-5} per year, it should be shown that safe shutdown is not prevented. This verification uses spectra in SKI 92:3 with an exceedance probability of 10^{-5} (hereafter referred to as the '10^{-5} earthquake'), reduced by 15% to account for area-specific seismic ground movement characteristics, resulting in a PGA of 0.092 g.

Equipment that is replaced or added after 1 January 1992 should be evaluated with respect to earthquake requirements according to the "Swedish earthquake", in combination with other safety requirements. The '10^{-5} earthquake' without reduction is used in the structural verification of reconstructions

The mitigation system specifically designed for severe accidents, introduced following Swedish government decision No. 13 of 1986, are designed to a 'safe shutdown earthquake', or SSE, with ground acceleration of 0.15 g. (The same as for Forsmark 3).

Forsmark 3

In the construction of the plant, buildings and systems significant to safety were designed to a ground response spectrum as stated in the USNRC Regulatory Guide (R.G.) 1.60, scaled to 0.15 g for horizontal ground acceleration and 0.10 g for vertical ground acceleration.

Combinations of ground movements are calculated in accordance with R.G. 1.92 and R.G. 1.60 is used for response spectra and time-history curves.

2.A.1.1.2 Methodology used to evaluate the design basis earthquake

A step-wise approach was implemented at Forsmark 1 and 2 to fulfill the ability to withstand a Design Base Earthquake (DBE). In addition, a Seismic Margin Assessment (SMA) has been conducted.

The ground response spectra prescribed in R.G. 1.60 have been developed by NRC based on work by Newmark et al2. These ground response spectra are in turn based on statistical treatment of severe earthquakes which have occurred in the USA. The ground response spectra in R.G. 1.60 are scaled to 1 g. For earthquake design in the USA, R.G. 1.60 is used, scaled to local accelerations. When Forsmark 3 was designed, as well as in the FRISK project, the same procedure was chosen. The ground response in R.G. 1.60 is characterized by relatively high acceleration at low frequencies (< 10 Hz).

2.A.1.1.3 Conclusion on the adequacy of the design basis for the earthquake

It is shown in a separate report that the impact of the Swedish 10^{-5} earthquake, with regards to damage potential, does not exceed intensity VI on the Modified Mercalli Intensity scale (MMI). MMI VI is the level where limited earthquake damage appears in weak buildings. From this point of view, the risk of an 10^{-5} earthquake damaging a nuclear power plant can be assumed to be very low, even if the plant was not constructed for earthquake loads. It is shown that the Swedish 10^{-5} earthquake without local correction, with a PGA value of around 0.11g, can be compared to an R.G. 1.60 earthquake with a PGA value of around 0.03g-0.05g, which is considered to be low/not dangerous by the majority of experts in the field. This reasoning is based on the observation that earthquake damage does not primarily depend on ground acceleration, but rather correlates better with an integrated measurement of ground movement, called CAV (cumulative absolute velocity).

Utilizing the concept of damage potential, floor response spectra for the 10^{-5} earthquake may be significantly reduced, especially at high frequencies. This is due to the poor modelling of both incoherence and inelasticity of building structures in current calculation methodologies. This line of argument also means that an earthquake in accordance with R.G. 1.60 scaled to 0.15 g (used for design of Forsmark 3) is around 3-5 times stronger than the Swedish 10^{-5} earthquake.

Calculations based on CAV are used in the USA to assess the impact of occurred earthquakes, but the application to design verifications has been very limited. For that reason, we utilize only traditional methods based on ground acceleration when assessing plant compliance in section 2.3 of this work. The concept of CAV may however be used to verify that the methodology for earthquake verification is sufficiently conservative with regard to local conditions. In practice, this conclusion is only utilized in section 3.4, where margins to loss of containment integrity are discussed.

In the SAR of Forsmark 1 and 2 it is stated that for an earthquake with annual exceedance probability 10^{-7} it should be verified that the radiological confinement function of the containment (the containment function) can be maintained. That the containment's structural integrity should be verified for an 10^{-7} earthquake is considered to be a reasonable requirement to eliminate the risk of containment collapse. To maintain the containment function however, is a much harder requirement, since this implies that also reactivity control, containment isolation and protection, and the severe accident mitigation systems must be maintained. This requirement should be placed in relation to other risks which could compromise the containment function. Eliminating emissions in an 10^{-7} earthquake will not appreciably affect the total risk. Based on this, FKA identifies the Swedish 10^{-5} earthquake as being DBE for Forsmark 1 and 2, and an earthquake in accordance with R.G. 1.60 scaled to 0.15g as being DBE for Forsmark 3. The Swedish 10^{-7} earthquake will only be considered where margins for the containment function beyond design are discussed. The Swedish 10^{-7} earthquake corresponds to a PGA of 0.37 g with local reductions, and 0.41 g without local reductions.

The conclusion regarding the adequacy of the design basis is that the DBE for Forsmark 1, 2, and 3 is very conservative with respect to the damage potential of the earthquakes that might appear in the area with a probability of 10^{-5} per year.

2.A.1.2 Provisions to protect the plants against the design basis earthquake

2.A.1.2.1 Identification of the key structures, systems and components

Key structures, systems and components (SSCs) have, in the original design, been assigned safety class 1-3. As Forsmark 1 and 2 were not originally designed for earthquakes, the structures, systems and components were subsequently assigned a seismic classification. The basic requirement in this classification is that safe shutdown should be achieved at Forsmark 1 and 2 using only seismically classified SSCs. Forsmark 3 have originally been design in the same way. The exact lists of which SSCs are seismically classified differ slightly between the units, but the basis of the classification is the same.

In order to ensure a suitable construction in respect of seismic safety, the systems, components and constructions of the units have been divided into three seismic classes.

Seismic class 1 includes buildings, systems and components whose function is required to achieve a safe shutdown after an earthquake. Hence, this class includes structures, systems and components where the function, tightness and mechanical integrity are required during and after an earthquake.

Seismic class P also includes structures, systems and components where only the tightness and mechanical integrity are required during and after an earthquake. No active function is required from equipment in seismic class P.

Seismic class N contains structures, systems and components which are not required during or after an earthquake. Certain systems belonging to seismic class N must in some cases be entirely or partly verified for seismic loads. This is especially applicable if this equipment may damage adjacent equipment belonging to seismic class 1 or P. This is often called the jeopardizing principle.

The strategy for the central control room and building is that the safety of the operators should not be compromised. The operators must be able to get out of the control room and into various areas of the plant.

The safety functions are the functions protecting the plant's barriers. They activate to counteract failing of the barriers in case of an incident. The safety functions are activated automatically and will minimize the consequences of incorrect plant operation and incorrect function of plant equipment.

The safety functions are:

- Reactivity control
- Pressure relief of the primary system
- Core cooling
- Residual heat removal
- Isolation and protection of the containment
- Emergency ventilation

The safety functions also include power supply to the objects. Safety functions are realized by equipment in safety class 1-3. The safety functions are diversified, but generally only one realization of the function is seismically classified. The seismically classified system are however redundant. A brief description of the safety functions credited during and after an earthquake is given below.

Reactivity control

The safety function Reactivity Control should, when required, make the reactor sub critical and then maintain sub criticality. The reactivity control during and after an earthquake is performed by the control rod insertion function.

Pressure relief of the primary system

The pressure relief safety function of the primary system should contribute in maintaining the integrity of the primary system. The pressure relief function is performed by the pressure relief system. Active depressurization (ADS) is performed by the pressure relief system for Forsmark 1 and 2, and by the safety injection system for Forsmark 3.

Core cooling

The core cooling safety function should prevent the fuel pellet and fuel cladding barriers from overheating.

Core cooling at Forsmark 3 during and after an earthquake is done with the auxiliary feed water system, as for other initiating events that do not lead to primary system depressurization.

Core cooling at Forsmark 1 and 2 during and after an earthquake is done with the safety injection system. The auxiliary feed water system is not seismically classified due to its dependence on the system for storage and distribution of demineralized water.

Residual heat removal

The residual heat removal safety function should guarantee the removal of heat generated following the shutdown of the chain reaction in the reactor.

Residual heat removal of the core during and after an earthquake at Forsmark 1,2 and 3 are performed by the cooling trains for Containment spray system. The spent fuel pools are cooled by the cooling train for Spent fuel pit cooling and cleaning system.

The plant has no automatic SCRAM in the event of an earthquake, since an earthquake does not necessarily imply that the plant needs to shut down. If an earthquake leads to the failure of any non-seismically classified equipment, the failure (not the earthquake as such) will initiate a SCRAM (via ordinary RPS circuits).

Hence, SCRAM will only be initiated if any system (operating or safety system) is affected in such a way that SCRAM is required. This can happen either as a consequence of monitoring system being affected in such a way that SCRAM is initiated, or if an operating system being affected in such a way that plant operation goes beyond normal parameters, so that the protection systems are triggered.

Possible scenarios include:

1. An earthquake trips sensitive monitoring equipment. This leads to trip of a RPS circuit (IA-, M- Y- etc.), which in turn initiates SCRAM.
2. An earthquake leads to the failure of operating equipment. Typical faults can be faults in the turbine island, giving turbine trip without bypass, spurious turbine valve closure, or faults in the feed water system. This in turn leads to a SCRAM initiated by high or low level or pressure in the reactor pressure vessel.

3 The earthquake leads to loss of off-site power. A SCRAM is initiated if house-load operation fails.

Consequently, there are two possible scenarios: Either SCRAM is not required and the plant continues in operation, or a SCRAM is required and in that case, there are a number of ways to initiate SCRAM. If a SCRAM is initiated, it is assumed that off-site power is lost. This is standard analysis methodology, but it is also expected in the case of earthquake. It is also assumed that all non-seismically classified equipment is unavailable. Hence, the event sequences will be the same regardless of the cause of the SCRAM.

The main operator interventions to mitigate the consequences of an earthquake are the emergency operating procedures (ÖSI). ÖSI is used by the shift supervisor and is a procedure at plant level for independent verification. This also indicates when and what plant specific accident procedures (ASI) should be used. ÖSI is used regardless of the initiating event.

Forsmark 3 has a specific accident procedure for safe shutdown after an earthquake, using only seismically classified equipment.

There is no operating limit earthquake (OBE) defined in Forsmark, since the earthquakes that are expected in the area are so moderate. There is earthquake monitoring equipment at Forsmark 3 that may be used after an earthquake to conclude whether investigations need to be performed before the plant can be restarted.

In connection to the seismic verification of Forsmark 1 and 2, jeopardizing of the safety system from non-seismically classified systems was identified. Forsmark 3 has been constructed so that safety systems will not be jeopardized by non-seismic equipment in the event of an earthquake. Below are examples of functions which can be degraded by an earthquake. None of these affect the ability to achieve safe shutdown.

Loss of ultimate heat sink

The cooling water intake is verified for earthquakes. However, an earthquake may compromise the strainer cleaning function if there is particularly high contamination in the seawater. If the strainer cleaning function is lost, the auxiliary cooling water flow is maintained by reduction of the main cooling water flow and manual clearing of the strainers.

For Forsmark 3, the auxiliary cooling water flow is filtered using self-cleaning filters. These filters are designed for seismic loads.

Loss of offsite power

The plant is designed to achieve safe shutdown without off-site power. Safety systems will receive power supply from the on-site emergency power supply (diesel generators). This means that the combination of earthquake and loss of off-site power will be similar to only earthquake.

For Forsmark 1 and 2, the common storage tank for diesel oil is not designed to seismic loads. This will affect the time until delivery of diesel oil becomes necessary.

Fire protection for safe shutdown is achieved through passive protection in the form of fire cells, separation, and the use of low-flammable materials. The passive protection is not affected by an earthquake. The active fire protection is a fire-fighting water system and sprinkler central points, combined with equipment for human interventions. There is a fire station located at Forsmark. The active fire protection is not necessary after a DBE. Since the seismically classified safety systems (c.f. section 2.3.1) consists of four separated sub trains, the consequences of the combination of fire and earthquake will not be worse than for only fire.

For all units in Forsmark, structural verification of the primary system has been performed with consideration of seismic loads. Due to that, pipe rupture of the primary system is not expected to occur as a consequence of an earthquake. It is possible, but highly unlikely, for an external pipe break to occur in systems which have not been designed for seismic loads. This is not expected to cause worse flooding than pipe breaks already identified in the internal flooding analyses.

For the ground response levels that the Swedish 10^{-5} earthquake corresponds to, only slight damage of building structures are expected, even if they are constructed without regard to earthquakes. Therefore, there is good reason to assume that an 10^{-5} earthquake will not affect infrastructure such as roads, bridges, railways and ports.

2.A.1.3 Compliance of the plant with its current licensing basis:

In this section the plant compliance with its current requirements are described. Plant compliance is based on requirements for design and for operation. A number of deviations have been identified for Forsmark 1 and 2. These deviations may affect the emergency power supply.

2.A.1.3.1 Licensee's general process to ensure compliance (e.g., periodic maintenance, inspections, testing);

Robust construction, maintenance, and sufficient operational readiness verification is fundamental to prevent damage to key structures, systems and components (SSCs) during and after an earthquake.

Robust construction is achieved by designing seismically classified systems to seismic loads. For components, this is achieved by consideration of seismic demands in the purchase process. For piping systems, the verification is performed by applying seismic loads specified in the design specifications in the stress analysis. The ultimate system is the certificate of conformance that is issued in accordance with SSMFS 2008:13 Chapter 5.

All SSCs have a maintenance program. This program is based on an analysis identifying component materials as well as documentation of ageing mechanisms that could affect the SSC (SSMFS 2008:1 Chapter 5)

Adequate operational readiness verification is managed by the Standard Technical Specifications *(säkerhetstekniska driftförutsättningar*, STF). These specifications comprise limits within which operation of the plant is permitted, with consideration to the safety of its surroundings. The STF are based on the SAR and the analysis requirements applicable to the plant's ability to withstand events and accidents. The aim of the STF is to impose conditions for operation so that the plant, in case of an initial event and with the most limiting single failure applied, will behave within the prescribed limits designated in the safety analysis.

In conjunction with resetting after intervention in the plant, the affected component or system functions must be verified for operational readiness. The principle is that "double barriers" must be applied to all safety-related items. Diversification must be aimed for. In the majority of cases, this means line-up and comprehensive testing after system intervention.

Testing is implemented both following intervention for maintenance and modifications, but also periodically in accordance with prescribed intervals.

Forsmark 1 and 2:

The selected method to verify that safe shutdown can be achieved for Forsmark 1 and in case of an earthquake was SMA (Seismic Margin Assessment). An SMA was performed at Forsmark 1 and 2 in the late 1990's .

SMA is a methodology which combines experience-based evaluations and calculations to assess a plant's ability to withstand earthquakes. SMA is similar to the method developed by the Seismic Qualification Utility Group (SQUG) to meet the requirements from NRC in Unresolved Safety Issue A-46, but has some key advantages. Primarily, each component is described in terms of which earthquake level (in PGA) that a component with high reliability can withstand. This is called HCLPF (High Confidence of Low Probability of Failure). Using these values, the bounding earthquake level that the plant can withstand can be established, and limiting components can be identified.

The use of SMA for earthquake is hence supported by IAEA. In generic letter 88-20, NRC identifies SMA as one acceptable method for seismic verification. Based on these, FKA concludes that SMA is an applicable method for seismic verification.

In the SMA a number of deviations, components that did not meet the requirements, were identified. Under the condition that these outliers have been taken care of, the SMA is the licensing earthquake verification for Forsmark 1 and 2. After that, equipment replaced or installed is verified for the "Swedish earthquake" in addition to other safety requirements.

All piping systems redesigned in preparation for the planned power uprate at Forsmark 1 and 2, or systems for which operating conditions will be altered by the power uprate, have been structurally verified. For seismically classified systems, earthquake loads are included in this verification. In addition, a large number of seismically classified systems have been entirely or partially verified since the earthquake requirements were introduced.

Forsmark 3

Since the unit was originally designed for earthquake selected system functions required for safe shutdown must not be damaged by an earthquake. Consequently, the equipment included in these system functions is designed to meet the functional requirements after an SSE. Selected systems required to maintain the reactor in cold shutdown mode must also be designed to an earthquake.

The original methodology of verifying earthquake requirements is stated in the SAR Chapter 9.12. In short, this methodology includes calculations, tests or other assessments. Calculations have generally been performed for piping systems that have been designed to the ASME III criteria, tests have been performed for e.g. electrical equipment. Other assessments, including walk downs, have been performed for identification of risk for damage to safety systems from non-seismically verified equipment.

2.A.1.3.2 Licensee's general process to ensure that off-site mobile equipment/supplies considered in emergency procedures are available and fit for duty

In case of an earthquake, no mobile equipment is credited. The mobile equipment that is credited during a severe accident is not considered to be affected by DBE.

2.A.1.3.3 Any known deviation, and consequences of these deviations in terms of safety; planning of remediation actions

Forsmark 1 and 2

The performed SMA identified a number of deviations, components which did not fulfil the criteria of the method. In this section, these deviations are described. Some of these deviationshave been corrected since the evaluation was performed. Others can probably be screened due to new knowledge of their seismic ruggedness. Such knowledge may for instance be extracted from the performed structural verification in preparation for the power uprate. The complete compilation of the remaining deviations is still ongoing. No comprehensive investigation of the consequences of the deviations has been performed. Below, a preliminary assessment is presented. This assessment is based partly on the character of the deviations, as they are described in a separate report and partly on whether the failures that might be the consequence of the deviations may be counteracted by operator intervention.

It can generally be concluded that building structures, piping and pressure vessels as well as cable routing all meet the requirements. The status for power supply is more uncertain.

There are two main reasons why certain components do not meet the requirements:

- Insufficient anchorage of mechanical components.
- Uncertainty as to whether unintentional operation of relays affects the requirement for safe shutdown.

Unintentional operation of relays ("relay chatter") may occur since a number of relays in Forsmark 1 and 2 have lower seismic capacity than required for their location. This can in turn depend on both the relays having low seismic capacity, and on the cabinets amplifying the floor response more than they would, had they been designed for earthquake. The consequence of relay chatter may be that objects are given incorrect signals. Exactly what occurs to the plant during the earthquake itself cannot be established. It is likely that the I, Y, A, and M RPS circuits trigger, which in turn initiates SCRAM. However, it cannot be excluded that safety systems initiated by the RPS will be stopped by relay chatter during the earthquake. A tripped SCRAM cannot be blocked by erroneous I&C signals, so control rods will be inserted. Pressure relief of the primary system can be performed either with electrical initiation or with local pressure-controlled opening. This safety function is therefore also able to be performed in the case of chattering relays. For the safety functions core cooling, residual heat removal and containment isolation, it cannot be excluded that valves, pumps and other items are in the wrong position after an earthquake. However, there is no indication that the relays would be permanently damaged by an earthquake. Hence operator intervention will remedy the errors caused by the relay chatter. A general survey shows that this intervention can, to a large extent, be performed from the control room. Manual intervention may also be required in local switchyards. Operator intervention however is only useful in cases where the safety functions are undamaged and power supply is available, see below.

The category of insufficient anchorage of mechanical components includes a large number of component types. An important category is insufficiently anchored electrical cabinets. The identified modifications often imply anchoring them at additional points in the floor and/or anchoring adjacent electrical cabinets to each other. These cabinets have, in many cases, been modified but a number of deviations remain. The consequence of insufficient anchorage is often cabinet interaction, which may lead to relay chatter. As noted above, the effect of relay chatter may be remedied by operator intervention. It can, however, not be excluded that cabinets shake heavily, and even turn over. In those cases the function will be permanently lost.

The assessment indicates that the most severe deviations are those that affect the on-site emergency power supply (diesel engine generated power). This category of deviations contains switchboards for the diesels, diesel oil tanks and the compressed air diesel engine starting system. A number of switchboards, located at the 4th floor of the diesel building, are placed on a raised floor which have not been designed for earthquake loads. If the floor can not provide a sufficient load path, the switchboards may be damaged. The consequence of this may be that pumps in the safety injection system and containment spray system will not be powered. The diesel oil tanks are not anchored. This may lead to fuel line rupture, which in turn will prevent the diesel engines from starting. The consequence of this is that safety functions will not be powered if off-site power is unavailable. A cable routing in the diesel building may fall down during an earthquake. It may land on a safety relief valve of system 752, which may lead to depressurization of the system. In that case, the diesel engine cannot start. In summary, there are a number of deviations that may prevent power supply from diesel generators. These components have a HCLPF of 0.04 g. This means, according to the SMA methodology, that the probability of failure is around 50 % in case of a DBE.

There are also lesser deviations in the process system. This often concerns supports which need to be reviewed. Many of the components identified by the SMA survey as uncertain have however subsequently been included in structural verification prior to the power uprate. Knowledge from these analyses can likely be used to review the conclusions from the SMA.

A large number of deviations concern the ventilation systems. These deviations are of lesser importance though, since ventilation is not vital, at least not early in the sequence.

In summary, it can be concluded that the reactivity control, pressure relief and containment isolation safety functions will most likely function after an earthquake, whilst the situation is more uncertain for core cooling and residual heat removal. The most important deviations are found in the emergency power supply. The conclusion is that it can not be verified that the units can be safely shutdown after a DBE. If power supply can be obtained in at least one subtrain, core damage is likely to be avoided, by means of operator intervention. If both off-site and on-site power is lost in all subtrains, the event will develop into a severe accident (SBO).

The deviations that affect the emergency power supply will be corrected during the outage in 2012. The remaining deviations will be taken care of as soon as possible. Until these deviations have been corrected, the level of earthquake that the plant is verified to withstand is PGA=0.04 g.

Forsmark 3

For Forsmark 3, no deviations in the unit's seismic verification have been identified. The plant was originally designed for earthquake. Modifications of the unit have been designed considering seismic requirements.

2.A.1.3.4 Specific compliance check already initiated by the licensee following Fukushima NPP accident

On 17 March 2011, WANO sent a letter requesting power plants to directly use the recommendations in WANO SOER 2011-2. Forsmark responded to the recommendations and identified the necessary measures. No significant deviations within earthquake protection were discovered during the review.

2.A.2 Evaluation of safety margins

For Forsmark 1 and 2, after deviations have been taken care of, fuel damage may be avoided up to PGA=0.21 g. The containment function is estimated to withstand the Swedish 10^{-7} earthquake, PGA=0.41 g.

For Forsmark 3, fuel damage may be avoided up to 0.35 g. The containment function is estimated to withstand the Swedish 10^{-7} earthquake, PGA=0.41 g.

2.A.2.1 Range of earthquake leading to severe fuel damage

In this section, margins above DBE, i.e. the Swedish 10^{-5} earthquake for Forsmark 1 and 2 and R.G. 1.60 scaled to 0.15 g for Forsmark 3, to loss of safety functions or until damage to fuel is estimated.

Forsmark 1 and 2

Safety system

Provided that the defects identified in section 2.3.3 have been remedied, the margin can be calculated in the same manner as for Forsmark 3 below. The result is that fuel damage may be avoided up to 2.3*DBE=0.21 g.

With the present plant configuration, a number of components have a HCLPF of 0.04 g. Using the line of arguments from above, we estimate that duel damage may be avoided up to 2.3*0.04 g=0.09 g.

Buildings

The containment structure is not compromised at an 10^{-7} earthquake, which meant that it is intact up to 0.41 g. This is also true for the reactor building, the electrical building and the diesel building. The intake building may obtain local damage, but the ability to transport cooling water to the residual heat removal systems is estimated to be unaffected. The integrity of the spent fuel pools is estimated to be preserved up to the 10^{-7} earthquake, 0.41 g.

Forsmark 3

Safety system and building structures

In its original design and in modifications, seismically classed equipment has been verified. The verification aims to show that there are sufficient margins in the capability of a component to withstand a DBE. It can conservatively be assumed that this implies that all seismically classed equipment at Forsmark 3 has an HCLPF, according to the SMA method, of PGA = 0.15 g. HCLPF is mathematically defined as the value at which it can be shown, with 95% confidence, that the probability of failure is less than 5%.

This means that the Median capacity exceeds the HCLPF capacity by a factor of 2.3. Considering the available redundancy (in most cases it is sufficient that equipment in one or two subtrains is available), it may be estimated that safe shutdown can be achieved in the case of an earthquake which exceeds the DBE by 2.3, i.e. an earthquake with PGA=0.35g.

The spent fuel pools have been verified for the Swedish 10^{-7} earthquake, PGA=0.41 g.

2.A.2.1.1 Identification of weak points and specification of any cliff edge effects

Forsmark 1 and 2

In the performed SMA, a number of deviations in the verification of safe shutdown after an earthquake were identified. These deviations are shown in brief above. In conjunction to the SMA, a simplified seismic PSA was performed, in order to support the prioritization of measures. The analysis shows that the seismic capacity for critical components must be increased to 0.09 g in order for the contribution to the core damage frequency from earthquake to be less than 10^{-5} per year.

Furthermore, the analysis shows that if the HCLPF for critical components is greater than 0.1g, the core damage frequency from earthquake will be $6.9*10^{-6}$ per reactor year. The analysis is conducted for power operation, but the core damage contribution in other operating modes is estimated to be in the same order of magnitude.

There is a large number of components which have an HCLPF just above an 10^{-5} earthquake; these components may affect all safety functions, which mean that weaknesses are assessed as being evenly distributed throughout all systems. Structures, process systems and systems independent of power supply can be excluded as "weak points".

Forsmark 3

No HCLPF has been produced for Forsmark 3, but it is estimated that it is likely that the active safety system functions will be lost before the integrity of the systems will be lost.

2.A.2.1.2 Indicate any provisions that can be envisaged to prevent cliff edge effects or to increase robustness of the plant

By far the most important modifications to improve the robustness of the plants is to remedy the deviations identified in the SMA. Apart from that, no simple measures have been identified to prevent cliff edge effects. To identify such measures, further analyses are required, e.g.:

- Perform new calculations of the 10^{-7} earthquake for the reactor building, containment, scrubber building and piping in the unfiltered pressure relief system, the filtered containment venting system, system for water filling of reactor containment and the system for water filling of lower drywell.
- Perform a SMA for Forsmark 3
- Implement the earthquake in the PSA
- Evaluate the adequacy of both the Swedish 10^{-5} earthquake and the R.G. 1.60 earthquake, considering the knowledge that has emerged since these ground response spectra were first established. Special attention should be paid to the conclusions on damage potential.

A few measures to improve the situation during and after an earthquake have been identified:

- The fire-fighting water system is not designed for DBE. In order to lessen the consequences of an earthquake, a number of operating scenarios to deal with a loss of the fire-fighting water system have been produced.
- There are no accident procedure for safe shutdown of Forsmark 1 and 2 using only seismic classified equipment.

2.A.2.2 Range of earthquake leading to loss of confinement integrity

This section focuses on showing that, for an earthquake above the DBE equivalent to the Swedish 10^{-7} earthquake (PGA=0.41 g), the containment's radiological confinement function can be maintained. This means that the consequences of such an event would not significantly exceed the consequences of a severe accident (SBO/H5.2 in SAR section 9.18). The reason for not seeking margins above PGA=0.41 g is that the uncertainties becomes too large for loads so far above the DBE. From the results it can also be concluded that further margins are likely to be small.

With regards to damage potential, the Swedish 10^{-5} earthquake corresponds to a R.G. 1.60 earthquake of 0.03 – 0.05 g. This means that a R.G.1.60 earthquake of 0.15 g is 3–5 times stronger than the Swedish 10^{-5} earthquake. The Swedish 10^{-7} earthquake is around 4 times stronger than the Swedish 10^{-5} earthquake.

Hence, it is reasonable to assume that systems designed to the R.G. 1.60 earthquake of 0.15 g will also be able to withstand the Swedish 10^{-7} earthquake. It must be noted that the arguments regarding damage potential can be applied to building structures and mechanical equipment, but not necessarily to e.g. relays. However, there are no indications that relays are permanently damaged by chatter. This means that these errors are limited to the duration of the earthquake, and may be counteracted by operator intervention. Hence, relay chatter will not affect the possibility to maintain the containment function.

Below, the building structures and safety systems are discussed that need to remain available in order to prevent the consequences of a severe earthquake from going beyond the consequences of a SBO.

Forsmark 1 and 2

Buildings

For the robustness of the containment, reactor building, electrical building, diesel building and the intake building, see section 3.1.1.2. For the scrubber building, FKA notes that it has been designed to the R.G. earthquake of 0.15 g, and hence, with regards to damage potential, is estimated to withstand the 10^{-7} earthquake.

Reactivity control

The scram valves in the scram system are pneumatic, and open in case of loss of air supply. The magnetic valves are provided with close circuit current hence the scram valves will open in case of loss of electrical power supply. This means that the control rod insertion is not dependent on I&C or power supply, i.e. as long as the scram system maintains its structural integrity, the rods will be able to be inserted.

The ability of the system to withstand earthquake is investigated for the 10^{-5} earthquake. New calculations have been performed for the 10^{-7} earthquake. The calculations show that the scram system will be able to meet the requirements set following an earthquake of 10^{-7}, i.e. inserting the control rods into the core.

Containment isolation

To ensure the containment function, the containment isolation valves must be closed. A reasonable assumption when evaluating an earthquake with frequency level 10^{-7} per year is that no single failure has to be postulated. This means that the outer isolations valves may be closed manually in case the automatic closure fails. The verified HCLPF capacity of isolations valves is from 0.0092g to 0.65g if the deviations are resolved. Considering the possibility for operator intervention, it is however reasonable to assume that the function can be performed for considerably more severe earthquakes.

Water filling of lower drywell

The wet well consists of a partly water filled annular area enclosing the lower part of the drywell. In case of a core meltdown, water filling of the lower drywell must be achieved. This is done by opening of valves in the part of containment spray system which connects the wet well to the lower drywell. The valves are powered from UPS, and can also be powered by a mobile diesel generator. It is assessed that water filling of lower drywell can be performed in case of a 10^{-7} earthquake.

Water filling of the containment

Water filling of the reactor containment may be performed using a fire-fighting vehicle, which is connected to the system for water filling of reactor containment. The system connects to containment spray system outside of the reactor containment. The only requirement in order for the function to be maintained is that the system for water filling of reactor containment and part of containment spray system retain their structural integrity so that the leakage is limited.

The system for water filling of reactor containment is verified for the R.G. 1.60 earthquakes. Again, it can be assumed reasonable that the system will meet its requirements after a 10^{-7} earthquake. The containment spray system has been analyzed for 10^{-5} earthquakes. A simple scale shows that the system in an 10^{-7} earthquake will have a stress ratio somewhat above unity inside the containment (max 117%), but this is not considered to prevent water filling.

Assessment of the containment function

The containment structure is not compromised in an 10^{-7} earthquake. The reactivity control and the filtering function are not considered to be compromised. The possibility of manually isolating the containment after an 10^{-7} earthquake has not been fully verified. It is generally considered that emissions from a damaged core can be filtered via the system for water filling of reactor containment.

Forsmark 3

Buildings

The integrity of the containment has been verified for the 10^{-7} earthquake (PGA=0.41 g). The integrity of the scrubber building has not been analyzed in detail, but since it has been designed for the same loads as the containment (R.G. 1.60 earthquake of 0.15 g) it is assessed as being able to withstand 0.41 g. This is supported by the damage potential argument above. The same assessment is made for the other buildings.

Reactivity Control

The scram valves in the scram system are pneumatic, and open in case of loss of air supply. The magnetic valves are provided with close circuit current hence the scram valves will open in case of loss of electrical power supply. This means that the control rod insertion is not dependent on I&C or power supply, i.e. as long as the scram system maintains its structural integrity, the rods will be inserted.

The system has been structurally verified in conjunction with the increase in crud flow temperature. The greatest stress ratio for the load combination containing earthquakes is less than 0.59. This load combination is however dominated by other loads, and it has been shown that the stress ratio is less than unity even for an earthquake four times larger than the DBE.

Containment isolation

To ensure the containment function, the containment isolation valves must be closed. A reasonable assumption when evaluating an earthquake with frequency level 10^{-7} per year is that no single failure has to be postulated. This means that the outer isolations valves may be closed manually in case the automatic closure fails. Hence, it is reasonable to assume that the function can be performed for the 10^{-7} earthquake. This has however not been verified.

Water filling of lower drywell

The wet well consists of a partly water filled annular area enclosing the lower part of the drywell. In case of a core meltdown, water filling of the lower drywell must be achieved. This is performed by opening the valves in the water filling system that connects wet well to the lower drywell. The pipes of the water filling system are short and strait. The system is located in the lower part of the building, where the earthquake loads are small. The valves operate with compressed nitrogen from the pressurized nitrogen system, but may also be operated using nitrogen bottles. The water filling of the lower drywell is therefore judged to be able to be accomplished in case of an 10^{-7} earthquake.

Water filling of the containment

Water filling of the reactor containment may be performed using a fire-fighting vehicle, which is connected to the system for water filling of reactor containment. The system connects to containment spray system outside of the reactor containment. The only requirement in order for the function to be maintained is that the system for water filling of reactor containment and part of containment spray system need to retain their structural integrity so that the leakage is limited.

Both the system for water filling of reactor containment and containment spray system are verified for R.G. 1.60 earthquakes. Considering the damage potential, it can be assumed reasonable that the system will meet its requirements after an 10^{-7} earthquake.

Assessment of the containment function

The overall assessment is that the containment function will not be compromised for the 10^{-7} earthquake, i.e. PGA=0.41 g. Some uncertainty remains regarding the containment isolation. Considering the uncertainties in the assessment, further margins are estimated to be small.

2.A.2.3 Earthquake exceeding the design basis earthquake for the plants and consequent flooding exceeding design basis flood

An event which can cause both a high water level and an earthquake can occur only if there is a tsunami in the Forsmark area. This is regarded very unlikely. Flooding due to human activity is not considered to be a risk, as there are no nearby dams of a size to cause any damage to the power plant.

In the event that both the DBE and DBF are exceeded, cooling of the plant's cores or fuel pools cannot be guaranteed. Nothing has occurred which could dramatically compromise the containment's reactivity control and filtering functions, in the event of both the DBE and DBF being exceeded.

2.A.2.3.1 Results from the licensee assessments of protection against earthquakes

Applied DBE for both Forsmark 1, 2 and 3 is highly conservative in terms of damage potential of earthquakes which with the probability of 10^{-5} per year may occur in the area.

For Forsmark 1 and 2 it is estimated that, with some probability, fuel damage will be avoided up to the level 2.3xDBE = 0.21 g provided that measures will be taken to handle the weaknesses identified. Median capacities for components exceeding HCLPF capacity by a factor of 2.3 have been used for this estimation

For Forsmark 3 a HCLPF equal to DBE = 0.15 g according to the SMA method has been used for earthquake classified equipment. With regard to that and similarly as in the case Forsmark 1 and 2 it is estimated that, with some probability, fuel damage in Forsmark 3 will be avoided up to 2.3xDBE = 0.35 g.

Reactor containments and fuel pools are considered to maintain its structural integrity up to a 10^{-7} earthquake without local correction. The isolation of the containment has not been verified fully.

Scrubber buildings are expected to remain intact for a 10^{-7} earthquake, provided that the earthquake damage potential, expressed by the parameter CAV (Cumulative Absolute Velocity) is taken into account. According to ref [2] have empirically shown that CAV compared with PGA gives a much more consistent measure of the different seismic damage potential.

2.A.2.4 Measures which can be envisaged to increase robustness of the plants against earthquakes

The most important measure to strengthen the Forsmark 1 and 2's robustness is to remedy the shortcomings identified in SMA-validation.

No simple measures to prevent threshold effects are identified. To find such measures there is a need for other more extensive analyses such as new calculations regarding the 10^{-7} earthquake for building structures, seismic PSA and SMA verification of Forsmark 3.

2.B. Oskarshamn

2.B.1 Design basis

2.B.1.1 Earthquake against which the plant is designed

2.B.1.1.1 Characteristics of the design basis earthquake (DBE)

Oskarshamn 1, 2 and 3

Oskarshamn 1 has afterwards been upgraded to withstand a 10^{-5} earthquake. In accordance with the transition plan from the current licensing basis to the requirements in Swedish regulation SSMFS 2008:17, Oskarshamn 2 shall be verified to withstand a seismic event at latest 2012-12-31. The Swedish 10^{-5} earthquake is used as DBE.

The design basis earthquake consists of envelope ground response spectra. The spectral values represent limits of single-degree-of-freedom responses that are expected to be exceeded at the frequency of 10^{-5} per year at site.

The envelope ground response spectra are valid for a typical Swedish hard rock site. The peak ground acceleration (PGA) is 0.11 g in the horizontal and 0.09 g in the vertical direction.

Oskarshamn 3 was originally designed against earthquake induced ground motions based on US NRC Regulatory Guide 1.60 scaled to PGA 0.15 g horizontal and 0.1 g vertical.

2.B.1.1.2 Methodology used to evaluate the design basis earthquake

Oskarshamn 1, 2 and 3

In the mid 1990s a complete SMA (Seismic Margin Assessment) was performed at Oskarshamn 2. The ground response spectra used within these works are the same as the DBE for Oskarshamn 1 and Oskarshamn 3, i.e., the ground response spectra where spectral values are expected to be exceeded at a frequency of 10^{-5} per year and site. Within the ongoing modernization and power uprate project PLEX (Plant Life Extension), one of the goals is to ensure that the plant can withstand a seismic event. The DBE applied within project PLEX is the same as for Oskarshamn 1 and Oskarshamn 3.

2.B.1.1.3 Conclusion on the adequacy of the design basis for the earthquake

Oskarshamn 1, 2 and 3

Since the design basis earthquake is based on Swedish seismological and geological conditions the adequacy is considered to be acceptable.

2.B.1.2 Provisions to protect the plant against the design basis Earthquake

Oskarshamn 1 and 3

The requirement in case of a design basis earthquake (hereafter called DBE) is to bring and maintain the reactor to a safe shut down condition. DBE is assumed to occur during all operational phases including refueling outage. The requirements during power operation can be expressed as:

- A DBE shall not cause an un-isolated loss of reactor coolant exceeding what corresponds from a pipe with an inner diameter of 19 mm, i.e. DBE shall not cause a LOCA. The reactor shall be shut down by hydraulic inserting of control rods (SCRAM).
- The pressure relief of the RCPB shall be ensured by pressure controlled opening of the blow-down valves and safety relief valves.
- If required it should be possible to close containment isolation valves in systems not required after DBE.
- Supply make-up water to the reactor shall be ensured with the low pressure core cooling system.
- Residual heat removal shall be ensured with the residual heat removal system.
- The condition cold shutdown reactor shall be possible to reach via feed-and-bleed of the reactor with the core cooling system and opening of the motor operated relief valves (MORV) in the pressure relief system.
- The cooling of the storage pool for spent nuclear fuel shall be ensured.

As defense-in-depth tasks, in case of core damage due to an earthquake, the containment isolation function and the mitigation systems should fulfill their functions. The defense-in-depth tasks may be expressed as:

- Isolation of RCPB and reactor containment should be ensured and the pressure-suppression function should be ensured later in the sequence.
- Water filling and pressure relief of the reactor containment.
- Pressure relief of the reactor pressure vessel to avoid core meltdown through the reactor pressure vessel (RPV) at high reactor pressure.

2.B.1.2.1 Identification of the key structures, systems and components

To fulfill the above mentioned requirements buildings, systems and equipment are classified and verified to fulfill their safety tasks in case of DBE. The main principle is that two sub-divisions shall be able to fulfill the required safety functions hence handling a single failure. Only seismic classified buildings, systems and equipment are assumed to maintain their functions after a DBE. Non seismic classified buildings, systems and equipment must not jeopardize equipment required after a DBE. Below each required safety function and other credited functions and structures are discussed. For a more comprehensive description, refer to the Safety Analysis Report (SAR). It should be noted that the defense-in-depth tasks are not discussed in those sections because core damage is not anticipated at DBE.

Safety functions

Reactivity control

For reactivity control only hydraulic insertion of the control rods is credited. Hydraulic insertion of the control rods is fulfilled in less than 4 seconds after scram is initiated by the reactor protection system (RPS). In case of a single failure the worst consequence is that one control rod group (scram group) is not inserted into the reactor core. However the reactor will be sub-critical even if one scram group is not inserted.

Emergency core cooling

For core cooling only the following systems are credited:

- Pressure relief system
- Condensation pool system
- Auxiliary feed water system (Oskarshamn 3)
- Emergency core cooling system (Oskarshamn 1)

Primary system integrity protection

For primary system integrity protection the pressure relief system is credited.

Pressure relief of the RCPB may be activated by the RPS at certain pressures and water levels in the RPV. In addition the blow-down valves and safety valves are designed for pressure controlled opening. A single failure of one blow-down valve or safety valve does not prevent the function.

Residual heat removal

For residual heat removal only the following system are credited:

- Pressure relief system.
- Containment heat removal system.
- Primary cooling water system for safety systems.
- Secondary cooling water system for safety systems.

Containment isolation

All containment isolation valves are seismic classified and verified. A DBE may cause different scenario regarding which function or functions that will be activated by the RPS. Therefore automatic closure of containment isolation valves in systems, not required after DBE, may not be performed. However manually (from emergency control room (ECR) closure of motor and pneumatic operated valves may be done if required.

Barriers and structures

RCPB

The RCPB will not be affected of a DBE. Pipes and other pressure retaining components has been verified in accordance with design specifications for pressure retaining and load bearing components (KFM), which include loads resulting from DBE. Therefore no LOCA is assumed at DBE. Due to the low frequency of a DBE and the fact that RCPB is verified against the DBE, a LOCA does not have to be considered concurrently with the DBE according to SAR. The frequency of such an event is extremely improbable and is therefore considered as a residual risk.

Reactor Containment

The reactor containment will not be affected at a DBE. The containment and the reactor building are verified in accordance with design specification for buildings which include loads resulting from DBE.

Other buildings and structures of importance

The following buildings are required to maintain their structure after a DBE. The buildings contain systems or components required to bring and maintain the reactor at a safe shut down condition or are required to enable operating personnel to perform appropriate operational measures.

Oskarshamn 1

- Reactor building including containment and fuel pools
- Pipe culvert 3.51 from the electrical control building (EKB) to the reactor building.
- Turbine building and intermediate building
- Electrical system building
- Screen house and pump building for system 714
- Active workshop building
- Electrical control building (EKB)

The main control room (CKR) is verified for DBE only to the extent that it must not jeopardize the operators and functions needed for bringing the unit to a cold shut down and holding it in that condition. In case the main control room is not functional after a DBE, required maneuvering and monitoring measures can be handled from the emergency control room (ECR) which is designed for DBE. The ECR is a mirror of the CKR in respect to safety and safety related functions.

Oskarshamn 3

- Reactor building
- Cooling water channels
- Culvert Q02.22 and Q02.21
- Auxiliary building
- Auxiliary building
- Control building

- Diesel buildings
- Water intake building
- Cooling water pump building
- Reactor pools and spent fuel pools

The main control room (CKR) and the control building are designed in such a manner that a DBE may not jeopardize the safety of the operators in the CKR. The maneuvering equipment is designed to prevent unwarranted operation of safety related equipment during and after a DBE. Manual measures to bring and maintain the reactor in a safe condition are not required within 30 minutes after a DBE.

After a DBE, monitoring is possible with seismic classified and verified equipment. It is possible to perform manual maneuvering of required equipment after a DBE from the CKR. The maneuvering may also be performed locally.

The above mentioned buildings at Oskarshamn 1 and 3 can withstand a DBE. The buildings are verified in accordance with design specification for buildings.

The primary cooling water system for safety systems in Oskarshamn 1 is credited after a DBE. Therefore the water provision to system 714 must not be totally blocked, hence the following structure has been verified against seismic loads.

- Inlet channel
- Outlet channel
- Return channel
- Swell pool

Electrical power supply

Oskarsham 1

Loss of off-site power and loss of house load operation are assumed at DBE. Further the gas turbine backed power and the diesel generator are not credited since they are not seismic classified. Only diesel generators in sub division A and B are credited. One diesel generator is sufficient to fulfill the required safety function thereby a single failure can be handled. The diesel generators will start automatically in case of loss of the ordinary power supply. The diesel generators in sub A and B are physical and functionally separated and installed in the electrical control building (EKB). Each diesel generator is provided with a diesel storage tank with a capacity for 24 hours full power operation. After 24 hours, manual measures are credited for refilling the diesel storage tanks from e.g. mobile fuel trucks.

Refilling of the diesel storage tanks is normally performed from the gas turbine storage tanks. However the gas turbine storage tanks are not seismic classified, and neither are the oil transfer pumps supplying diesel oil from the gas turbine storage tanks to the diesel storage tanks.

Within the Seismic Margin Assessment (SMA) for Oskarshamn 2 the gas turbine storage tanks and the large oil storage tanks were evaluated. Calculations of the HCLPF (High Confidence of Low Probability of Failure) capacity for the tanks were performed and the results show capacities above the RLE (Review Level Earthquake). The RLE is the site specific ground spectra for an earthquake with frequency level 10^{-5} per year, which means the general envelope ground response spectra for a typical Swedish hard rock site multiplied by 0.85. However the supporting calculations referred to in the SMA have not been located. Despite the missing calculations, it is reasonable to assume that the integrity of the tanks will be preserved in case of DBE. The oil transfer pumps have not been evaluated in the SMA for Oskarshamn 2 and may therefore not be considered as seismically adequate.

There is no requirement in the STF concerning diesel oil in the storage tanks. All uninterruptible power supplies credited after a DBE are installed in the EKB. There are two battery backed systems which is a 400V/230 VAC battery backed system and one system which is a 110 V DC battery backed system. With respect to seismic requirement the batteries in sub-division C and D may be recharged with diesel generators in sub-division A and B in case the non-seismic classified diesel generators in sub-division C and D are unavailable. The safety requirement is that the batteries shall supply the connected consumers for a period of at least two hours given all connected loads are at nominal power consumption.

Oskarshamn 3

Loss of off-site power and loss of house load operation is assumed at DBE. Further manual connection of gas turbine backed power is not credited because it is not seismic classified. Diesel generators in sub divisions A, B, C and D (system 651) are credited. Two diesel generators are sufficient to fulfill the required safety function. The diesel generators will start automatically in case of loss of the ordinary power supply. The diesel generators are physical and functionally separated and installed in the diesel buildings KA, KB, KC and KD. Each diesel generator is provided with a diesel storage tank with a capacity for 7 days full power operation.

After 7 days, manual measures are credited for refilling the diesel storage tanks. Uninterruptible power supplies credited after a DBE consist of battery backed DC and AC systems. Three battery backed DC systems supply required consumers. Each system consists of four physically and functionally separated sub-divisions (A, B, C and D). There is also a 400V AC battery backed system with four physically and functionally separated sub divisions (A, B, C and D). The batteries in each sub-division may be recharged with the diesel generator belonging to their sub-division. The safety requirement is that the batteries shall supply the connected consumers for a period of at least 2 hours given all connected loads are at nominal power consumption.

Oskarshamn 2

The transition plan from the current licensing basis to the requirements in Swedish regulation SSMFS 2008:17 stipulates that the plant shall be verified to withstand a seismic event at latest 2012-12-31.

In accordance with current licensing basis, the formal requirement is that the mitigation systems should be able to fulfill their functions for a DBE. The mitigation systems were installed in 1988 in accordance with a government decision. The mitigation systems ensure pressure relief of the reactor containment and a limited release of radioactive nuclides to the environment in case of a core meltdown. The requirement concerning release of radioactive nuclides is that less than 0.1 % of the core inventory of the isotopes cesium-134 and cesium-137 in a reactor core with a thermal power of 1800 MW should be released, with the prerequisite that other nuclides of importance concerning ground contamination are separated in the same proportion as the cesium isotopes.

The mitigation systems consist of the following system functions:

- Sprinkling and water filling of the reactor containment. Water supply is possible from the fire water system using a direct driven diesel water pump. Further it is possible to use fire trucks by the existing connections in the plant the pipes, including the connection for fire trucks, are seismic classified.
- Controlled pressure relief of the reactor containment is performed via the MVSS (Multi venturi scrubber system) connected to the reactor containment with a pipe. The

pipe between the reactor containment and MVSS is provided with a valve and a rupture disc in parallel thereby ensuring the controlled pressure relief function even if an active single failure occurs. The radioactive nuclides that may be released from the containment atmosphere after opening of the valve or the rupture disc are separated in the MVSS, which has a decontamination factor of 500. The MVSS requires no electrical power supply to perform its function.
- Pressure relief of the reactor containment in case of a pipe break in the reactor containment assuming the pressure suppression function is degraded.
- Activity monitoring after the MVSS.

According to Safety Analysis Report (SAR) the mitigation systems are designed in accordance with the US NRC Regulatory Guide 1.60 ground response spectra scaled to PGA 0.15g horizontal and 0.1g vertical.

Extensive analyses have been carried out evaluating the plant's ability to withstand an earthquake beyond what is required within the current licensing basis. Evaluations of structures, systems and components have been performed in different ways, such as calculations and assessments using experience-based methods. The experience-based method used is the Seismic Margin Assessment (SMA). Earthquake induced loads have been considered when modifications were carried out at Oskarshamn 2. A summary of the evaluations that were performed is provided below.

<u>Seismic Margin Assessment for Oskarshamn 2</u>

A description of the applied SMA methodology is presented briefly in the section describing Oskarshamn 3 below. The same Review Level Earthquake (RLE) as for Oskarshamn 3 was used. The functions evaluated within the SMA, required to reach a safe shutdown condition after a SE, are as follows:

1. Reactivity control
2. Reactor coolant system inventory control
3. Reactor coolant pressure control
4. Residual heat removal
5. Containment isolation and integrity

The structures and areas where the systems and components are located include the following:

- Reactor containment
- Reactor building
- Electrical building
- Pump house
- Diesel building
- Yard

Based on the systems required to perform the functions to reach a safe shutdown condition, a Safe Shutdown Equipment List (SSEL) was developed. The majority of the equipment in the SSEL was judged to be seismically adequate. However some deficiencies were identified by the Seismic Review Team (SRT). The main concern was insufficient anchoring of equipment, mainly electrical equipment such as charger/inverters and distribution panels and motor control centers. Further there were relays and contactors not fulfilling the criteria in the relay evaluations. Regarding relays, the deficiencies predominantly concern RXMA 2 model relays. The buildings where equipment in the SSEL is located were judged to withstand a SE.

Structural Analyses for Oskarshamn 2

Structural analyses of pipes and other pressure retaining components has been performed in accordance with design specifications for pressure retaining and load bearing components (KFM), which include loads resulting from SE. Important systems and barriers analyzed against SE induced loads include:

- Reactor Coolant Pressure Boundary
- Low Pressure Core Cooling System
- Containment heat removal system
- Scram system

Seismic Margin Assessment for Oskarshamn 3

The Seismic Margin Assessment (SMA) was introduced by the Electric Power Research Institute (EPRI). The SMA methodology is a derivative of the SQUG (Seismic Qualification Utilities Group) experience based evaluation methodology. The methods were developed by SQUG to address GSI (Generic Safety Issue) A-46 concerns. GSI A-46, issued by the USNRC (United States Nuclear Regulatory Commission) addressed the ability of older commercial nuclear plants that lacked seismic design to safely shut down in the event of an earthquake. SMA was an extension of the SQUG method for plants that have a seismic design basis, and the results are used to express a plant's ability to shut down at earthquake levels beyond the design basis.

Seismically induced flooding and fire

Oskarshamn 1 and 3

Flooding caused by DBE has been analyzed for Oskarshamn 1 and 3. The analysis of flooding events and the evaluation of protection measures are based on the methodology in ANSI/ANS 56.11 – 1988 "Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants". The analysis was emphasized that flooding may endanger electric and I&C-components of system required after DBE. The most critical equipment are level and pressure transmitters in the reactor building at level + 95 m (the level is related to normal Baltic Sea water level at + 100 m).

Safety equipment required after DBE installed lower than level + 95 m in the reactor building, such as emergency core cooling system and residual heat removal system, are installed in watertight compartments, hence flooding of those components will not occur. For some identified leakage sources, manual measures are required to terminate the leakage. Manual closure of valves must be accomplished to prevent an unacceptable volume of water from being discharged into the reactor building. Most important is to stop leakage from the demineralized water system, sewage water system and fire water system. The maximum leakage flow rates from those system are conservative 324 m3/h (the system for storage and distribution of demineralized water), 180 m3/h (system 766) and 175 m3/h (system 861). After 30 minutes isolation of the leakage is assumed. Accordingly approximately 150 m3 of water could be discharged into the reactor building in case of a pipe rupture of the system for storage and distribution of demineralized water.

Leakage from broken tanks is not assumed to be stopped. Instantaneous emptying of the internal tank volume is assumed.

All areas in the reactor building withstand the pressure of a water level of 6.6 m inside the building. Thereby the integrity of the building is not jeopardized.

In case a seismic induced flooding affect the reactor building from the outside, for example due to rock cracks or cracks in the cooling water channel, the reactor building withstand the pressure of a water level up to + 101.5 meter. (1.5 m above the normal average sea water level in the Baltic Sea). According to Swedish Pilot Part A issued by the Swedish Maritime Administration (Sjofartsverket) a sea water level of 1.35 m above normal average sea water level is the highest registered during 1887-1983 at the north cape of the island Oland, about 25 km east of the site.

As an addition to the ENSREG specification of the stress test, SSM points out that fire as a consequence of earthquake may jeopardize buildings, systems or components required to bring and maintain the reactor at a safe shut down condition.

Fire analyses at Oskarshamn 1 and 3 is in general performed according to SAR. Effects of fire as a result of an earthquake have not been performed.

Oskarshamn 2

In the SMA for Oskarshamn 2, careful considerations were given to this issue. No seismically induced flooding concerns were found. However issues regarding unwarranted spraying by the fire sprinkler system were noted. Those issues are taken into consideration within the ongoing modernization and power uprate project, PLEX. It should also be mentioned that in the SMA, seismically induced fire concerns were noticed in the area housing the gas turbines, in particular the fuel oil reservoirs. However, these items were subsequently removed from the final Safe Shutdown Equipment List (SSEL) when it was confirmed that the two diesel powered emergency generators could be reconfigured to provide two separate shutdown paths.

Fire analysis at Oskarshamn 2 is in general performed according to SAR. Effects of fire as a result of an earthquake have not been performed.

2.B.1.3 Compliance of the plant with its current licensing basis

2.B.1.3.1 Licensee's general process to ensure compliance (e.g., periodic maintenance, inspections, testing);

Oskarshamn 1

Seismic classified structures, systems and components have been verified against seismic induced loads by different methods. This can be summarized as follows.

- Within the safety modernization project (MOD) all seismic classified buildings were qualified against DBE by calculations.
- Seismic classified piping systems installed within project MOD were qualified by calculations in accordance with the KFM. Further, existing piping systems that belong to the

RCPB were qualified by calculations in accordance with the KFM. Other existing piping systems that became seismic classified within project MOD were qualified using experience based method, which are described below.

- Seismic classified mechanical and electrical components installed within project MOD are qualified by calculations and tests in accordance with the requirements in Technical requirements for mechanical equipment (TBM) and Technical requirements for electrical equipment (TBE). The calculations and tests are based on DBE. This includes among other things all seismic classified components installed in the Electrical Control building (EKB).

- Existing mechanical and electrical components that became seismic classified within project MOD were qualified using the SQUG based experience data evaluation methodology, which is described below. This mainly concerned existing components such as valves and pumps in the reactor building.
- The methods used to verify the requirement that non seismic classified buildings, systems or equipment must not jeopardize equipment required after a DBE have been adapted from case to case. The main method used has been SQUG, but other evaluations methods have also been used, such as the seismically induced flooding analysis.

The Seismic Qualification Utilities Group (SQUG) developed *"Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment"*, to address questions posed by the U. S. Nuclear Regulatory Commission regarding the seismic qualification of equipment in nuclear power plants that were built prior to the time when specific seismic requirements existed. The USNRC and other international regulators have endorsed the methodology as one approach to demonstrate the seismic capability of selected equipment in nuclear power facilities. The methodology described in GIP requires an equipment walkdown by qualified seismic engineers and the documentation of the walkdown observations on Screening Evaluation Work Sheets (SEWS). SEWS must be signed by at least two members of the Seismic Review Team (SRT). Those who sign the SEWS must be certified by SQUG to have successfully completed the official walkdown screening and seismic evaluation training course. There are 22 different SEWS, one for each of various types of equipment commonly found in nuclear power plants, and the checklist items on the SEWS are unique for each equipment type. The SEWS are divided into four sections, each of which is a basic element of the equipment evaluation process. Those four basic elements determine an equipment item's ability to successfully withstand an earthquake.

Piping is not included in the GIP procedures. In the EPRI Seismic Margin Assessment (SMA) criteria, piping may be screened out for earthquakes with 5 % damped peak ground motion spectral acceleration of up to 0.8 g. The EPRI screening guidelines are based on the successful performance of piping systems in strong motion earthquakes plus the fact that the piping in US plants already had a seismic design for a lower level earthquake. For Oskarshamn 1, which was a plant with no seismic design basis for piping, emphasis was placed on the seismic experience and the very low seismic demand on the piping.

The overall requirements regarding maintenance, testing and inspections to maintain function, material status and condition of buildings, systems and components are described in a document presenting requirements of the activities.

<u>Oskarhamn 2</u>

The overall requirements regarding maintenance, testing and inspections to maintain function, material status and condition of buildings, systems and components are described in "Ledningsdokument Verksamhetskrav".

<u>Oskarshamn 3</u>

Seismic classified structures, systems and components have been verified against seismic induced loads by different methods. This can be summarized as follows.

- Seismic verification of buildings was originally done by calculations. The seismic loads were based on US NRC Regulatory Guide 1.60 scaled to PGA 0.15g horizontal and 0.1g vertical. Within the power uprate project PULS, all seismic classified buildings were re-qualified against the current DBE by calculations in accordance with the KFB. The verification reports are presented in OKG's documentation system (H-dok).

- Verification of seismic classified piping systems was originally performed by calculations. The seismic loads were based on US NRC Regulatory Guide 1.60 scaled to PGA 0.15g horizontal and 0.1g vertical. Within project PULS, all seismic classified piping systems were re-qualified against the current DBE by calculations in accordance with the KFM. The verification reports are presented in OKG's documentation system (G-dok).
- Seismic verification of mechanical and electrical components was originally done by calculations and tests. The seismic loads were based on US NRC Regulatory Guide 1.60 scaled to PGA 0.15g horizontal and 0.1g vertical. Within the power uprate project PULS, mechanical and electrical components were requalified using the SMA methodology that is described below. The results from the calculations and tests are presented in OKG's documentation system (K-dok).
- Seismic classified mechanical and electrical components installed within project PULS are qualified by calculations and tests in accordance with the requirements in Technical requirements for mechanical equipment (TBM) and Technical requirements for electrical equipment (TBE). The calculations and tests are based on the current DBE.
- The methods used to verify the requirement that non seismic classified buildings, system or equipment must not jeopardize equipment required after a DBE has been adapted from case to case. The main method used has been SMA, but other evaluations methods have also been used, such as the seismically induced flooding analysis.

2.B.1.3.2 Any known deviation, and consequences of these deviations in terms of safety; planning of remediation actions

<u>**Oskarshamn 1**</u>

To ensure operability of required safety functions in the long term in case of DBE provision of supplies such as diesel oil, lubricant and fuel oil filters may be needed. Therefore it is recommended to perform an updated survey of the inventory of supplies of importance stored in the plant or on site. The survey should emphasis on that appropriate amount is stored on site and that the storage/storages may not be jeopardized by DBE. The ability to transport required supply from the on-site storage location to the machinery/equipment concerned should be evaluated.

Further, the ability to provide external supply to the site should be evaluated. The amount of provisions stored on site should be based on the time required for replenishment from off-site sources.

There is one issue regarding that the inner ceiling in the main control room (CKR), which is not verified against DBE, could jeopardize the operators'safety. This issue is handled within a project. In accordance with the regulator transition plan to the requirements in Swedish regulation SSMFS 2008:17, this issue should be addressed at the latest 2012-12-31.

<u>**Oskarshamn 2**</u>

According to OKG there are no known deviations against the current licensing basis. Deviations identified when the plant was evaluated against SE are taken into consideration within the ongoing modernization and power uprate project.

Oskarshamn 3

Within the work to re-qualify the plant against the Swedish earthquake with the frequency of 10^{-5} per year, some outliers were identified in the SMA. Most of these outliers have been further evaluated and are now considered to be seismically adequate. However some outliers require other measures e.g., modified anchoring. Measures were carried out during the 2011 Oskarshamn 3 refueling outage and the remaining outliers are planned to be handled during the 2012 Oskarshamn 3 refueling outage.

2.B.1.3.3 Specific compliance check already initiated by the licensee following Fukushima NPP accident

A specific check has been done in accordance with SOER 2011-12 issued by WANO. Measures carried out are summarized in OKGs response to WANO. (2011-10724E utg 1.0 Response from Oskarshamn NPP regarding to WANO SOER82011-2 Fukushima Daiichi Earthquake and Tsunami)

2.B.2 Evaluation of safety margins

2.B.2.1 Range of earthquake leading to severe fuel damage

Oskarshamn 1 and 3

No seismic PSA has been performed for Oskarshamn 1. The margins regarding capacity to achieve a safe shutdown condition in case of an earthquake causing ground motions exceeding DBE are therefore difficult to predict. However a rough estimate presented below indicates margin. Seismic classified structures, systems and components are verified either by calculations, tests or experience based methods, e.g., SQUG. The HCLPF (High Confidence of Low Probability of Failure) associated with the SMA methodology mean 95 % confidence of less than 5 % probability of failure. The SMA methodology has much in common with the SQUG methodology. It may conservatively be assumed that the HCLPF capacity of all structures, systems and components precisely corresponds to values required to withstand a DBE. The hybrid method may be used to estimate the median seismic capacity.

The median seismic capacity for seismic classified structures, systems and components is calculated to 2.5 times the HCLPF capacity. The existing redundancy of components, and the capability to handle a single failure in required safety function, indicates margins exist to achieve a safe shutdown condition in case of an earthquake exceeding DBE. The ability to achieve a safe shutdown condition is estimated to be comparatively high in case of an earthquake causing ground motions that exceed the corresponding values for DBE by a factor of approximately 2.5.

The estimated margin is associated with uncertainties. The conservative assumption that the HCLPF capacity of all structures, systems and components precisely corresponds to values required to withstand a DBE is probably too conservative. The evaluations of structures and piping systems indicate margins beyond the factor of 2.5 calculated using the hybrid method. However, margins concerning electrical and I&C equipment have not been evaluated further within this stress test due to the limited time available. Therefore, it would be reasonable to carry out further investigations to evaluate the margins of structures, systems and components. Such investigations should emphasize evaluating margins to reach a safe shutdown condition. A reasonable approach would be to evaluate only one sub division in the systems required to bring the reactor to a safe shutdown condition. In general, the designs of the sub divisions in required systems are similar.

The assumed earthquake corresponds to a severe beyond design basis earthquake, where the spectral values represent limits of single-degree-of-freedom responses that are expected to be exceeded at the frequency of 10^{-7} per year. The approach to focus on more careful analysis of those system functions is reasonable. If the mitigation systems, the spent fuel pool integrity and the spent fuel pool cooling, are judged to be ensured the margins are considered to be adequate. The evaluated earthquake is considerably more severe than the DBE. By ensuring the mentioned functions the consequences to the environment are judged to be acceptable and in accordance with the requirements in the government decision regarding release of radioactive nuclides to the society and the environment in case of a core meltdown.

The analysis is emphasized to evaluate the mitigation systems ability to perform its function.

The approach when analyzing the margins of earthquake ground motions beyond the DBE is as follows:

- The assumed earthquake corresponds to a severe beyond design basis earthquake, where the spectral values represent limits of single-degree-of-freedom responses that are expected to be exceeded at the frequency of 10^{-7} per year. The peak ground acceleration (PGA) is 0.42 g. Compared to the DBE the PGA values are about 4 times higher. The spectral values are considerably beyond the values for DBE.
- Evaluate the mitigation systems ability to fulfill their functions. The mitigation systems ensure pressure relief of the reactor containment and a limited release of radioactive nuclides to the environment in case of a core meltdown. The requirement concerning release of radioactive nuclides is that less than 0.1 % of the core inventory of the isotopes cesium-134 and cesium-137 in a reactor core with a thermal power of 1800 MW should be released, with the prerequisite that other nuclides of importance concerning ground contamination are separated in the same proportion as the cesium isotopes. The mitigation systems were installed in 1988 in accordance with a government decision.
- According to the government decision regarding the mitigation systems, extremely improbable events do not need to be considered. A spontaneous rupture of the reactor pressure vessel is usually referred to as an example of such an extremely improbable event. The expected frequency of a spontaneous rupture of the reactor pressure vessel is $2.7 \cdot 10^{-7}$ per year according to WASH-1400. The assumed earthquake with frequency of 10^{-7} per year at the site therefore is considered as a residual risk. Therefore it is not reasonable to assume earthquakes with lower frequencies than 10^{-7} per year at the site.

Oskarshamn 2

The analysis is emphasized to evaluate ability of the mitigation systems to perform their function. The approach when analyzing the margins of earthquake ground motions beyond the SE is the same as for Oskarshamn 1 and Oskarshamn 3.

The assumed earthquake with frequency of 10^{-7} per year is therefore considered as a residual risk. Therefore, it is not reasonable to assume earthquakes with lower frequencies than 10^{-7} per year.

2.B.2.2 Range of earthquake leading to loss of containment integrity

Oskarshamn 1, 2 and 3

Mitigation systems

The mitigation systems that ensure limited release of radioactive nuclides in case of core meltdown consist of the following system functions:

- Spray and water filling of the reactor containment. Water supply is taken from the fire water system using a direct driven diesel water pump. Further it is possible to use fire trucks by the existing connections in the plant. The pipes including the connection for fire trucks are seismic classified.
- Controlled pressure relief of the reactor containment is performed via the MVSS (Multi venturi scrubber system) connected to the reactor containment with a pipe. Both the containment and the MVSS are inerted during power operation. The pipe between the reactor containment and MVSS is provided with a valve and a rupture disc in parallel thereby ensuring the controlled pressure relief function even if an active single failure is assumed. The radioactive nuclides that may be released from the containment atmosphere after opening of the valve or the rupture disc are separated in the MVSS, which has a decontamination factor of 500. The MVSS requires no electrical power supply to perform its function.

The containment isolation provision is evaluated because of its importance to ensure the required limited release of radioactive nuclides.

The scram system is evaluated because in present analysis of severe accidents sub-critical condition is assumed.

The integrity of RCPB is evaluated even though the mitigation systems may perform their function in case of a LOCA.

Reactor Containment and Fuel Storage Pools

The reactor containment has been evaluated against DBE. After a general survey of the stress levels, some structural elements were identified as critical.

Thus, on the basis of the calculations and engineering judgments, it is likely that the integrity of the reactor containment and the fuel storage pools will be preserved in case of an earthquake with frequency level 10^{-7}per year.

However, the assessment is based on rough up-scaling calculation methods and engineering judgments on a best estimate basis due to the limited time available for this study. Therefore, it may be reasonable to carry out further investigations regarding the structural integrity of the reactor containment and fuel storage pools.

Scrubber building

The scrubber building for Oskarshamn 1 and 2 is the same as for Oskarshamn 3. Therefore the results for Oskarshamn 3 are valid also for the other two reactors. The conclusion is that it is likely that the integrity of the scrubber building will be preserved in case of an earthquake with frequency level 10^{-7} per year.

However, the assessment is based on rough up-scaling calculation methods and engineering judgments on a best estimate basis due to the limited time available for this study. Therefore, it may be reasonable to carry out further investigations regarding the structural integrity of the scrubber building.

Riser pipes in the container spray system

There are two separate riser pipes to spray the drywell in case the ordinary spray system is unavailable. The riser pipes are also utilized for water filling of the reactor containment. Water supply is available either from the fire water system using a direct driven diesel water pump or by fire trucks using existing connections to the riser pipes.

An evaluation of the riser pipes ability to supply water for spraying and filling of the reactor containment has been carried out. Corresponding riser pipes for Oskarshamn 3 are evaluated in a separate report. However these pipes are installed with the same prerequisites as for Oskarshamn 1 and 2 (e.g ground response spectra according to Regulatory Guide 1.60 scaled to PGA values of 0.15g horizontal and 0.1g vertical) and the results are therefore judged to be applicable for also for these two reactors. The basis for the evaluation is existing analysis of the pipes against DBE. The spectra used for DBE is then uprated with a factor of 4 in the analysis.

According the analyses the riser pipes are judged to maintain their structure and the pipe supports will fulfill their function. The judgment based on the calculations is that the riser pipes can fulfill their function in case of an earthquake with frequency level 10^{-7} per year.

Containment isolation provision

To ensure the containment integrity closure of containment isolation valves must be performed. An assumption reasonable when evaluating an earthquake with frequency level 10^{-7} per year is that no single failure has to be postulated. Also local manually closure of valves is reasonable to credit in case electrical power supply or pneumatic gas-supply is assumed to fail. An approached used when evaluating different types of containment isolation valves (e.g., motor operated gate valve) ability to close is that one or more representative valves in each category is evaluated. If the valves are judged to be able to fulfill their function also other valves in that category are judged to fulfill their function.

In a separate report valves have been evaluated for earthquake 10^{-7} per year. The basis for the evaluation is existing analysis of the valves against DBE. The spectra used for DBE is then uprated with a factor of 4 in the analysis.

Based on calculations for the different valve types, engineering judgment indicates that containment isolation is capable of being achieved by closure of isolation valves in case of an earthquake with frequency level 10^{-7} per year.

Pressure relief pipe between reactor containment and scrubber building

The pipe between the reactor containment and MVSS allows a controlled pressure relief of the reactor containment. An evaluation of the pipe has been carried. The basis for the evaluation is existing analysis of the pipe against DBE. The spectra used for DBE is then uprated with a factor of 4. According to the evaluation the pipe is judged to maintain its structure and the pipe supports will fulfill their function.

The judgment based on the evaluation is that the pipe can fulfill its function in case of an earthquake with frequency level 10^{-7} per year. However, it is reasonable to perform further evaluations of the pipe. This pipe is very important to fulfilling the requirements regarding release of radioactive nuclides to the society and to the environment in case of a core meltdown.

Scram system

The scram valves in the scram system are of pneumatic type, which opens in case of loss of air supply. The magnetic valves are provided with close circuit current hence the scram valves will open in case of loss of electrical power supply. An evaluation of the hydraulic insertion of the control rod is performed in a separate report. Summarized the following issues have been evaluated:

- The effect on the scram system. The system is judged to be able to fulfill its function.
- The effect on core support structures and elements. The relevant parts are judged be able to fulfill their function.
- The effect on the fuel. The fuel is judged to maintain its integrity and fuel channel deformations are not judged to jeopardize control rod insertion.

The evaluation is based on analysis results for an earthquake with frequency level 10^{-5} per year, existing margins and up-scaled analysis results to 10^{-7} per year earthquake, etc.

Based on the results in the separate report, engineering judgment indicates that the hydraulic insertion of control rods is also capable of being accomplished in case of an earthquake with frequency level 10^{-7} per year. A subcritical condition is therefore assured.

Reactor coolant pressure boundary

The main recirculation loops have been evaluated in a separate report. The basis for the evaluation is existing analysis of the loops against DBE. The spectra used for DBE is then uprated with a factor of 4. The highest utilization ratio for the main pipes of the loops is approximately 0.85 (against ASME III Service Limit Level D) due to loads resulting from an earthquake with frequency level 10^{-7} per year. However, some smaller diameter pipes connected to the main pipe have a utilization ratio above 1. According to the report these pipes, with utilization ratios above 1, are judged to maintain their integrity based on empirical results.

The piping containment penetrations are stronger than the connecting pipes. Engineering judgment indicates that margins until a pipe break may occur are significant. On the basis of the results it is likely that the integrity of the RCPB will be maintained in case of an earthquake with frequency level 10^{-7} per year.

Depressurization

It is assumed that the seismic qualified motor operated valves (MORV) in the pressure relief system open, thereby ensuring pressure relief of the reactor pressure vessel (RPV) and avoiding core meltdown through the RPV at high reactor pressure. This assumption should be confirmed by e.g. a SMA evaluation against the earthquake with frequency level 10^{-7} per year.

Conclusion

Based on the results above, the following conclusions are made in case of an earthquake with frequency level 10^{-7} per year:

- The hydraulic insertion of control rods is capable of being accomplished. A sub-critical condition is therefore assured.
- The margins until a pipe break may occur are significant based on empirical results. Therefore, the integrity of the RCPB is judged to be maintained.
- Containment isolation is capable of being achieved by closure of isolation valves.
- The integrity of the reactor containment and the scrubber building are likely to be preserved.
- The pipe between the reactor containment and MVSS that allows a controlled pressure relief of the reactor containments capable of fulfilling its function.
- The riser pipes used to supply water for sprinkling and filling of the reactor containment are capable of fulfilling their function.

The summarized judgment is that a severe beyond design basis earthquake with frequency level of 10^{-7} per year at the site does not prevent the mitigation systems from fulfilling their functions. If the severe earthquake is postulated to cause a core meltdown, the event is therefore in principle similar to the design sequence station blackout.

Spent fuel pool cooling

An evaluation has been performed of the integrity of the spent fuel pool in case of a water temperature of 100°C. According to the report the concrete pool walls are investigated by using an approximate approach reducing the cross section capacity due to temperature gradient. It is concluded from this study that it is reasonable to assume the concrete structure will have sufficient integrity even if its capacity is exceeded by a small amount. The steel liner at the inside of the pool structure is vulnerable for compression stress as the thin steel plate tends to buckle.

Engineering judgment indicates that the integrity of the spent fuel pool is likely to be preserved at a water temperature of 100°C. The required feed-and-bleed demand depends on the operational state. The most demanding is during outage condition. In case all other pumps are assumed to be unavailable, the spent fuel pool cooling can be accomplished by feed-and-bleed using fire trucks. Since no damage to the fuel occurs the release of radioactivity to the environment will be extremely low. In general it is essential that pipes connected to the spent fuel pool maintain their integrity to prevent loss of fuel pool water. In the performed evaluation a 3600 mm long DN100 pipe connecting two parts of the spent fuel pool is analyzed for Oskarshamn 1. The analysis of this pipe shows that a 10^{-7} earthquake increases the utilization ratio marginally.

A possible improvement would be to install new pipes to provide fire water to the spent fuel pool. As a suggestion these pipes should be capable of being provided with water in the same manner that is used for sprinkling and water filling of the reactor containment with the mitigation systems. Such an installation would minimize manual measures required to establish feed-and-bleed of the spent fuel pool.

2.B.2.3 Earthquake exceeding the design basis earthquake for the plants and consequent flooding exceeding design basis flood

The site is located at the peninsula Simpevarp. Within the surrounding landscape there are no rivers that may flood the site. Neither are there any lakes or large water reservoirs at high level that could flood the site. The phenomenon tsunami is not relevant because of the site location at the west coast of the Baltic Sea. The area is seismically calm and the Baltic Sea is too shallow for a tsunami to be created. In case of a severe beyond design basis earthquake the normal average sea water level in the Baltic Sea is not assumed to be higher than the calculated sea water level with recurrence frequency 10^{-2} per year which according to SAR corresponds to 1.18 m above normal average sea water level. Therefore in case of a seismically induced flooding, for example due to rock cracks or cracks in the cooling water channel, the water level that may affect the reactor building from the outside is lower than the design water level in the shaft were the reactor building is erected.

The largest water content that could cause an internal flooding of the plant, beyond what is assumed at DBE, is either the water in the fuel storage pools or the condensation pool. However, according above presented evaluations the integrity of the fuel storage pools and the reactor containment are likely to be preserved in case of a severe beyond design basis earthquake with frequency 10^{-7} per year.

2.B.2.3.1 Results from the licensee assessments of protection against earthquakes

The plants are expected to withstand a DBE. However, it is noted that the central control room (CKR) of Oskarshamn 1 is not designed to withstand a DBE. There is a risk that the ceiling in the building will endanger the operator safety. This item is treated in a special project and will be reported by the end of 2012.

It is also estimated that with some probability the plant can be brought to safe mode up to a 2.5 xDBE earthquake. Furthermore, it is also expected that the mitigation systems should be able to fulfill its intended function after a 10^{-7} earthquake.

2.B.2.4 Measures which can be envisaged to increase robustness of the plant against earthquakes

A possible improvement would be to install new pipes to provide fire water to the spent fuel pool. As a suggestion, these pipes should be capable of being provided with water in the same manner as is used for sprinkling and water filling of the reactor containment with the mitigation systems. Such an installation would minimize manual measures required to establish feed-and-bleed of the spent fuel pool in case all spent fuel pool cooling systems are inoperable.

The integrity assessments of the reactor containment, scrubber building and spent fuel pools are based on rough up-scaling calculation methods and engineering judgments on a best estimate basis due to the limited time available for this study. Therefore, it may be reasonable to carry out further investigations regarding the structural integrity of the reactor containment, scrubber building and fuel storage pools. The pipe between the reactor containment and the multi venturi scrubber system (MVSS) that allows a controlled pressure relief of the reactor containment should be evaluated further. The function of the pipe is very important to fulfilling the requirements regarding release of radioactive nuclides to the society and to the environment in case of a core meltdown.

Further investigations may be performed to evaluate the margins of structures, systems and components against ground motions exceeding DBE. Such investigations should emphasize evaluating margins to reach a safe shutdown condition. A reasonable approach would be to evaluate only one sub division in the systems required to bring the reactor to a safe shutdown condition. In general the designs of the sub divisions in required systems are similar.

The Screening Evaluation Work Sheet (SEWS) applied in the SQUG assessments does not explicit consider aspects of seismic induced fire. Therefore a very limited SMA could be performed to cover this issue. As a suggestion, this can be done in the same manner at Oskarshamn 1 and 2 as for Oskarshamn 3. The SEWS applied for Oskarshamn 3 requires to check that the equipment is not subject to flammable materials or fire ignition sources..

2.C. Ringhals

2.C.1 Design basis

2.C.1.1 Earthquake against which the plants are designed

2.C.1.2 Characteristics of the design basis earthquake (DBE)

In the original design, seismic loads were not included as design bases since the loads from earthquakes were considered small. The robustness of the structures called on for other reasons were considered enough to withstand the possible earthquakes.

DBE based on RG 1.60

Seismic loads have been applied for the Containment Filtered Vents (CFVs) and mobile units for water and power supply. The CFVs were installed during the late 80's in accordance with a Swedish government decision.

DBE based on the Swedish 10^{-5} earthquake

Seismic events with a probability of 10^{-5} per year have been considered at plant modifications performed since the beginning of the 90's, concerning equipment important for reactor safety.

The plant shall be verified to withstand a seismic event by the end of 2013. As RLE (Review Level Earthquake) the 10^{-5} earthquake described above is used.

2.C.1.2.1 Methodology used to evaluate the design basis earthquake

DBE based on RG 1.60

The RG 1.60 spectra was intended to represent the probability level 10^{-5} and the PGA was therefore assessed with that probability level as a target. For frequencies exceeding about 10 Hz the agreement is rather good between this spectrum and the Swedish hard rock spectra where spectral values are expected to be exceeded at a frequency of 10^{-5} annual. For low frequencies the RG 1.60 spectra are associated with much lower probabilities. Within the frequency range 2-5 Hz (the range of fundamental frequencies of reactor buildings and containments) the RG 1.60 spectrum appears to correspond to the probability range of 10^{-8} – 10^{-6} annual events per site. The RG 1.60 spectra are thus conservative compared with the Swedish hard rock spectra for low frequencies that are fundamental for building structures.

Since the agreement between the RG 1.60 spectra and the Swedish hard rock spectra are rather good at high frequencies, the RG 1.60 spectrum is judged to be adequate for the CFV. At plant modifications concerning CFV, the envelope of the Swedish 10^{-5} earthquake and the original DBE based on RG 1.60 is applied.

DBE based on the Swedish 10^{-5} earthquake

In accordance with previous description the envelope ground response spectra are valid for a typical Swedish hard rock site. In general, hard Swedish rock is not unlike the rock sites in Eastern North America, where recent hazard studies yield response spectra that have a characteristic shape similar to the Swedish 10^{-5} hard rock spectra.

Since the DBE is based on Swedish seismological and geological conditions evaluated in a study performed in the early 90's, which is also supported by more recent hazard studies, the adequacy of the DBE is judged to be acceptable.

2.C.1.2.2 Conclusion on the adequacy of the design basis for the earthquake

DBE based on RG 1.60

Since the agreement between the RG 1.60 spectra and the Swedish hard rock spectra are rather good at high frequencies, the RG 1.60 spectra is judged to be adequate for the CFV. At plant modifications concerning CFV, the envelope of the Swedish 10^{-5} earthquake and the original DBE based on RG 1.60 is applied.

DBE based on the Swedish 10^{-5} earthquake

Since the DBE is based on Swedish seismological and geological conditions evaluated in a study performed in the early 90's, which is also supported by more recent hazard studies, the adequacy of the DBE is judged to be acceptable.

2.C.1.3 Provisions to protect the plants against the design basis earthquake

According to regulation SSMFS 2008:17 (14 §) issued by the Swedish Radiation Safety Authority the nuclear reactor shall be designed to withstand natural phenomena that can lead to a radiological accident. Earthquake is stated as one event, among several others, that should be considered. Work is currently on-going at Ringhals in order to fulfill this regulation. The plant shall be verified to withstand a seismic event by the end of 2013, thus to be able to reach a safe cold shut-down state using safety class equipment. As RLE (Review Level Earthquake) the 10^{-5} earthquake described above is used. Since the reactors at Ringhals are at various stages in this earthquake qualification process, the following subchapters are in some cases divided into subchapters based on the different reactors.

2.C.1.3.1 Identification of the key structures, systems and components

To satisfy Ringhals' safe shutdown performance criteria, the following safe shutdown functions are required:

- Reactivity control
- RCS Inventory control
- RCS Pressure control
- Residual heat removal
- Containment isolation and integrity
- Support functions

Ringhals 1

The key structures, systems and components which are needed for achieving safe shutdown state have been identified for Ringhals 1 and put together in a Safe Shutdown Equipment List (SSEL).

At Ringhals 1, a new physically and functionally separated and diversified reactor protection system called DPS (Diversified Plant Section) has been installed. The DPS works in parallel with the Original Plant Section, OPS. Each plant section contains redundant trains with functional separation and is thus singe failure proof. The purpose of the DPS is mainly to cope with the event fire but also with the events earthquake and lightning. The DPS includes power supply, process sensors, logics, process systems and buildings for execution of the safety functions, all independent of the OPS. The OPS and DPS can independently of each other handle the safety functions required for the safe shutdown performance criteria, with the exception of actuation of containment isolation that DPS has limited capability to handle. DPS only isolates the DPS systems and the main steam lines. There is no need for further isolation functions in DPS since the RCPB remain intact at the DBE and the containment function is not challenged.

The DPS is designed for the Swedish 10^{-5} earthquake and the main part of DPS is placed in buildings that are designed and verified for the Swedish 10^{-5} earthquake. The entire plant section OPS is conservatively assumed to be disabled during an earthquake. This conservative postulation is based on that everything not qualified for earthquake fails. In reality many systems and functions will probably function beyond the 10^{-5} earthquake since part of OPS is qualified for earthquake and due to the fact that other loads envelop the seismic loads in many cases.

Ringhals 1 has performed structural analyses for seismic loads of the Swedish 10^{-5} earthquake for the safe shutdown functions also in OPS. Structural mechanical verification is presented in the systems' status reports (FSAR Reference part, R300-series). The structural mechanical analyses forthe RCPB have been completed.

Since work is ongoing to verify the plants for earthquake some measures are remaining before Ringhals 1 is fully verified for earthquake. The additional seismic evaluation, that mainly concerns seismic interaction, is performed using calculations and experience-based methods. The experience-based method used is the Seismic Margin Assessment (SMA). Guidelines for performing the SMA are provided in EPRI NP-6041-SL.

Ringhals 2-4

Based on the systems required to perform the functions to reach a safe shutdown condition, Safe Shutdown Equipment Lists (SSEL) have been developed and used in the walkdowns that have been performed at Ringhals 2-4. Distribution systems, i.e. piping, ductwork and cable raceways, have been part of the walkdown. Such systems are reviewed on a plant wide basis rather than identifying and evaluating specific components. The results from the walkdowns are now being compiled.

At Ringhals 2 all reactor control and protection equipment including the control room have recently been replaced with seismically qualified components in a modernization project.

Since work is ongoing to verify the plants for earthquake some measures are remaining before Ringhals 2-4 are fully verified for earthquake.

The seismic evaluation is performed using calculations and experience-based methods. The experience-based method used is the Seismic Margin Assessment (SMA). Guidelines for performing the SMA are provided in EPRI NP-6041-SL.

The operator actions taken after an earthquake depend on the consequences of the earthquake. There are no specific operating procedures connected to an earthquake. In case the earthquake will impact the plant, actions will be taken according to the plant's operating procedure for the disturbance in question.

Ringhals 1 and Ringhals 3 have earthquake monitoring systems. The accelerographs are placed in the reactor building for Ringhals 1 and in the CFV building for Ringhals 3. Alarm is given in the respective control room when PGA 0.01 g is exceeded. Ringhals 1 has an instruction for how the operators should handle the alarm from the earthquake monitoring system. The alarm will be registered and reported to the measurement department at Ringhals. The plant has no OBE (Operating Basis Earthquake) defined, so an alarm from the earthquake monitoring system will have no direct consequences for the operation of the plant.

The CFV is designed to withstand the DBE based on RG 1.60. The CFV and the mobile units for water and power supply were designed to be completely autonomous systems and structures thus as little sensitive as possible to consequential damages from an earthquake. In addition seismic interaction has been considered and for instance an evaluation of nearby building parts has been performed.

The description below is valid for the DBE based on the Swedish 10^{-5} earthquake.

Ringhals 1

In the design of DPS, seismic interaction concerns were considered as a design requirement. Verification is on-going and included in the walk downs. In order to avoid seismic interaction concerning I&C and electric power, the earthquake qualified plant section DPS is functionally separated from the plant section OPS that is not qualified for earthquake. Systems belonging to RCPB and non-isolating connecting systems including check valves and isolation valves will maintain mechanical integrity and active function at a seismic event, which are verified in the systems' status reports (FSAR Reference part, R300-series) and will thus not cause a flooding at the DBE.

In the analyses, fire is assumed to occur in plant section OPS as a consequential effect of the earthquake. Fire is not assumed to take place in DPS that is designed to withstand the Swedish 10^{-5} earthquake. A fire in OPS is an acceptable consequence of the earthquake.

Ringhals 2-4

As described earlier an evaluation of the plants is being performed using Seismic Margin Assessment (SMA). The walkdowns are being performed according to guidance in EPRINP-6041-SL. According to this guide the walkdowns should consider failure of SSCs not designed to withstand the RLE that can cause damage of SSCs that need to remain available.

Piping was part of the distribution system walkdown, asmentioned in section 3.2.1.2. Such systems are reviewed on a plant wide basis rather than identifying and evaluating specific components. The piping systems for Ringhals 2, 3 and 4 were combined and the bounding cases were selected for limited analytical review. The equipment required for safe shut down has been defined also considering events such as fire, steam release outside containment and flooding. However, there are a number of pipe segments in non-safety class piping, where pipe breaks might cause flooding of multiple safety class pump rooms with consequential core damage in case no manual action is taken. These segments need further investigation regarding seismic durability.

Design basis scenario for the CFV is station blackout with loss of all auxiliary feedwater which includes loss of external electrical power supply from outer grid during 24 hours. The battery secured net that is providing power to instrument and control equipment is assumed functional.

Loss of external power supply is also assumed in the analysis of earthquake performed within Ringhals' ongoing earthquake qualification process.

2.C.1.4 Compliance of the plants with its current licensing basis

2.C.1.4.1 Licensee's general process to ensure compliance (e.g., periodic maintenance, inspections, testing);

It is of utmost importance that systems important to safety are operable at all times. To ensure the operability there are periodic testing enjoined in the technical specifications for the systems important to safety.

Periodic maintenance is carried out at Ringhals. Preventive maintenance is carried out in order to reduce the probability of fault or degradation to maintain the required function. The preventive maintenance can be divided into condition based maintenance and predetermined maintenance (overhaul). The condition based maintenance is used in order to supervise components and systems in operation. Examples of method used for the condition based maintenance are vibration measurement, pressure-, temperature- and flow measurement, oil analysis and motor current analysis. The predetermined maintenance is conducted according to a determined time schedule or after a specific operating time. Corrective maintenance is carried out after faults have been identified in order to restore the required function.

No off-site mobile equipment/supplies are credited in emergency procedures due to a DBE. The plant has therefore no process to ensure that such equipment/supplies are available and remain fit for duty.

2.C.1.4.2 Any known deviation, and consequences of these deviations in terms of safety; planning of remediation actions

DBE based on RG 1.60

There are no known deviations for the CFVs.

DBE based on the Swedish 10^{-5} earthquake

Work is on-going to fulfill the regulation SSMFS 2008:17 (14 §) regarding earthquake. The plant shall be verified to withstand a seismic event by the end of 2013. There are thus known deviations and work is on-going to identify and take care of all deviations in order to fulfil the regulation SSMFS 2008:17 according to the time schedule. The seismic evaluation is performed using calculations and experience-based methods. The experience-based method used is the Seismic Margin Assessment (SMA). Guidelines for performing the SMA are provided in EPRI NP-6041-SL. The basic procedural steps of a seismic walkdown are to:

- Apply walkdown screening criteria in EPRI NP-6041-SL to the equipment in the Safe Shutdown Equipment List (SSEL).
- Define possible failure modes (e.g., functionality, structural integrity, or anchorage failure) of the SSEL items that do not meet the walkdown screening criteria and identify what further evaluations are required.
- Identify equipment or structures that are not included in the SSEL but whose structural failure may impact the nearby SSEL items (i.e., seismic interaction concerns).
- Observe and record any noted seismic deficiencies.
- Document the walkdown observations.

Ringhals 1

In general seismic interactions concerns are remaining before Ringhals 1 is fully verified for earthquake. The seismic interaction concerns are mainly handled by the SMA. Some work is also remaining to complete the earthquake verification of buildings and structures. Earlier rough evaluations of the reactor building indicate that the roof of this building can be a weakness. Refined analyses are therefore necessary in order to investigate if reinforcement of the reactor building's roof may be necessary. In case the roof cannot withstand the 10^{-5} earthquake, roof elements can fall down into the spent fuel pool. This can cause damages to the fuel and endanger the possibilities for external cooling.

Ringhals 2-4

The majority of the equipment in the SSEL was judged to be seismically adequate. However, the Seismic Review Team (SRT) identified some deficiencies. The main concern was insufficient anchorage of equipment. The relays at Ringhals 3 and 4 are housed in standard ASEA type VSG cabinets, whose anchorage was found to have insufficient seismic capacity.

Given that the anchorage is modified, then capacity/demand screening of ASEA control relays indicates that all relays except the ASEA model RXMA2 can be screened as adequate for the DBE.

At Ringhals 2 the major part of instrumentation and control components including the control room have been replaced with seismically qualified equipment. Therefore the I&C at Ringhals 2 could be screened out. There was, however, one exception. Similar relays and cabinets as described above for Ringhals 3 and 4 are present in the diesel control room, and therefore reported as a deficiency.

Another identified deficiency is the control room ceiling at Ringhals 3 and 4. The primary concern is questionable capacity within the support details. Fans on vibration isolators were also identified as a deficiency on all 3 units.

The primary concern is questionable seismic capacity of the vibration isolators. A few deficiencies in the distribution systems (piping, ductwork and cable raceways) were identified, mainly related to the anchorage of the distribution systems.

Some work is also remaining with completion of the earthquake verification of buildings and structures.

2.C.1.4.3 Specific check following Fukushima NPP accident

Ringhals has responded to the WANO SOER 2011-2. No gap concerning earthquake was identified.

Safe shutdown at the Swedish 10^{-5} earthquake

Ringhals 1

As discussed earlier DPS, that is qualified for the Swedish 10^{-5} earthquake, can independently of OPS bring the plant to safe shutdown. The measures remaining before Ringhals 1 is fully verified for earthquake is mainly seismic interaction concerns. Based on this it is judged that Ringhals 1 can reach safe shutdown in case of a 10^{-5} earthquake.

Ringhals 2-4

Based on the results from the SMA for Ringhals 2 and considering that the major part of instrumentation and control components including the control room have been replaced with seismically qualified equipment it is judged that Ringhals 2 can reach safe shutdown in case of a 10^{-5} earthquake.

Based on the results from the SMA for Ringhals 3 and 4 and also considering that the Swedish 10^{-5} earthquake corresponds to intensity VI of the modified Mercalli scale according to section 2, it is also judged that Ringhals 3 and 4 can reach safe shutdown in case of a 10^{-5} earthquake.

2.C.2 Evaluation of safety margins

2.C.2.1 Range of earthquake leading to severe fuel damage

As discussed earlier, seismic qualification is on-going at Ringhals. As RLE the Swedish 10^{-5} earthquake is used. Measures are remaining for all four units before they are fully qualified for the 10^{-5} earthquake. Since there are deviations for the 10^{-5} earthquake, an evaluation of the range of earthquake severity above which loss of fundamental safety functions or severe damage to the fuel becomes unavoidable is not performed for an earthquake causing ground motions exceeding the 10^{-5} earthquake.

Safe shutdown for the Swedish 10^{-5} earthquake is discussed earlier.

However, in this section a more detailed evaluation has been performed of the mitigation systems ability to fulfil its function at a severe beyond design basis earthquake corresponding to an exceedance frequency of 10^{-7} per site and year. In addition the spent fuel pools integrity and the ability to ensure spent fuel pool cooling in the long term is evaluated in the following section. The approach to focus on more detailed assessments of those system functions is reasonable. If the mitigation systems and the spent fuel pool integrity and the spent fuel pool cooling are judged to be ensured in case of an earthquake with frequency level 10^{-7} per year, the margins are considered to be adequate.

The evaluated earthquake is considerably more severe than the DBE. By ensuring the mentioned functions the consequences to the environment is judged to be acceptable and in accordance with the requirements in the government decision regarding release of radioactive nuclides to the society and the environment in case of a core meltdown.

2.C.2.2 Range of earthquake leading to loss of containment integrity

Range of earthquake the plant can withstand without losing confinement integrity is discussed hereunder.

The assessment is emphasized to evaluate the mitigation systems ability to perform its function. The approach when analysing the margins of earthquake ground motions beyond the Swedish 10^{-5} earthquake is as follows:

- The assumed earthquake corresponds to a severe beyond design basis earthquake, were the spectral values represent limits of single degree-of-freedom responses that are not expected to be exceeded with a 80 % confidence at the frequency of 10^{-7} per year, which is the strongest seismic level developed for Swedish conditions. The peak ground acceleration (PGA) is 0.41 g. Compared to the Swedish 10^{-5} earthquake the PGA value is about 4 times higher. The frequency of such an earthquake corresponds to events classified as residual risk and the spectral values are considerably beyond the values for the Swedish 10^{-5} earthquake.
- Evaluate the mitigation systems ability to fulfil its function. The mitigation systems ensure pressure relief of the reactor containment and a limited release of radioactive nuclides to the environment in case of a core meltdown. The requirement concerning release of radioactive nuclides is that less than 0.1 % of the core inventory of the isotopes cesium-134 and cesium-137 in a reactor core with a thermal power of 1800 MW should be released, with the prerequisite that other nuclides of importance concerning ground contamination are separated in the same proportion as the cesium isotopes.

- Evaluate the integrity of the spent fuel pools and ability to ensure spent fuel pool cooling in the long term.
- The initial condition is full power operation, since this initial condition will give rise to the most severe consequences in case of loss of confinement integrity.

Mitigation systems

The mitigation systems that ensure limited release of radioactive nuclides in case of core meltdown consist of the following system functions:

- Redundant water supply to the containment. Water supply is available from the mobile unit for water and power supply.
- Controlled pressure relief of the reactor containment is performed via the MVSS (Multi Venturi Scrubber System) connected to the reactor containment with piping equipped with rupture disc and valves.
- Non-filtered pressure relief of the reactor containment at LOCA in combination with loss of pressure suppression function. (Only for BWR.)

As stated in above the design basis earthquake for the above severe accident systems is the RG 1.60 earthquake, which envelopes the Swedish 10^{-5} and 10^{-7} earthquakes at frequencies below 10 Hz and 4 Hz respectively. The mobile redundant water supply is not sensitive to high frequencies due to the damping in the vehicle suspension system. The CFV buildings and the pressure relief systems are passive systems, with the exception of the isolation valves of the Ringhals 1 non-filtered pressure relief. These structures are mainly sensitive for large displacements and not for high frequencies with low displacements. Based on the above it is judged that the CFV systems will sustain also a Swedish 10^{-7} earthquake, with the possible exception of the above-mentioned valves.

The containment isolation provision is evaluated because of its importance to ensure the required limited release of radioactive nuclides. Also other systems credited at severe accidents are evaluated.

Ringhals 1

Containment

In a separate report the structural integrity of the containment building at Ringhals 1 is verified for the DBE (Swedish 10^{-5} earthquake). A capacity check of how the structures in the containment building withstands an earthquake has not been performed before. Capacity checks are therefore performed both for an earthquake with frequency level 10^{-5} and with the frequency level 10^{-7}.

The analysis assumptions and capacity controls are done in accordance with ASCE and the Swedish handbook BBK04. For the case of an earthquake with frequency level 10^{-7} Eurocode 8 is also applied.

The capacity checks for the containment building show that the DBE load with the probability level 10^{-5} is well within the limits of the structures. The analysis with a more severe loading, with probability level 10^{-7}, is based on rough up-scaling calculation methods and engineering judgments. The main conclusions from the simplified analysis with an earthquake with probability level 10^{-7} are that it is reasonable to assume that the structures have sufficient integrity to preserve the confinement of radioactivity. This should be seen as an engineering judgment since there are some uncertainties in how non-linear response can be treated in the analysis.

CFV building

The CFV building was originally designed against earthquake induced ground motions based on US NRC Regulatory guide 1.60 scaled to a PGA of 0.15 g horizontally and 0.10 g vertically.

An assessment of the structural integrity of the CFV building is performed for the earthquake with frequency level 10^{-7} per year. The analysis assumptions as well as the load combination are done in accordance with ASCE and Eurocode standards.

The results show that the structural elements in the CFV building at Ringhals 1 have the capacity to withstand the 10^{-7} earthquake, but there are some uncertainties in how non-linear response can be treated in the analysis. The displacements in the structural parts of the building were checked and they were less than 20 mm for all parts, which is considered acceptable.

An engineering judgment is that local damages may appear, but the integrity of the CFV building will be preserved in case of an earthquake with frequency level 10^{-7} per year.

Containment isolation

To ensure the containment integrity, closure of containment isolation valves must be performed. Containment isolation is performed when signals are received and the valves are provided with required power supply in order to close.

A reasonable assumption when evaluating an earthquake with frequency level 10^{-7} per year is that no single failure has to be postulated. The screening of the isolation valves has been made based on the isolation valve information in a separate report.

Ringhals 1 is equipped with double inner check valves for the major release paths from the reactor vessel. For active closing of the remaining release paths DPS has limited capability of containment isolation, while OPS is fully capable of this. OPS is not qualified for earthquake while DPS is designed for the 10^{-5} earthquake.

Inspections and analyses performed in the 90's have however assessed that most isolation valves will manage the 10^{-7} earthquake. An engineering judgment based on this is that it is likely that at least the fail-safe functions will remain intact.

Assuming that single failure must not be applied for the 10^{-7} earthquake, the required penetration configuration types are deemed to function. Closed piping loops outside the containment do not require closing as long as the structural integrity in the loop outside containment remains intact. The piping penetrating the containment is qualified for the 10^{-5} earthquake and is robust. Structural mechanical verification is presented in the systems' status reports (FSAR Reference part, R300-series). An engineering judgment based on this is that it will also manage the 10^{-7} earthquake.

The containment isolation function is, based on the discussions above, judged to be sufficiently functional at the 10^{-7} earthquake.

Reactor shutdown

The reactor shutdown function (pneumatic control rod insertion) is actuated by I&C qualified for the Swedish 10^{-5} earthquake. Scram is executed electrically both from the OPS and the DPS. Further, the design is fail-safe in the aspect that scram is executed automatically at loss of power for the scram systems. Scram is a battery backed DPS function deemed to be robust.

Evaluation of the structural integrity of the scram system piping has been performed for the 10^{-7} earthquake. Based on the results the piping system is judged to fulfil its function.

However, further evaluations, particularly regarding the supports of the scram water and gas tanks, are needed before the scram function is verified for the 10^{-7} earthquake.

RCPB integrity

The pressure relief system of Ringhals 1 contains 20 Safety Relief Valves (SRV), 10 Power Operated Relief Valves (PORV) and 2 control valves. All functions are fail-safe closed.

The six PORVs and two control valves controlled by DPS are designed and qualified for electrical opening at the 10^{-5} earthquake. The remaining four P ORVs are controlled by the OPS, and are thus not qualified for earthquake. RCPB has been verified for the 10^{-5} earthquake. Structural evaluations have also been made of RCPB at the 10^{-7} earthquake.

Even though the results in some parts shows a utilization ratio above 1, more detailed analysis will likely show acceptable values. Therefore RCPB is judged to keep its integrity at the 10^{-7} earthquake.

Depressurization of the reactor vessel to avoid high pressure meltthrough

Depressurization can be executed by electrical opening of PORVs or the two control valves. The PORVs and control valves are qualified for the 10^{-5} earthquake. In the safety analyses it is assumed that the RCPB is manually depressurized before melt-through of the RCPB. This must be done before the batteries cease to provide power to the pressure relief valves. Both OPS and DPS have the ability to depressurize the reactor vessel. DPS is qualified for the 10^{-5} earthquake and the battery backed DPS function is robust. Based on this and the margins in design analyses, an engineering judgment is that the function will be preserved in case of an earthquake with frequency level 10^{-7} per year.

Ringhals 2-4

Containment

The structural integrity of the containment building at Ringhals 4 is verified for the DBE (Swedish 10^{-5} earthquake). The main difference between Ringhals 3 and Ringhals 4 is in the design of the steam generator towers. This difference has been considered in in a separate report, therefore this study is also valid for Ringhals 3. The containment building at Ringhals 2 is judged to be as rigid as or more rigid than the containment building at Ringhals 4. A capacity check of how the structures in the containment building withstand an earthquake has not been performed before. Capacity checks are therefore performed both for an earthquake with frequency level 10^{-5} and with the frequency level 10^{-7}. The analysis assumptions and capacity controls are done in accordance with ASCE and the Swedish handbook BBK04. For the case of an earthquake with frequency level 10^{-7} Eurocode 8 is also applied.

The capacity checks for the containment building show that the DBE load with the probability level 10^{-5} is well within the limits of the structures. The analysis with a more severe loading, with probability level 10^{-7}, is based on rough up-scaling calculation methods and engineering judgments. The main conclusions from this analysis are that it is reasonable to assume that the structures have sufficient integrity to preserve the confinement of radioactivity. This should be seen as an engineering judgment since there are some uncertainties in how non-linear response can be treated in the analysis.

CFV buildings

The CFV buildings were originally designed against earthquake induced ground motions based on US NRC Regulatory guide 1.60 scaled to PGA of 0.15 g horizontally and 0.10 g vertically, see section 3.1.1.1. The CFV buildings at Ringhals 2 - Ringhals 4 are almost the same. The only thing that differs is that the CFV building at Ringhals 2 is founded entirely on compact gravel while the CVF buildings at Ringhals 3 and Ringhals 4 are founded entirely on the bedrock.

An assessment of the structural integrity of the CFV buildings has been performed for the earthquake with frequency level 10^{-7} per year. The analysis assumptions as well as the load combination are done in accordance with ASCE and Eurocode standards. The results show that the structural elements in the CFV building at Ringhals 2 have the capacity to withstand the 10^{-7} earthquake, but there are some uncertainties in how non-linear response can be treated in the analysis. An engineering judgment based on the results is that local damages may appear, but the integrity of the CFV building will be preserved in case of an earthquake with frequency level 10^{-7} per year.

The analyses of the CFV buildings at Ringhals 3 and 4 show that, without taking the non-linear response into account, these buildings can withstand the 10^{-7} earthquake. Displacements in the structural parts of the buildings were checked and they were less than 20 mm for all parts, which is considered acceptable.

Riser pipes in containment spray system

Piping for redundant water supply to containment consists of two pipelines each one connected to one containment spray header.

An evaluation of the raiser pipes ability to supply water for sprinkling and filling of the reactor containment at Ringhals 2 has been carried out.

The system is similar at Ringhals 3 and 4. Even though the results in some parts shows a utilization ratio above 1, more detailed analysis will likely reduce the utilization ratio considerably. Therefore the piping system is judged to keep its integrity at the 10^{-7} earthquake.

Containment isolation

To ensure the containment integrity, closure of containment isolation valves must be performed. Containment isolation is performed when signals are received and the valves are provided with required power supply in order to close.

The PWR units are similar in this respect and Ringhals 4 is used as an example below.

At Ringhals 4 there are 35 motor operated valves. The number of these valves that receive containment isolation signal and shall change position from normal operation to post LOCA is 13, which can be summed up to 7 containment penetrations that are equipped with 1 or 2 valves.

There are 44 air operated containment isolation valves at Ringhals 4. The number of these valves that receive a containment isolation signal and shall change position from normal operation to post LOCA is 20. (All of these go to the safe position in case of inadvertent actuation failure.)

As specified above there are seven containment penetrations that are equipped with 1 or 2 motor operated isolation valves, which should change position at a containment isolation signal. However, for the related piping systems, some of the pipes penetrating the containment are closed inside containment and some have check valves as inside isolation valves. A reasonable assumption when evaluating an earthquake with frequency level 10^{-7} per year is that no single failure has to be postulated. If this is considered, there is only one penetration that will have an unisolated flow path in case of an isolation power failure, namely the seal water return line. However, a comparison with a similar relief scenario in the probabilistic safety analyses reveals that the releases at a severe accident via the seal return line are within the acceptance criterion even if the effect of the auxiliary building ventilation filters are neglected. The main reason is the small equivalent area of the seal return release path.

Systems required to prevent by-pass of the reactor containment

Containment isolation is required to accomplish filtered vent of the containment at severe accidents. However there are additional possible release paths. Thus the following systems/equipment have been evaluated from a seismic point of view.

Piping systems that form a part of the reactor containment and do not have any containment isolation valves:

- Main steam piping inside containment
- Main feedwater piping inside containment
- Auxiliary feed water piping inside containment (Ringhals 3/4)
- SG tubing

In a separate report the main steam, main feedwater and auxiliary feedwater piping inside containment at Ringhals 4 are evaluated. For the main feedwater piping the utilization ratio is lower than 1. For the main steam and auxiliary feedwater piping the utilization ratio is somewhat higher than 1. However, more detailed analysis is deemed to show acceptable values.

The SG tubing at Ringhals 4 has been evaluated for DBE. Based on the margins, the SG tubing is judged to fulfil its function in case of an earthquake with frequency level 10^{-7} per year.

The evaluation above is made for Ringhals 4, but is judged to be representative also for Ringhals 2 and 3.

Components which are pressurized by the RCS and give a direct release path from the RCS to the environment in case of failure:

- Check valves in low head SI (8" and 10")
- Isolation valves between RCS and suction side of RHR (12" motor operated gate valves)

The 8" and 10" check valves in the low head safety injection lines connects to the RCS cold legs and are closed during normal operation. There are three valves in series in each line. They are specified for the seismic requirement 9.1 g in all directions. This is well above the maximum acceleration at a 10^{-5} earthquake at the location of the valves closest to the cold legs which is 4 g.

The 12" motor operated gate valves connects to the RCS hot legs. There are two valves in series in each of the two lines. The valves are closed during normal operation and the breakers are opened. It is shown in a separate report that the valves will maintain their integrity and operability for seismic inertia loads equivalent to 4.5 g in any direction. The maximum acceleration at the hot legs is 2.5 g at a 10^{-5} earthquake.

The evaluation above is made for Ringhals 3 and Ringhals 4, but is judged to be representative also for Ringhals 2.

The conclusion is that the components that could give a direct release path from the RCS to the environment are qualified for earthquake loads, which implies an inherent ruggedness and are therefore judged to maintain their integrity in case of an earthquake with frequency level 10^{-7} per year.

Systems other than CFV systems credited at severe accidents

In the deterministic safety analyses of severe accidents a number of components or systems in addition to the CFV systems are credited. For a successful mitigation of the severe accident, these systems should perform their required function also during the accident in combination with an earthquake.

The analyzed severe accident events are loss of all power including all auxiliary feedwater pumps and the same event in combination with a 292 mm diameter cold leg LOCA. The following functions are credited:

- Reactor trip
- Pressurizer safety valve relief
- Battery back-up power

A reactor trip is required for reactivity control at an initiating event.

At Ringhals 2, I&C protection and control equipment as well as transmitters, cables and reactor trip breakers have been replaced by seismically qualified equipment (DBE level).

At Ringhals 3 and 4 a seismically qualified (DBE level) diversified protection system has been installed during RA11 including reactor trip breakers. The diversified protection system uses only steam generator low level as the trip signal.

Another important component for the reactor trip function is the Control Rod Drive Mechanism (CRDM). The Ringhals CRDMs are free standing without the lateral support usually found in seismically designed plants. CRDMs at Ringhals 4 are verified to fulfil the criteria related to the DBE.

However, preliminary rough elevations of the earthquake with frequency level 10^{-7} per year indicate that it will be difficult to show that the function will be preserved.

The conclusion is that it is very likely that reactor trip will occur at the DBE but at the earthquake with frequency level 10^{-7} per year it is difficult to show that the function will be preserved.

At the limiting severe accident event the pressure in the RCS will increase due to lack of cooling of the core residual heat and the pressurizer safety valves should open when the set point is reached. Thus the function of the safety valves is required.

The Crosby valves used at Ringhals 2-4 have been verified for 7.1 g in the horizontal direction and 6.0 g in the vertical. This implies an inherent ruggedness and it is judged to fulfil its function in case of an earthquake with frequency level 10^{-7} per year.

Ordinary battery backup power at severe accident is required for reactor trip and containment isolation. The batteries and their supports are verified for the DBE for Ringhals 2, 3 and 4. Based on this and the margins in design analyses it is judged that the function will be preserved in case of an earthquake with frequency level 10^{-7} per year.

Spent fuel pool integrity and cooling

Ringhals 1

Spent fuel pool

In a separate report the structural integrity of the spent fuel pool at Ringhals 1 is verified for the DBE (Swedish 10^{-5} earthquake). A capacity check of how the structures in the fuel pool withstand an earthquake has not been performed before. Capacity checks are therefore performed both for an earthquake with frequency level 10^{-5} and with the frequency level 10^{-7}. The analysis assumptions and capacity controls are done in accordance with ASCE and the Swedish handbook BBK04. For the case of an earthquake with frequency level 10^{-7}, Eurocode 8 is also applied.

The capacity checks for the fuel pool show that the DBE load with the probability level 10^{-5} is well within the limits of the structures. The analysis with a more severe loading, with probability level 10^{-7}, is based on rough up scaling calculation methods and engineering judgments. The main conclusions from the analysis with an earthquake with probability level 10^{-7} are that the structures have sufficient integrity to preserve the confinement of radioactivity and maintain the required minimum water level to cover the fuel elements. This should be seen as an engineering judgment since there are some uncertainties in how non-linear response can be treated in the analysis.

The roof of the reactor building can be a weakness for an earthquake with frequency level of 10^{-5}. For the more severe loading, with probability level 10^{-7}, current evaluation indicates that it will be difficult to show that integrity will be preserved.

Spent fuel pool cooling

In order to maintain spent fuel cooling in case of a seismic event the integrity of the fuel building, the spent fuel pools including the gates, the spent fuel storage racks, the fuel and the spent fuel pit cooling system including the cooling chain to the sea are required.

All equipment for supervision and cooling belong to plant section OPS. There are thus neither seismic qualification nor assessment present other than the fact that the equipment is of a robust design and that other loads envelop the seismic loads in many frequency ranges. The fuel support assembly is verified for the 10^{-5} earthquake. The verification is made towards ASME level D and includes conservative assumptions, for instance that damping from the surrounding water is neglected. Hence, an engineering judgment is that the fuel support assembly will manage the earthquake with frequency level 10^{-7} per year.

The gates have not been evaluated. At severe leakage the water level will sink to a few decimeter above the fuel elements.

Temporary feeding of the spent fuel pool can be provided by the fire protection system via fire hose. There is however no instruction for this action, and the fire protection system is not seismically qualified. If the water level in the spent fuel pool has sunk to a level where the radiation shielding is lost before the operators have been aware of the problem, it will also be extremely difficult to use this manual temporary solution due to the high radiation level in the reactor building that must be entered.

Ringhals 2-4

Fuel buildings

In a separate report the structural integrity of the fuel buildings at Ringhals 2, 3 and 4 for the DBE (Swedish 10^{-5} earthquake) is verified. The analysis assumptions as well as the load combination are done in accordance with ASCE and Eurocode standards. A capacity check of how the structures in the fuel buildings withstand an earthquake has not been performed before.

Capacity checks are therefore performed both for an earthquake with frequency level 10^{-5} and with the frequency level 10^{-7}.

The fuel buildings at Ringhals 3 and Ringhals 4 are identical. The fuel building at Ringhals 2 differs a bit from the other two. The differences are found in the roof construction and the foundation of the buildings. One other important difference between the fuel buildings at Ringhals 2 and Ringhals 3 and 4 is the amount of reinforcement in the upper part of the outside walls. The structure at Ringhals 3 and 4 only has approximately half the amount of reinforcement in comparison to Ringhals 2.

The capacity checks for the Ringhals 2, 3 and 4 building show that the DBE load with the probability level 10^{-5} is well within the limits of the structures. The analysis with a more severe loading, with probability level 10^{-7}, shows that the capacity utilization limit is exceeded for several elements, like the outer walls in the hall above level +115. Even if the capacity limits are reached this does not mean that the whole structure collapses. Most likely the result will be local damages of the structure. The displacements under the 10^{-5} loading can be considered marginal, while under the 10^{-7} loading they are significant. The deformations up to 16 mm practically mean that the expansion joint capacity of 20 mm is used entirely by the fuel building and raises issues in terms of collisions with the structures that are in direct contact with the building.

According to a separate report it can be concluded that the integrity of the fuel buildings will be preserved in case of an earthquake with frequency level 10^{-5} per year.

For the case of an earthquake with frequency level 10^{-7} the results show that there most likely will be local damages but that it is unlikely that the fuel buildings collapse. The structure of the fuel pools is robust and the local damages described above will not appear in these structures.

Spent fuel pool cooling

In order to maintain spent fuel cooling in case of a seismic event the integrity of the fuel building, the spent fuel pools including the gates, the spent fuel storage racks, the fuel and the spent fuel pit cooling system including the cooling chain to the sea are required.

The spent fuel pit cooling systems were not seismically qualified and no seismic margins evaluations based on walk-downs have been performed.

However there are a number of alternate ways to cool the fuel. Examples of those are:

- Alternate cooling on primary and secondary side of the spent fuel pit heat exchangers. To restore the circulation on the component cooling side of the spent fuel pit heat exchangers, mobile pump units can be used, and to restore the circulation on the primary side SF pump no 3 can be used. The mobile pumps are qualified for the RG 1.60 seismic event and have its own power supply. The mobile SF pump is located at Ringhals 3 fuel building but needs electric power for operation. The mobile SF pump is not seismically qualified.
- Feed and bleed with fire protection system or mobile unit for water and power supply. Feed and bleed of the pits with fire protection system has the advantage that the pumps are diesel driven and do not require electrical power. The fire protection system is not seismically qualified. This alternate spent fuel cooling also requires entrance to the spent fuel buildings, which might be difficult in case of low water coverage of the fuel. However, feed and bleed can also be performed using mobile pump units. This system is qualified for the RG 1.60 seismic event. The two mobile units have four sets of diesel driven pumps. There is however no instruction covering the use at three plants at the same time.

All of the methods described above require entrance to the fuel building. It is however possible to use the demineralized water system for feed and bleed of the spent fuel pits without entrance. This system requires electrical power and is not seismically qualified.

No seismic qualification has been performed for the spent fuel cooling system or alternate cooling modes except for the mobile unit for water and power supply, which however requires entrance to the fuel building.

There are three types of spent fuel racks. The racks designed for densely packed fuel are of seismic design. However fuel racks are not very sensitive to the high frequencies dominating the Swedish spectra as the racks have significant lateral gaps and high damping from the surrounding water.

The spent fuel racks needs further investigations before more definite conclusions can be drawn.

The gates in the spent fuel pool have not been evaluated. At severe leakage the water level will sink to a few decimetre above the fuel elements.

2.C.2.3 Earthquake exceeding the design basis earthquake for the plants and consequent flooding exceeding design basis flood

In the flooding report the source of flooding considered is high sea water level. The only phenomenon that relates a high sea water level to an earthquake is the case of tsunami. According to SMHI no tsunami has ever been documented in Swedish waters. Since the plants are not located close to an active seismic region the probability of a tsunami reaching the Ringhals site is extremely low. Outside the plant site there are no constructions that will contribute with a huge amount of water (no dams for example). Some water tanks are found within the plant site. If a tank breaks because of an earthquake the amount of fluid will be limited and the flooding effect only local.

Another source of water within the plant site could be from the fuel pools. But as can be seen in the analyses presented neither the pools in the Reactor Building at Ringhals 1 nor the fuel pools in Fuel Buildings at Ringhals 2 – Ringhals 4 will get damages causing leakage by an earthquake.

In case of an earthquake it is possible that new cracks appear or old cracks are enlarged that may lead to an increased flow of water to the drainage shaft. It is also possible that the cooling water tunnels break and that the water fills up the drainage shafts. The magnitude of an earthquake is identified at the ground level. At the ground level the magnitude is higher than inside the bedrock. It is known that the size of the movement is smaller inside tunnels and mines than at the ground surface. This indicates that the size of damages in cooling water tunnels probably is small even after an earthquake with the probability of 10^{-7}.

As mentioned in the section above it is in case of an earthquake possible that new cracks appear or old cracks are enlarged in the bedrock and that this may lead to an increased flow of water to the drainage shaft. It is also possible that the cooling water tunnels break and that the water fills up the drainage shafts. A case studied in the report of flooding is that the water raises inside the drainage shafts. From the drainage shaft water enters the building. This scenario is described in the flooding analysis. Earthquake in combination with filling up the drainage shafts may be a larger problem than the scenario described in the flooding analysis since equipment that is not earthquake qualified can't be counted in for. During Severe Accidents the two functions most relevant for a flooding scenario are the function Containment integrity protection system and Filtering and Containment Water Injection. These functions are not threatened when the water increases from the bottom of the building.

2.C.2.3.1 Results from the licensee assessments of protection against earthquakes

Ringhals 1

An engineering judgment is that the systems required for mitigation are able to fulfil its functions in case of a 10^{-7} earthquake. However, for the scram function, further evaluations are needed. The consequences of scram function failure at severe accidents are not analyzed.

Based on analysis results the integrity of the fuel storage pools will be preserved in case of a 10^{-7} earthquake.

No seismic qualification has been performed for the spent fuel cooling system or the alternate cooling mode with the fire brigade. This alternate cooling mode requires entrance to the reactor building and is thus possible to use only if the water level in the spent fuel pool is enough to provide radiation shielding.

Ringhals 2-4

An engineering judgment is that the systems required for mitigation are able to fulfil its functions in case of a 10^{-7} earthquake. However, for the reactor trip function, further evaluations are needed. The consequences of reactor trip failure at severe accidents are not analysed.

Concerning the fuel pool buildings analysis results indicates that for the case of a 10^{-7} earthquake there most likely will be local damages but it is unlikely that the fuel buildings collapses. The structure of the fuel pools is robust and the local damages described above will not appear in these structures.

No seismic qualification has been performed for the spent fuel cooling system or alternate cooling modes except for the mobile unit for water and power supply, which however requires entrance to the fuel building. The alternate cooling modes are thus possible to use only if the water level in the spent fuel pool is enough to provide radiation shielding.

2.C.2.4 Measures which can be envisaged to increase robustness of the plants against earthquakes

The most important measure to strengthen the robustness of Ringhals 1-4 is to remedy the shortcomings identified in SMA-validation.

Earlier rough evaluations of the reactor building at Ringhals 1indicate that the roof of this building can be a weakness. In case the roof cannot withstand the 10^{-5} earthquake, roof elements can fall down into the spent fuel pool.

No simple measures to prevent threshold effects are identified.

2.2 Assessments and conclusions regarding earthquake

In Sweden only the latest plants, Forsmark 3 and Oskarshamn3, were originally designed to be resistant to earthquake. Even the mitigation systems specifically designed for severe accidents, introduced in the 1980s in accordance with a government decision, are seismically designed. The design is based on ground response spectrum as stated in the USNRC Regulatory Guide 1.60.

However, since the early of the 1990s the Swedish earthquake level of 10^{-5} per year and site is considered in the design specifications during plant modifications and new installations. This is thus to be regarded as the designed basis earthquake, DBE.

Furthermore, the use of the Swedish earthquake level of 10^{-7} per year at site for the evaluation of structures, systems and components needed to prevent radioactive releases to the environment is according to SSM a reasonable choice. Compared to the DBE the intensity of 10^{-7} earthquake is approximately four times stronger.

2.2.1. Licensees assessment and conclusions

According to the licensees the Swedish plants are able to achieve a safe shutdown condition in case of a DBE, provided that the deficiencies identified in some plants have been remedied.

The mitigation systems are judged to be able to fulfil their function in case of the Swedish 10^{-7} – earthquake.

The integrity of reactor containments, spent fuel pools and other important buildings are estimated to be preserved in case of the 10^{-7} – earthquake. However, there is a need for refined analyses and further investigations before definite conclusions are possible.

Further investigations may also be performed for some plants to evaluate the margins of structures, systems and components against ground motions exceeding DBE. Such investigations should emphasize evaluating margins to reach a safe shutdown condition.

No simple measures to prevent threshold effects have been identified.

Some measures to improve the situation during and after an earthquake have been identified, such as alternate ways to cool the spent fuel pools.

2.2.2. Licensees recommendations for potential improvements

The licensees have identified the following recommendations for further evaluations an reassessments:

a) The most important measure for Forsmark 1 and 2 to strengthen the robustness is to remedy the shortcomings identified in SMA-validation.

b) To find measures against threshold effects there is a need for other more extensive analyses such as new calculations regarding the 10^{-7} earthquake for building structures, seismic PSA and SMA verification of Forsmark 3.

c) A possible improvement at Oskarshamn would be to install new pipes to provide fire water to the spent fuel pool. As a suggestion, these pipes should be capable of being provided with water in the same manner as is used for sprinkling and water filling of the reactor containment with the mitigation systems. Such an installation would minimize manual measures required to establish feed-and-bleed of the spent fuel pool in case all spent fuel pool cooling systems are inoperable.

d) At Oskarshamn the integrity assessments of the reactor containment, scrubber building and spent fuel pools are based on rough up-scaling calculation methods and engineering judgments on a best estimate basis due to the limited time available for this study. Therefore, it may be reasonable to carry out further investigations regarding the structural integrity of the reactor containment, scrubber building and fuel storage pools. The pipe between the reactor containment and the multi venturi scrubber system (MVSS) that allows a controlled pressure relief of the reactor containment should be evaluated further. The function of the pipe is very important to fulfilling the requirements regarding release of radioactive nuclides to the society and to the environment in case of a core meltdown.

e) Further investigations may be performed to evaluate the margins of structures, systems and components in Oskarshamn against ground motions exceeding DBE. Such investigations should emphasize evaluating margins to reach a safe shutdown condition. A reasonable approach would be to evaluate only one sub division in the systems required to bring the reactor to a safe shutdown condition. In general the designs of the sub divisions in required systems are similar.

f) The Screening Evaluation Work Sheet (SEWS) applied in the SQUG assessments does not explicit consider aspects of seismic induced fire. Therefore a very limited SMA could be performed to cover this issue. As a suggestion, this can be done in the same manner at Oskarshamn 1 and 2 as for Oskarshamn 3. The SEWS applied for Oskarshamn 3 requires to check that the equipment is not subject to flammable materials or fire ignition sources.

g) In conformity to Forsmark the most important measure to strengthen the robustness of Ringhals 1-4 is to remedy the shortcomings identified in SMA-validation.

h) Earlier rough evaluations of the reactor building at Ringhals 1indicate that the roof of the building can be a weakness. In case the roof cannot withstand the 10^{-5} earthquake, roof elements may fall down into the spent fuel pool.

i) Another identified deficiency is the control room ceiling at Ringhals 3 and 4. The primary concern is questionable capacity within the support details.

2.2.3. SSMs assessments and conclusions

SSM:s assessment regarding the licensees conclusions is that data is somewhat lacking for demonstrating functions needed to bring the reactors Oskarshamn 2, Forsmark 1, Forsmark 2, Ringhals 2, Ringhals 3 and Ringhals 4 to a safe state after a DBE.

The licensees recommendations for further evaluations and/or specific measures are judged to be appropriate.

There are uncertainties whether all the important functions required for safe shutdown of the plants mentioned above will work as intended during and after a DBE. This because of the identified deficiencies in the performed analyses.

The impact of these deficiencies on safe shutdown must be evaluated in an appropriate manner. Existing knowledge from already implemented structural verifications of various systems and components shall be used to such evaluations.

SSM further notes that the evaluation of buildings and systems for the 10^{-7} earthquake is mainly based on rough analyses and engineering judgments. It is therefore necessary to conduct detailed analyses and investigations to more reliably verify the ability of these structures to withstand a 10^{-7} earthquake.

The utilities thus have to develop detailed programs including information on how and when the

- identified deficiencies will be addressed
- detailed analyses and investigations should be performed
- identified measures that have been judged to improve the situation during and after an earthquake should be carried through, as well as
- measures needed to prevent the so called threshold values

Especially for the reactors at Forsmark and Ringhals there is a need to carry out more detailed analyses for earthquake-induced flooding, where for example leakage from broken water storage tanks and cracks in the cooling water channels are taken into account.

3 Flooding

3.1 Introduction

In 2004 the first issue of Regulations and General Recommendations concerning "Design and Construction of Nuclear Power Reactors" was released. The regulation stipulates in the 14^{th} paragraph that "The nuclear reactor shall be dimensioned to withstand natural phenomena and other events..."The licensee's interpretation of the14^{th} paragraph is that the plant must be able to withstand a an fast external event such as Flooding with a frequency of 10^{-4} to 10^{-6} annually.

The only source of Flooding considered in Swedish units is from high sea water level, because no Swedish unit is located in close proximity to any other water source (such as dams or rivers) than the sea. Waves are not included in the design base flood and are not thought to affect any unit, since no unit is located in direct contact with the sea. Tsunamis, seiches, tides and other phenomena are considered to be covered by high sea water levels.

The design basis sea water level is +2.02 m above normal sea water level for Oskarshamn NPP:s, +2,65 m for Ringhals NPP:s and +3,0 m for Forsmarks NPP:s. The frequency of an increase of the sea water level above the design basis flooding is estimated to 10^{-5} annual at the site for Oskarshamn and Ringhals. The design basis flooding at the Forsmark site estimated to 10^{-6} annual. These frequencies are indications that the design bases for flooding are adequate.

All units are designed to handle the event flooding through high ground water level and all units are equipped with drainage systems to remove ground water from the area between the rock and the buildings. As long as the drainage systems are functioning no flooding due to high ground water level is likely to occur. In case of failure of the drainage function it is assumed that flooding will take many hours before any severe damage will occur to the plants why manual action is likely to be successfully performed. Only Oskarshamn units have drainage systems which supply is backed-up by emergency diesel generators.

3.A. Forsmark

3.A.1 Design basis

3.A.1.1 Flooding against which the plants are designed

3.A.1.1.1 Characteristics of the design basis flood

The events handled are external flooding. Flooding caused by failure of equipment inside any Forsmark unit is not considered.

The design basis flooding are for all Forsmark units the ground level which is +3.0 meters above the normal sea water level.

Apart from design base flood, all the buildings at Forsmark 1 and 2 were originally designed according to the Swedish building code from 1967, and are supplemented with loads from a number of external events, including external flooding up to the 100 year level (+1.8 m).

Apart from the design base flood, the buildings at Forsmark 3 were originally designed according to the Swedish building code from 1975.

3.A.1.1.2 Methodology used to evaluate the design basis flood.

The variation of the water level in the Baltic Sea depends on the water flow through the Öresund and Danish straits as well as the water flow from rivers to the Baltic Sea. The volume of water flows through the straits depends of air pressure variations and the associated wind conditions. Strong winds from the west to northwest cause the strongest inflows of water while winds from the east to northeast cause corresponding outflows of water. The sea water level is also locally affected by the wind.

In the Baltic, the influence of tides is very small, at most about 0.1 m.

The water level in Forsmark has been registered for 30 years. The highest water level was measured in January 2007 +1.44 m. The design basis corresponds to an increase of the sea water level to +3 m. Such an increase of the sea water level is estimated to 10^{-6} annual at the site.

The phenomenon tsunami isn't relevant because of the site location at the west side of the Baltic Sea (east coast of Sweden). The area is seismically stable and the Baltic Sea is too shallow for a significant tsunami to be created. A seiche (a short term local sea water level change) is a local effect with a short duration that is caused by a local change in pressure that moves at the same speed as a long wave. The speed of the long wave is determined by the depth in the area. The change in pressure creates the wave, and if the speed of the wave and the pressure change resonate there is a major change in sea water level. A seiche of a full 1.23 m (difference between high and low water) and lasting 12 minutes has been observed near Forsmark.

Tsunamis and seiches are considered to be covered by high sea water levels.

The Swedish Meteorological and Hydrological Institute have estimated the return period for extreme precipitation. Twelve measuring stations have been weighted together over a time period of 121 years. The extreme value for precipitation is assumed to be 155 mm over a 24 hour period. Precipitations for shorter time periods are not specified due do a lack of measurement data.

3.A.1.1.3 Conclusion on the adequacy of protection against external flooding

The estimated frequency for a sea water level of +3 m at Forsmark is estimated to 10^{-6} annual. This is an indication that the design basis flooding is adequate.

All known phenomena (increase in sea water level, rain, etc.), as described above, are considered to be covered by design basis flooding. The design basis flooding is thereby considered to be adequate.

3.A.1.2 Provisions to protect the plants against the design basis flood

The design base flood for Forsmark 1, 2 and 3 is fulfilled by having critical parts for bringing the plant to safe shutdown and cooling of the spent fuel pools located above ground level or in watertight rooms (H-areas) containing important safety equipment for the respective subdivision on top basement floor. However, there is one weakness, for Forsmark 1 and 2, in secondary cooling for emergency diesel generators. If the pumps in the secondary cooling system are started when the sea water level has passed +2.5 m, the emergency diesel generators cooling can be jeopardized.

High sea water level

A sea water level above approximately +2.0 m (+2,2 m for Forsmark 3) entails water in the waste removal duct for the travelling band screens, resulting in reduced capacity in terms of removing the filtered waste material. This means that the main cooling water flow must be reduced in the event of a sea water level of +1.6 m (+1,8 m for Forsmark 3), and completely shut off with a sea water level of +2.0 m (+2,2 m for Forsmark 3).

When the sea water level approaches +3.0 m, the capacity of the waste removal duct is so low that it may be necessary to remove the filtered waste manually. If the cooling water flow is limited to what is required solely for residual heat removal, there will be a small amount of waste, even in the event of extremely high amounts of sedimentation and debris in the water, such as during storms. If only auxiliary cooling water is available, the amount of waste will be so small that by manually removing the waste, the residual heat removal can remain intact up to a sea water level of +3.0 m (+4.1 m for Forsmark 3).

If the flooding is due to high sea water level, an alarm is issued in the control room at Forsmark 1 and 3 when the sea water level exceeds +1.6 m. The control room at Forsmark 1 and 3 then staffs the cooling water intake building of Forsmark 1 and 2 respectively Forsmark 3. The control room of Forsmark 1 also notifies the control room at Forsmark 2 about the alarm. In the event of rising sea water level, all Forsmark units begin to reduce their main cooling water flow and initiate controlled shutdown. This is undertaken according to a separate emergency operating procedure, but for Forsmark 3 the procedures do not indicate at exactly which level this is necessary. If the reason for the flooding is high winds, waves will form in the channel. The tops of the waves will be higher than the sea water level recorded by the sea water monitoring system. However, since the plant is not located directly at the open sea, the sea water level needs to be much above +1.8 m before waves start hitting the any Forsmark unit's reactor building.

Since all of Forsmark units will reduce their power when the water reaches the same sea water level, there is a risk of the external power grid becoming unstable. This must therefore be coordinated. Should the grid fail because of shutdowns due to high sea water level, the stations can still perform and maintain safe shutdown using backup diesel units.

High ground water level

The normal average ground water level is approximately the same as the normal average sea water level of the Baltic Sea.

Forsmark 1, 2 and 3 are equipped with a storm drain systems and rock drainage. An extreme precipitation in a limited period of time and a simultaneously long-lasting loss of off-site power, may lead to flooding of sumps and bedrock gaps to a maximum level that is somewhat higher than the current level in the cooling water channel. Once the causes of the flooding have been remedied or the flooding has stopped, submersible pumps can be used. The rock drainage is routinely tested to ensure required function.

For Forsmark 1 and 2 ground water levels above +1.8 m will result in that the basement levels at Forsmark 1 and 2 (except for the watertight H-areas) will be flooded. Emergency diesel generators, pumps and other equipment that are part of any safety function are located above ground level (+3.0 m), or in the watertight H-areas below ground level. Cables below the ground level are designed to be watertight and therefore no short-circuits are assumed.

Forsmark 3 is designed to withstand a water pressure corresponding to an external flood up to ground level, + 3.0 m, both for groundwater and for sea water, however not designed to be watertight in the long term. If flooding occurs due to high groundwater levels, submersible pumps can be lowered into the pump cavities to reduce the water level in the bedrock gap.

3.A.1.3 Plants compliance with its current licensing basis

3.A.1.3.1 General process to ensure compliance

There is no unique process verifying resistance against flooding. The capacity to withstand flooding is achieved by having the original design requirements, the design requirements for replacement parts and for new installations and the safety systems having high availability. Processes for achieving a high level of availability in terms of the safety systems and for handling original design requirements and design requirements for system changes are listed below.

- Technical specifications provide a set framework for the operation of the plant ensuring the safety of the surrounding area.
- Component or system functions affected by maintenance should be verified for operational readiness before restart.
- A maintenance program exists for all SSCs, including inspection of water tightness of H-areas.

3.A.1.3.2 Known deviations

There are no known deviations against current licensing basis for any Forsmark unit.

3.A.1.3.3 Specific check following fukushima npp accident

A specific check has been done in accordance with SOER 2011-12 issued by WANO. Forsmark has responded to the recommendations and has identified actions. However, no actions regarding external flooding have yet been initiated.

3.A.2 Evaluation of safety margins

3.A.2.1 Estimation of safety margin against flooding

The frequency of the design basis sea water level is 10^{-6} annual at site and corresponds to a sea water level of +3 m. Below is an assessment, for all Forsmark units, regarding which levels the plants can withstand when it comes to external flooding and groundwater follows.

Flooding of the interior of the buildings up to ground level can result in all rooms up to ground level gradually being filled with water (by time also the watertight H-areas in Forsmark 1 and 2). Equipment in rooms filled with water is not designed to function under water. Though critical parts for bringing the plant to safe shutdown are located above ground level or in watertight rooms (H-areas) containing important safety equipment for the respective subdivision on top basement floor. The emergency diesel generators are all placed on foundations somewhat above ground level and an increase in the water level a few decimeters above ground level will not impair functioning.

The conclusion is that, for all Forsmark units, if an external flooding above ground level (a few decimeters over +3.0 m) due to high sea water levels occurs, the water level in the respective buildings rises to the same level. Equipment necessary for safety aren't designed to withstand such a water level and therefore damage to core is likely if no manual action is taken.

Flooding leading to core damage is not considered to simultaneously affect equipment that is required for severe accidents. All units are provided with equipment for filtered pressure release, Multiple Venturi Scrubber System (MVSS). Filtered depressurization of the reactor containment is of interest during external flooding above +3.0 m, when fuel damage is assumed. The system is passive at all Forsmark units and is considered to function. However, a water-filled lower dry-well is essential in order to an efficient use of MVSS. In addition, long-lasting extremely high sea water levels may complicate long-term measures to keep the mitigating systems functioning. Filling the containments with water should be possible with the system for water filling of reactor containment, even if the fire water system should fail, because fire truck can be used. One of two possible connections for water from the fire truck is located above the ground level. However, potential problems getting fire trucks into the flooded area and prioritization problems when several reactors are affected can potentially pose a problem.

In addition to the fact that the fuel in the cores is at risk of being damaged during extreme external water levels above design basis flooding, there is also a risk of uncovering the fuel in the spent fuel pools at these external water levels. However, this process is slower, which provides more time for countermeasures, for example adding external water with hoses in the spent fuel pools

In the event of a long-term loss of off-site power, the diesel storage tanks will need to be filled. A sea water level higher than +2.2 m will cause traffic disruptions on the roadways and the station area. It is likely that the disruptions will increase as the sea water level increases. Transportation roads may also become blocked.

Failure of safety functions

The table below shows a summary at which sea water levels each safety functions are assumed to withstand before failing. Since the situation is similar for Forsmark units the summary applies to all three facilities. The table primarily deals with a theoretically rapid course of events with no forewarning.

Function	**Level Forsmark 1, 2 and 3**	**Comments**
Reactivity control	> +3.0 m	The hydraulic scram systems for the respective reactors are fail-safe designed, which means that the reactivity control should function regardless of the water level.
Pressure relief	+3.0 m	Initial depressurization will be possible also for water levels above +3.0 m, which means that the pressure is limited to just below 8.0 MPa regardless of the water level. The function is, long term, limited by the capacity of batteries which won't be replenished when power supply from both off-site power and emergency diesel generators is lost. Therefor +3.0 m, long term, is setting the limitation for the function.
Emergency core cooling	+3.0 m	This function fails when water leaks into the buildings from ground level and causes flooding. There is a potential margin of protection as long as the water level is just over +3.0 m, and this level does not last too long.
Residual heat removal	+3.0 m	This function fails when water leaks into the buildings from ground level and causes flooding. There is a potential margin of protection as long as the water level

		is just over +3.0 m, and this level does not last too long. Residual heat removal can also be done using the MVSS which will function regardless of the water level.
Isolation and protection of reactor containment	+3.0 m	This function is questionable at water levels above +3.0 m, since some isolation valves are motor operated. There is a potential margin of protection as long as the water level is just over +3.0 m, and this level does not last too long.
Emergency ventilation	+3.0 m	Is assumed not to function when the flooding has caused loss of power supply, which means just above +3.0 m.

3.A.2.2 Measures which can be envisaged to increase robustness of the plants against flooding.

Update the technical manual for plant operators to include the scenario Flooding, since exceeding design base flood risk to eliminate the power supply to all of the safety systems when design basis flooding is exceeded. If the assessment is that the sea water level will exceed the design basis flood level, the control rods should be inserted, the pressure in the reactor should be reduced, the lower dry-well filled with water and the evacuation of personnel should begin.

Extend the contacts with Swedish Meteorological and Hydrological Institute regarding warnings of extreme weather.

A review should be done to check for gates and other forms of exterior doors to see how well they withstand external water levels above +3.0 m, and make reinforcements if needed. In such an evaluation the command center should also be taken into consideration.

Evaluated how the water is distributed inside the plants during an external flooding.

Procedures for a coordinated power reduction in the event of levels close to +1.6 m should be established in order to avoid shutting down all Forsmark units in such a tight interval that off-site is lost.

A review should be done to check the options and the procedures for using submersible, mobile pumps for rock drainage.

A review should be done to check the options for countering leaks during extreme rain.

The levels in the in the surge shafts, if the plants have to shut down at higher sea water levels than suggested in the procedures, should be studied.

For Forsmark 1 and 2, Update the procedures so that the cooling water pumps will be started before the water level +2.5 m is reached in order to eliminated the risk of not beeing able to start them. A review should be done to check if there is a need for a separate early warning system for Forsmark 2

For Forsmark 3, Update procedures to indicate the exact sea water level at which it is appropriate to shut down the plant. A review should be done to check if all buildings are completely leak-tight up to ground level or whether an external water level at or just below +3.0 m could lead to water leaking into some of the buildings.

3.B. Oskarshamn

3.B.1 Design basis

3.B.1.1 Flooding against which the plants are designed

3.B.1.1.1 Characteristics of the design basis flood (dbf)

The events handled are external flooding. Flooding caused by failure of equipment inside the Oskarshamn units or the effects of excessive precipitation is not considered. The design basis flooding consists of three different scenarios:

Sloshing

Sloshing is a phenomenon that occurs if the main cooling water pumps suddenly are tripped e. g., due to loss of off-site power. The sea water flow in cooling water channel/tunnel cannot interrupt momentarily when the pumps are tripped. This will cause an increase of the water level in the sea water inlet chamber. The level increase due to sloshing is combined with a high sea water level.

High sea water level

The design basis corresponds to an increase of the sea water level with +2.02 m above normal average sea water level.

High groundwater level

The normal average ground water level is approximately the same as the normal average sea water level of the Baltic Sea.

The ground surface level of Oskarshamn 1 and 2 is +6 m (above sea water level) and the ground surface level of Oskarshamn 3 is +3 m (above sea water level). The reactor building at each Oskarshamn unit is erected in a shaft which is blasted into the rock. The design basis water level in the erected shaft of Oskarshamn 3 is the ground surface level. The design basis water level in the erected shaft of Oskarshamn 2 is -10 m below normal sea. The design basis water level in the erected shaft of Oskarshamn 1 is +1.35 m.

The rock at the Oskarshamn site has very low groundwater bearing and the movement of the groundwater is limited to crack zones. Normal leakage rate to the area between reactor building of each Oskarshamn unit and the rock is in the range of 3-5 m^3/h.

3.B.1.1.2 Methodology used to evaluate the design basis flood.

High sea water levels

The variation of the water level in the Baltic Sea depends on the water flow through the Öresund and Danish straits as well as the water flow from rivers to the Baltic Sea. The volume of water flows through the straits depends of air pressure variations and the associated wind conditions. Strong winds from the west to northwest cause the strongest inflows of water while winds from the east to northeast cause corresponding outflows of water. The sea water level is also locally affected by the wind.

The influence of tides is very small in the Baltic Sea, at most about 0.1 m.

SMHI has registered the water level in Oskarshamn since 1975, and it quite well represents the water level outside the Oskarshamn site. The highest measured water level is +1,00 m. The design basis corresponds to an increase of the sea water level to +2.02 m. The frequency of such an increase of the sea water level is estimated to 10^{-5} annual at the site, This is an indication that the design basis flooding is adequeate.. The extreme sea water level (+ 2.02 m) has been calculated by the SMHI and is based on registration of the sea water level in the vicinity of the site during 45 years. According to SMHI an extrapolation up to twice the registration time may be done with reasonable accuracy. Hence the design basis sea water level frequency is associated with uncertainty.

Tsunami and high waves

The phenomenon tsunami isn't relevant because of the site location at the west side of the Baltic Sea (east coast of Sweden). The area is seismically stable and the Baltic Sea is too shallow for a tsunami to be created. Therefore no tsunami is assumed when evaluating margins. About 100 m outside the sea water intake of Oskarshamn 1 and 2 two small islands together with the harbor pier constitute a protective reef against sea waves. The buildings of Oskarshamn 1 and 2 are situated with distance from the shoreline and are erected on rock with a ground level of several meters above the normal average sea water level in the Baltic Sea. Extreme sea waves are therefore not evaluated since Oskarshamn 1 and 2 are assumed to be insensitive to wave surge.

Oskarshamn 3 is situated with a distance from the shoreline and is erected on rock with a ground level of +3 m. In the east direction, that in practice is facing open sea, wave surge may affect the shore line, the shore line is located approximately 200-500 meters from Oskarshamn 3. In the north direction an island ensures that high wave surge is of no concern and in the south direction the harbor pier and the Simpevarp peninsula protect the shoreline from wave surge.

3.B.1.1.3 Conclusion on the adequacy of protection against external flooding

The estimated frequency for a sea water level of +2,6 m at Oskarshamn site is estimated to 10^{-7}. This is an indication that the design basis flooding is adequate

All known phenomena (increase in sea water level, rain, etc.), as described above, are considered to be covered by design basis flooding. The design basis flooding is thereby considered to be adequate.

3.B.1.2 Provisions to protect the plants against the design basis flood

In general the Oskarshamn units are robust against external flooding because they are erected on rock with a ground level of +6 m (Oskarshamn 1 and 2) or +3 m (Oskarshamn 3). In addition, the buildings of all Oskarshamn units are situated with distance from the shoreline thus making them insensitive to wave surge.

Sloshing

Following the sea water coolant inlet tunnel (prior to the trash racks and travelling basket screens) there are open wider channels which constitutes inlet basins and therefore reduces the effect of sloshing. In Oskarshamn 1, provisions have been taken to protect components installed in pump buildings against sloshing. The structures of the sea water inlet channels and chambers in Oskarshamn 2 and 3 are designed to withstand a water level of +3 m.

High sea water level

The design basis is a sea water level of +2.02 m

For Oskarshamn 1, the structure of the sea water intake is designed to withstand a water level of +2.2 m. Therefore a margin of 0.18 m exits. No safety classified equipment is endangered in case the sea water level reaches the design basis level.

For Oskarshamn 2, the structures of the sea water inlet channels and chambers are designed to withstand a water level of +3.0 m. Therefore a margin of nearly 1 m exists. No safety classified equipment is endangered in case the sea water level reaches the design basis level.

For Oskarshamn 3, The lowest part of the ground level where the surrounding site fence gate faces the sea is +3.4 m. The height of the road bank is +4.3 m at this location. The lowest ground level facing the Baltic Sea is +4.3 m. This in practice constitutes a barrier. No safety classified equipment is endangered in case the sea water level reaches the design basis level.

High groundwater level

In Oskarshamn 1, the reactor building has been reinforced to withstand a flooding level up to +1.5 m in the area between the outer walls of the reactor building and the surrounding rock.

Around the reactor building drainage pipes are installed leading ground water to the drainage sump tank. From the drainage sump tank the water is discharged to the sea water inlet chamber by drainage pumps. The capacity of the three drainage pumps are 15 kg/s each i.e. a total capacity of 162 m^3/h. The drainage pumps have emergency diesel generators backed-up power supply. In case of failure of the drainage pumps and failure of countermeasures to repair/replace those drainage pumps the level in the shaft reaches -6 m after 90-100 h (assuming a leakage rate of 5 m^3/h). At level -6 m leakage water will also flood to the area below and around the intermediate building, which will significantly increase the total volume to fill. At level -2 m in the shaft also the area below the pump building will be flooded. The level -2 m is reached several days after the drainage pumps have failed. Time to provide provisional measures to drain the shaft e. g. mobile pumps exists

The hypothetic hazard events, spontaneous increase of ground water leakage through the rock or broken barrier between the cooling water channel and the shaft is assumed to cause a water level of +1.35 m in the shaft. Because the reactor building is reinforced to withstand this water level also components installed in the reactor building will be unaffected. The emergency diesel generators are located in buildings which are erected on ground rock with lowest levels of +2.8 m and +6.0 respectively and therefore are not endangered by high ground water level. Hence the emergency power supply is maintained. All required safety functions can be fulfilled in case of flooding of the shaft up to a water level of water level of +1,5 m where the reactor building is erected.

In Oskarshamn 2, the ground water leakage to the shaft around the reactor building is accumulated in the drainage sump tanks. From the drainage sump tanks the water is discharged to the slosh chamber (on the sea water outlet side) by the drainage pumps. The capacity of the drainage pumps is 11.7 kg/s each. The drainage pumps have gas turbine backed-up power supply. It is verified that the reactor building can withstand a water level of 6 meters in the shaft (-7.0 m). In case of failure of the drainage pumps and failure of countermeasures to repair those drainage pumps it will take several days to reach the level -7.0 m in shaft (assuming normal ground water leakage). Sufficient time to provide provisional measures to drain the shaft e. g. mobile pumps exists

In Oskarshamn 3, all safety classified buildings are designed to withstand a ground water level up to the ground level (+3.0 m).

3.B.1.3 Plants compliance with its current licensing basis

3.B.1.3.1 General process to ensure compliance

The water level in the drainage sump tanks, which accumulate water from the drainage pipes in the shaft where the reactor building is erected, is continuously measured. Alarm is given in the main control room in case of high water level in the drainage sump tanks. The level measurements also initiate start/stop of the drainage pumps.

In Oskarshamn 2 and 3 main control rooms the sea water level is continuously measured and monitored. Oskarshamn 1 control room is located adjacent to Oskarshamn 2 control room and will have information from there.

3.B.1.3.2 Known deviations

There are no known deviations against current licensing basis for any Oskarshamn unit.

Currently no alarm is registered in the main control room in case the sea water level would reach a level that in case of sloshing could cause an unacceptable high water level in the sea water inlet chambers. Such an alarm is therefore recommended to be installed. Further operational procedure should prohibit power operation in case the sea water level reaches such a level. The Operational Limits and Conditions (STF) do not prescribe this restriction. Therefore the STF is recommended to be complemented with such a restriction. The sea water level that should be used as a limit should be defined and it is also recommended that some margin is included.

3.B.1.3.3 Specific check following fukushima npp accident

A specific check has been done in accordance with SOER 2011-12 issued by WANO. Measures carried out are summarized in OKGs response to WANO.

3.B.2 Evaluation of safety margins

The frequency of the design basis sea water level is 10^{-5} annual at site and corresponds to a sea water level of + 2.02 m. By extrapolating the extreme sea water levels, calculated by the SMHI, a sea water level of + 2.6 m approximately corresponds to a frequency of 10^{-7} annual at site. A sea water level of +2.6 m could be assumed as an upper limit when evaluating the margins even though it corresponds to an extremely unlikely event. The uncertainty may therefore be high for a sea water level that corresponds to a frequency of 10^{-7} annual at site, since the uncertainty will increase when calculating sea water levels with very long recurrence time. Therefore, also cliff edge effects for sea water level above +2.6 m will be evaluated.

An increase of the sea water level does not occur momentarily. It may be assumed that this takes some hours. Manual initiated scram and prepared manual measures may therefore be credited.

A planned shutdown of each Oskarshamn units, in case the sea water level would reach a level that in case of sloshing could cause an unacceptable high water level in the sea water inlet chambers, prevents sloshing from causing flooding. It is assumed that the recommended measures will be implemented and therefore flooding due to sloshing is not evaluated.

Hypothetical hazard events resulting in spontaneous increase of ground water leakage through the rock or a broken barrier between the sea water cooling channel and the shaft are assumed. A spontaneous increase of ground water leakage or broken barrier, combined with a ground water or sea water level that exceeds +1.18 m, is assumed to be extremely improbably and is not evaluated for Oskarshamn 2. The frequency of a sea water level of +1.18 m is 10^{-2} annual at site. For Oskarshamn 1 the critical level is +1.35m and for Oskarshamn 3 there is no critical level.

3.B.2.1 Estimation of safety margin against flooding

As mentioned before, a sea water level of +2.6 m is estimated to a frequency of 10^{-7} annual at site.

Oskarshamn 1 and 2

One identified cliff edge effect is a sea water level higher than +3.0 m. Such a water level is higher than the structure of the intake channels and would flood the pump house. It cannot be excluded that sea water could enter the shaft where the reactor building is erected causing local damage on the outer walls of the reactor building. This may cause leakages into the reactor building.

The core cooling is judged to be ensured up to a sea water level slightly above ground level +6.0 m (considering pump foundations etc.).

Residual heat removal can be accomplished with the mitigation system, which does not require any electrical power supply and is not affected by flooding up to ground level (+6.0 m).

The spent fuel pool cooling can be accomplished by feed-and-bleed using fire trucks or the direct driven diesel fire water pump in case all ordinary cooling systems are unavailable

Oskarshamn 3

A cliff edge effect is a sea water level higher than +3.0 m. The storm water system draws off rain water from roofs, walls and yard to the Baltic Sea using the geodetic height difference. In case the sea water level rises above +3 m, sea water could flow from the Baltic Sea to the yard and thereby cause flooding. There are no shut-off valves in the storm water system that may be closed to prevent such back-flow. In case the yard is flooded it is assumed that water will enter safety classified buildings.

The ability to accomplish the core cooling is judged to be possible up to an internal water level slightly above ground level (considering pump foundations etc.)

Residual heat removal can be accomplished with the mitigation system, which does not require any electrical power supply and is not affected by flooding up to ground level.

The spent fuel pool cooling can be accomplished by feed-and-bleed using fire trucks or the direct driven diesel fire water pump in case all ordinary cooling systems are unavailable.

3.B.2.2 Measures which can be envisaged to increase robustness of the plants against flooding.

A planned shutdown in case the sea water level would reach a level that in case of sloshing could cause an unacceptable high water level in the sea water inlet chambers, will be implemented.

Oskarshamn 1 and 2

It could be possible (theoretically) to prevent sea water entering the inlet and outlet tunnels by e.g. high sluice gates and walls around the tunnel openings. Thereby the integrity of buildings would theoretically not be endangered up to an increased sea water level of +6 m.

Oskarshamn 3

In case the sea water level rises above +3 m, sea water could flow from the Baltic Sea to the yard via the storm water system. It would be possible (theoretically) to avoid such back-flow by installing a shut-off valve in the storm water system.

3.C. Ringhals

3.C.1 Design basis

3.C.1.1 Flooding against which the plants are designed

3.C.1.1.1 Characteristics of the design basis flood

The events handled are external flooding. Flooding caused by failure of equipment inside any Ringhals unit or the effects of excessive precipitation is not considered.

The design basis flood is, for all Ringhals units, +2.65 m above normal sea water level. The design base flood was introduced in order to comply with the requirements of § 14 of SSMFS 2008:17 "The Swedish Radiation Safety Authority's Regulations concerning the Design and Construction of Nuclear Power Reactors.

All Ringhals units are also designed to withstand flooding through high ground water level. The event high ground water level is linked to the design basis flood since a high sea water level can indirectly lead to an increase of the ground water level. High ground water level is according classified as a slow event. For a slow event operating provisions are credited in order to handle the event.

3.C.1.1.2 Methodology used to evaluate the design basis flood

The design bases flood is based on statistical data. Swedish Meteorological and Hydrological Institute data measured at the Ringhals site during the period 1887 to 2006 was used to calculate the design basis flood of +2.65 m.

The source of the flooding considered is high sea water level. No Ringhals unit is located in close proximity to any other water source (such as dams or rivers) than the sea. Waves are not included in the design base flood and are not thought to affect any Ringhals unit since no Ringhals unit is located in direct contact with the sea.

High sea water level can be caused by different phenomena (e.g. a storm or gust bumps) that, with the exception of tsunamis, all are assumed to be covered by the event statistics and different events leading to flooding are hence not analyzed separately.

Water level variations due to tide is approximately 0.3 m and hence quite small.

Sloshing is a phenomenon that may occur if the main cooling water pumps suddenly stop which will cause a raised water level in the inlet of the cooling water tunnel. Sloshing has a potential to damage the cooling water tunnel. Sloshing shafts have however been built in the inlets of the cooling water tunnels in order to protect the constructions against this phenomenon. Hence, sloshing is not analyzed further.

A tsunami can be caused by earthquakes, landslides, volcanic eruptions or meteorites. No Ringhals unit is located close to an active seismic region or active volcanoes and the surrounding sea does not have the hydrological conditions needed in order for a tsunami creating landslide to occur. The probability of a meteorite impact is very low and hence screened out. The probability of a tsunami reaching the Ringhals site is thereby extremely low and flooding due to tsunami is therefore screened out. The peninsula of Denmark and southern Norway also effectively shields all Ringhals units from a direct tsunami. According to SMHI no tsunami has ever been documented in Swedish waters.

3.C.1.1.3 Conclusion on the adequacy of the design basis flood

The estimated frequency for a sea water level of +2,65 m at Ringhals site is estimated to 10^{-5}. This is an indication that the design basis flooding is adequate

All known phenomena (increase in sea water level, rain, etc.), as described above, are considered to be covered by design basis flooding. The design basis flooding is thereby considered to be adequate.

3.C.1.2 Provisions to protect the plants against the design basis flood

The Ringhals sites ground elevation is +3 m and hence above the design basis flood level of +2.65 m. The only part of the units that would be affected by a design basis flood is the cooling water channels in connection with the turbine buildings.

A design basis flood would make it more difficult for personnel and equipment to access the Ringhals site since some of the ground level surrounding the Ringhals site is lower than all the Ringhals units' ground level, leading to flooding of the surrounding roads. The design basis flood will have no impact on any Ringhals unit hence requiring no extra personnel or equipment. It is also estimated that the design basis flood will only last for a couple of hours up to half a day.

High sea water level

The cooling water channels can withstand water pressure from a water level of +3.4 m. The value +3.4 m is the design basis flood compensated for the fact that the water level at the outlet is estimated to be 0.7 – 0,8 m higher than at the inlet during full power operation (the water level increase is caused by the main cooling pumps). A design basis flood will hence not impact any Ringhals unit in such a way that key structures, systems and components (SSC) for achieving safe shutdown or cooling of spent fuel pools needs to be identified, including provisions to maintain the water intake functions and emergency electrical power supply. However, in order to avoid water from the outlet through the open surge chamber to flood the courtyard, the power output needs to be reduced in order to reduce the water level at the outlet to below +3 m.

The operator actions taken depend on the consequences of the flooding. There are no specific operating procedures connected to flooding. However, there are some operating procedures connected to events likely to occur during a design basis flood such as blocked intake building. The design basis flood is also characterized as a fast event and all Ringhals units should hence be able to withstand it without operating provisions.

High ground water level

The event flooding through high ground water level is a slow event and therefore it's likely that manual action will and can be executed.

The normal average ground water level is approximately the same as the normal average sea water level of the Sea.

All Ringhals units are designed to handle the event flooding through high ground water level. The parts of buildings that are below the ground level are designed to resist the design value of flooding through drainage. The drainage is postulated in the calculation, which means that the construction is not designed to be leak tight. Pipes for drainage are installed around the buildings. The drainage pipes ends at different shafts equipped with pumps. From the shafts the water is pumped to the system for surface water. Shafts and pipes for the drainage system are not in contact directly with either the tunnels for the cooling water or the sea. In some shaft there is a continuous leakage of sea water from the discharge tunnel but it is not thought to increase considerably during a design basis flood. A leakage in the bedrock is not postulated to appear in connection to extreme flood. It can hence be concluded that the design basis flood cannot indirectly cause flooding of any Ringhals unit by increasing of the ground water level and hence no key SSC needed for achieving safe shutdown need to be identified for the event flooding through high ground water level.

The design provisions to protect the Ringhals units against flooding are aimed at protecting the Ringhals units from drainage water. The pumps in the drainage shafts start automatically when the water reaches a certain level. Beside the bottom slab of each Reactor Buildings there are shafts with pumps that lower the ground water level between -19 m to -27 m below average ground water level. Below the Turbine Buildings there are also shafts with pumps that lower the ground water level to between -8 m to -10 m.

The constructions of the lowest parts of each unit Turbine Building are not drained since these constructions are next to the tunnels for the cooling water. These constructions are designed to resist the design values of the sea water level. The structures in this part of the buildings are grouted directly against the rock with watertight concrete.

In the main drainage shafts there are two pumps and level indicators that will activate the pumps at different water levels. When the water reach the lowest level indicator one of the pumps starts and if the water level reaches the second level indicator the second pump starts and an alert is sent to the control room. The control room staff will then check that the level of the water is decreasing. If the water level continues to rise even with both pumps operating a decision of what measures are needed is made. One option is utilizing mobile submersible pumps that are available on site.

Loss of off-site power will however influence the event flooding through drainage since the pumps in the drainage shafts run on off-site power. The water level will hence rise in the shafts and around the lower parts of the buildings. The water flow to the shafts is however moderate and the volume that has to be filled before the water causes any major damage are quite large. It will most likely take many hours up to several days before any Ringhals unit will be severely affected by the flooding.

3.C.1.3 Plants compliance with its current licensing basis

3.C.1.3.1 General process to ensure compliance

Inspections of relief gates and doors are done annually and tests are done every second year. The inspections of the drainage pumps are included in the maintenance program and the pumps are tested once a year. That the drainage pumps are being operated continuously is also and indirect verification of the drainage pumps function.

3.C.1.3.2 Known deviations

The status of the drainage pipes is not checked regularlyand it has happened that the system has been clogged and the drainage water has hence not reached the pump pits resulting in internal flooding. Since a functioning drainage system is an essential part of the units' protection against flooding an overview of the status of the drainage system is therefore recommended.

The water level at the outlet is 0.7 – 0.8 m higher than at the inlet when the units are operating at full power. When the water level is extremely high there is thereby a risk that the courtyard will be flooded. The water would also enter the Ringhals 1 and 2 through a room between the two units where the outlet of the contaminated water system is located. Therefore, when the water level is extreme high, a need to reduce power and stop as many main cooling water pumps as necessary in order to reduce the water level difference exist. No instructions governing reduction of power and stopping main cooling water pumps at high water levels have however been found. The matter is also complicated by the fact that the water level is not measured in the inlet but in a bay in close proximity to the Ringhals site. The water level in the bay and in the intake channels can however differ with up to approximately 0.5 m which introduces a significant error when determining the water level in the intake channels.

3.C.1.3.3 Specific check following fukushima npp accident

A specific check has been done in accordance with SOER 2011-12 issued by WANO. Ringhals has responded to the recommendations. No gap concerning the capability to mitigate internal and external flooding events required by station design was identified. Ringhals is also participating in the work done by NOG (Nordic Owners Group) and PWROG (PWR Owners Group) in light of the Fukushima NPP accident.

3.C.2 Evaluation of safety margins

3.C.2.1 Estimation of safety margin against flooding

When considering flooding events with a probability smaller than the design base flood, flooding due to high sea water level is still deemed to be the most likely event that could lead to severe damage to the fuel and will therefore be analyzed. In the different flooding scenarios depending of the sea water level and an estimation of the risk of severe damage to the fuel for each scenario is presented. The different scenarios, valid for any Ringhals units at full power operation, are discussed below:

Scenario #	Sea water level	Comment	Fuel damage
1	below +2.65m	Design base flood the Ringhals units are designed to handle.	No
2	+2.65 m - +3.0 m	Flooding above design base flood but not above ground level.	No
3	+3.0 m - +3.3 m	Flooding above ground level, manageable amounts of water entering the buildings.	Unlikely
4	+3.3 m - +4.0 m	Flooding significantly above ground level, large amount of water entering the buildings.	Possible
5	Above +4.0 m	The main doors break and all buildings are internally flooded up to this level.	Yes

<u>**Sea water level below +2.65m (scenario 1)**</u>

The design base flood that has been discussed in the preceding section

<u>**Sea water level between +2.65 m and +3.0 m (scenario 2)**</u>

When the water remains below the ground level the cooling water tunnels in connection with the turbine buildings are (as for the design base flood) the only part of any Ringhals unit to affected by the flood. As previously stated, the cooling water channels can withstand the water pressure from a water level of +3.4 m. The power is assumed to be reduced at all Ringhals units and hence the water level is below +3.4 m at the outlet as well as the inlet. The cooling water channels will hence not be damaged and no water will enter any Ringhals unit.

Sea water level between +3.0 m and +3.3 m (scenario 3)

Water is exceeding the ground level but the amount of water entering the buildings is manageable.

The water itself doesn't cause any damage to buildings and doors at any Ringhals unit. Floating objects aren't assumed to threaten the integrity of any building or door. The only possibility of water to enter any building at any Ringhals units is from the ground level through various openings. Visual inspections of all Ringhals units show only a handful of various openings leading into the Ringhals unit's buildings. Hence, the leakage will be small and manageable and the volume to fill before severe damage occurs is large.

Water must rise to at least +3.3 m before significant amounts of water will enter any Ringhals unit. Water leaking into the buildings will be taken care of by the floor drain system in each Ringhals unit. Via pipes the water will flow to floor drain tanks that are emptied by pumps usually to the cooling water discharge tunnel. The capacity of the floor drain system in each Ringhals unit is however limited.

Sea water level between +3.3 m and +4.0 m (scenario 4)

Large amounts of water will enter the Ringhals units through various openings. There is also a risk that cracks will start to appear in the tunnels that will contribute to the inflow of sea water since the water level now exceeds the water level the cooling water tunnels have been shown to handle. The Ringhals units will hence be significantly affected by the flood but it is impossible to say with certainty whether or not this level of water will lead to severe damage to the fuel since that depends on a number of factors such as what rooms are flooded and the duration of the flooding.

Sea water level above +4.0 m (scenario 5)

Sea water reaches a level where the doors break. At all Ringhals units doors will break at sea water level of +4.0 m and water will instantly flood all Ringhals units up to the sea water level and fuel damage will occur.

Other modes than power operation

If flooding occur, for Ringhals 2, 3 or 4, when in mode 5* (static shutdown) the reactor coolant system is open to the containment atmosphere and decay heat cannot be removed by the steam generators. The reactor vessel head is mounted but the bolts are not necessarily mounted. If the unit would be flooded during this mode borated water can be transferred by gravity from the RWST to the reactor vessel. No recirculation is however available. It should however be noted that the units actively works at minimizing the time spent in mode 5* during shutdown and the planned time for mode 5* during shutdown is hence only 8 hours. During start up the Ringhals units remain longer in this mode but the need for residual heat removal is much smaller.

Failure of safety functions

The table below shows a summary at which water level inside the buildings at each Ringhals unit each safety functions are assumed to withstand before failing. The table primarily deals with a theoretically slow water filling of the buildings where the water level inside gradually increase (not scenario 5 above).

Function	Level Ringhals 1	Level Ringhals 2	Level Ringhals 3 and 4	Comments
Reactivity control	N/A	N/A	N/A	The scram systems for all units are fail-safe designed, which means that the reactivity control should function regardless of the water level.
Pressure relief	>+4.0 m	N/A	N/A	The pressure relief of Ringhals 1 is depending on uninterruptable power supply (batteries) which are located on +7.0 m and the pressure relief is assumed to function until the water level exceeds +4.0 m. Only pressure relief of primary system for Ringhals 2, 3 and 4 (reactors of PWR-type) is reported in the table. For Ringhals 1, 2 and 3 pressure relief will function regardless of water level.
Emergency core cooling	+3.0 m	-3.0 m (-17.0 m. if need for recirculation)	+3.0 m (-17.0 m if need for recirculation)	This function fails when pumps in the safety function is exposed to water. Emergency core cooling for Ringhals 1 is performed by normal system for emergency core cooling. Emergency core cooling for Ringhals 2, 3 and 4 is assumed to be "Bleed and Feed". A need for recirculation occurs when RWST is emptied.
Residual heat removal	+3.0 m	+3.0 m	+3.0 m	This function fails when pumps in the safety function is exposed to water. Residual heat removal for Ringhals 1 is performed by normal system for residual heat removal. Residual heat removal for Ringhals 2, 3 and 4 is assumed to be cooling on the secondary side using the steam generators.

				All units have the possible of cooling by MVSS
Isolation of reactor containment	N/A	N/A	N/A	The isolation valves at all Ringhals units are fail-close and therefore assumed to close even in the event of total loss of electrical power.
Containment pressure and temperature limitation	+3.0 m	-17.0 m	-17.0 m	This function fails when pumps in containment spray system is exposed to water.
Electrical power supply	>+4.0 m	>+4.0 m	>+4.0 m	The gas turbine has a ground level on +9.0 m and is assumed to function until the water level exceeds +4.0 m. All emergency diesel generators are located on ground level
Cooling of spent fuel pools	+3.0 m	+3.0 m	+3.0 m	Some of the system's necessary components are located at ground level, A water level inside the units above +3 m will lead to that cooling of spent fuel pool can't be done with the normal methods. Cooling using manual methods (i.e. hoses) aren't part of the assessments.

<u>**Severe accidents functions**</u>

Flooding leading to core damage is not considered to simultaneously affect equipment that is required for severe accidents. All units are equipped with a number of severe accident functions including equipment for filtered pressure release, Multiple Venturi Scrubber System (MVSS). Filtered depressurization of the reactor containment is of interest during external flooding the sets normal residual heat removal out of order. The MVSS is passive at all Ringhals units and is considered to function regardless of water level.

Water Injection during Severe Accidents supplies water to the containment sprays with the help of mobile pump units that are connected to the system piping for redundant water supply to containment. The mobile pump units have to be able to move around at the site in order for them to perform their function and will, for all Ringhals units, hence function at least until the water reaches to +3.5 m. It should however be noted that there are only two mobile pump units available for the four units.

3.C.2.2 Measures which can be envisaged to increase robustness of the plants against flooding.

Instructions to reduce power at high sea water levels in order to eliminate/reduce the water level difference of 0.7 – 0.8 m between the inlet and the outlet.

Eliminate the possibilities for watering entering the building in case of a sea water level between +3.0 m to +3.3 m. For example installing new doors, improve sealing's etc.

Further analysis of the integrity of doors in order to find and strengthen the weakest doors

Relocating of the switchover valves for cooling of emergency diesel generators to at least ground level would eliminated the possibility of not being able to switch from cooling fresh water to salt water.

3.2 Assessments and conclusions of Flooding

The assessments done by the licensee and the review made by the authority in the area of Flooding has resulted in conclusions and recommendations for further analyses which should be considered as potential measures to increase robustness of the plants.

3.2.1. The licensee assessments and conclusions

The source of the flooding considered is high sea water level. No Swedish unit is located in close proximity to any other water source (such as dams or rivers) than the sea. Waves are not included in the design base flood and are not thought to affect any unit, since no unit is located in direct contact with the sea.

The phenomenon tsunami isn't relevant because Sweden is seismically stable and the surrounding sea is too shallow for a significant tsunami to be created. A seiche (a short term local sea water level change) is a local effect with a short duration that is caused by a local change in pressure that moves at the same speed as a long wave. Tsunamis, seiches, tides and other phenomena are considered to be covered by high sea water levels.

The design basis sea water level is +2.02 m above normal sea water level for Oskarshamn NPP:s, +2,65 m for Ringhals NPP:s and +3,0 m for Forsmarks NPP:s. The frequency of an increase of the sea water level above the design basis flooding is estimated to 10^{-5} annual at the site for Oskarshamn and Ringhals. The design basis flooding at the Forsmark site estimated to 10^{-6} annual. These frequencies are indications that the design bases for flooding are adequate.

All Swedish NPP:s can withstand an external sea water level up to 3.0 m above normal sea water level without fuel damage. External flooding above this level is assumed to result in fuel damage in Forsmark and Oskarshamn NPP:s . Ringhals has analyzed further showing that an external water level of +3.3 m is not expected to cause core damage at any Ringhals unit.

All units are designed to handle the event flooding through high ground water level and all units are equipped with drainage systems to remove ground water from the area between the rock and the buildings. As long as the drainage systems are functioning no flooding due to high ground water level is likely to occur. In case of failure of the drainage function it is assumed that flooding will take many hours before any severe damage will occur to the plants why manual action is likely to be successfully performed. Only Oskarshamn units have drainage systems which supply is backed-up by emergency backup power.

Spent fuel pools will be affected by flooding at the same water level as corresponding reactor.

The containment filtered vent system is an important last line of defense capable of significantly mitigate accident scenarios with loss of several safety functions and core damage. The system is unlikely to be affected by flooding.

3.2.2. Licensees recommendations for potential improvements

The licensees have identified the following recommendations for further evaluations and reassessments.

- The stress test has identified many procedure improvements that can increase the ability to withstand high water levels. A planned shutdown in case the sea water level would reach certain levels is one identified procedure.
- A review should be done in Forsmark and Ringhals to check for gates and other forms of exterior doors to see how well they withstand external water levels above +3.0 m, and make reinforcements if needed. In such an evaluation the command centers should also be taken into consideration.
- Evaluation of how the water is distributed inside the plants during an external flooding
- For Oskarshamn 1 and 2, it could be possible (theoretically) to prevent sea water entering the inlet and outlet tunnels by e.g. high sluice gates and walls around the tunnel openings. Thereby the integrity of buildings would theoretically not be endangered up to an increased sea water level of +6 m.
- Oskarshamn 3: In case the sea water level rises above +3 m, sea water could flow from the Baltic Sea to the yard via the storm water system. It would be possible (theoretically) to avoid such back-flow by installing a shut-off valve in the storm water system.
- Relocating of the switchover valves for cooling of emergency diesel generators to at least ground level, at all Ringhals units, would eliminated the possibility of not being able to switch from cooling fresh water to salt water.

3.2.3. SSM's assessment and conclusions of Flooding

SSM:s overall assessment is that questions specified in the ENSREG document have essentially been answered in an acceptable way and that the assessment and presentation of NPPs ability to cope with the design base flooding and the plant's margins for a flooding event exceeding the design base flooding are analyzed and stressed to fuel damage.

The design base flooding fulfills requirements as specified in the SSM's regulations. However, the effect of waves has not been considered in combination with extreme sea level, in Ringhals and Forsmark. Further evaluations need to be made for the combination of extreme wind and extreme sea level for Ringhals and Forsmark, since historical extreme sea levels are associated with storm, and thus phenomena as wave setup and wave run-up and associated dynamic effect should be considered and investigated combined with extreme sea levels.

High ground water level is of particular concern for Ringhals 2, since the critical internal water level that is assumed to cause fuel damage is three meters below the average sea water level and the drainage pumps run on off-site power in Ringhals, the time frame until this is critical is very uncertain. Further evaluations need to be made for this problem in Ringhals.

The cliff-edge effects have been addressed and presented in essentially a satisfactory way. However, the presentation could have been more complete with regard to the flooding cliff-edges for mobile equipment and access to and from the nuclear power plants.

SSM:s assessment is that the conclusions by the licensees are relevant and reasonable. This is also the case concerning recommendations for further evaluations and/or specific measures which need to be preformed.

4 Extreme weather conditions

4.1 Introduction

In 2004 the first issue of Regulations and General Recommendations concerning "Design and Construction of Nuclear Power Reactors" was released. The regulation stipulates in the 14th paragraph that "The nuclear reactor shall be dimensioned to withstand natural phenomena and other events..."The licensee's interpretation of the14th paragraph is that the plant must be able to withstand a an fast external event such as extreme weather with a frequency of 10^{-4} to 10^{-6} annaully. For each class of natural phenomena that can lead to a radiological accident, there must be an established course of action for the situations where the design values might be exceeded.

The design base events for extreme weather includes, rain, wind, sea water level, outdoor temperature and lightning.

4.A. Forsmark

4.A.1 Design basis

4.A.1.1 Reassessment of weather conditions used as design basis

According to SSMFS 2008:17, nuclear reactors must be dimensioned to withstand natural phenomena and other events that occur outside or inside the facility that can lead to a radiological accident. For such natural phenomena and events, dimensioned values must be determined. Natural phenomena and events with rapid progression where there is no time for protective measures to be taken when they occur shall also be attributed an event class. For each class of natural phenomena that can lead to a radiological accident, there must be an established course of action for the situations where the design values might be exceeded.

Weather phenomena where a warning or an alarm allows the site to take protective actions of operation or to protect the plant against the forces of the external influence may be classified as slow course of events. Weather phenomena where a warning or an alarm cannot be guaranteed are classified as rapid course of events. For the rapid events in the event class, weathering is assumed to occur with a frequency of 10^{-5}/year unless otherwise noted and classified as an H4 event. Normal weather conditions are those that are expected to occur during the plant's lifetime.

Wind

Original design conditions for Forsmark 1 and 2 are derived from Swedish construction standard SBN 67. In the transition to the SSMFS 2008:17 regulation, updated dimensioned requirements for wind load as well as requirements for tornadoes and tornado-generated missiles were added.

At the coastal bands, the highest measured mean wind speed is 40 m/s with estimated gusts up to 55 m/s. Hence the design value of 80 m/s is considered to have a good margin for extreme wind.

Rainfall

Extreme rainfall can cause stress damage to buildings and flood and clog ventilation intakes. An extreme precipitation event is classified according to the SAR as a slow event.

Original design conditions for Forsmark 1 and 2 regarding snow load are derived from the Swedish building code SBN-67. The transition to SSMFS 2008:17 updated the dimensioned requirements for snow load in accordance with the Swedish National Board of Housing, Building and Planning's design regulations BKR 2003. Original design conditions for Forsmark 3 regarding snow load are derived from Swedish building code SBN-75.

Forsmark 1 and 2 stormwater system is dimensioned for 134 l / s on a surface of 1 ha. Forsmark 3 has a stormwater system designed for 1.5 m^3 / s on a surface of 19 ha.

The design snow load recurs each 50 years according to BKR 2003. During the extreme snowfall, the area around Forsmark can achieve the expected snow depth of 1 meter in 1-2 days. For Forsmark 1 and 2, the design values are 2.0 kN/m^2 and for Forsmark 3 1.4 kN/m^2. The design requirement 1.4 kN/m^2 corresponds to a snow depth of 1.4 meters of fresh snow or 1 meter of wind-packed snow. This is a slow event and thus the preventive measures will be rectified by removal of snow before the design load is reached.

For rain, the most relevant risk is probably either that the total amount of rainfall during a period is so large that the ground is saturated, flooding the rock shaft between the reactor building and the ground, or that the instantaneous volume becomes so great that the drainage from roof surfaces is insufficient. The first scenario is covered by flooding of water up to ground level as described in section 3, while the second case is not considered relevant as the building's roof is flat and covered by a PVC sheet, and therefore the risk of a larger amount of water entering is deemed to be low. In the event of a leakage because the PVC sheet is damaged, the probability is very low that more than one safety train will be affected as a result of a good physical separation of safety equipment.

Lightning

The event lightning is classified by SAR as a rapid sequence and belongs to the event class H3. For Forsmark 1 and 2, the design values have the maximum amplitude of 100 kA and a slope of about 80 kA/μs, which corresponds to 98% of all lightning strikes. For Forsmark 3, the design value has the maximum amplitude of 300 kA and a slope of 120 kA/μs, which is derived from a German industrial standard.

The facility is equipped with external lightning protection in the form of a "Faraday cage", which limits the field and lightning currents inside the building. The internal lightning protection is achieved through proper placement and earthing of components so that there is no impact on functionality on the backup safety function.

Studies on Olkiluoto 1 and 2 and at Oskarshamn indicate that a powerful lightning strike down the main stack of the reactor building, which is the highest point. The largest safety implication is a strong lightning impact down the stack to a diesel engine building, which is considered extremely unlikely and would only affect one of four diesel generators.

Outdoor air temperature

The events of extreme air temperatures are classified according to the SAR as a slow process.

Forsmark 1, 2 and 3 are dimensioned according to the Swedish building code SBN-67 with a lowest outdoor temperature of -18 ° C to maintain an acceptable indoor temperature of about +18 ° C. Ventilation systems are designed for +25 ° C. Forsmark 3 applies +32 ° C for safety systems.

Design values for high outdoor air temperature for Forsmark 1, 2 and 3 and low outdoor temperature for Forsmark 3 are probably based on dimensional requirements for ventilation systems.

According to SMHI, the highest and lowest recorded temperatures between 1941 and 2010 were +34.5 ° C and -28.9 ° C respectively at Örskär. The highest temperature ever recorded in Sweden since 1901 is +38 ° C (Ultuna, Uppland on 9 July 1933 and in Målilla, Småland on 29 of June).

Since the commissioning of the plants, temperatures beyond design have occurred. Experience from the plants shows that the plants' safety systems were unaffected on these occasions.

Low seawater level

The event extremely low water level is classified according to the SAR as a slow event.

Forsmark 1 and 2 are designed for extremely low water level +98,1 (2.0 m below mean sea level), and Forsmark 3 for extremely low water level +98.4 (1.7 m below mean sea level).

As regards high seawater level, this is described in the previous stress test work for flooding, see section 3.

Estimates from SMHI indicate a minimum sea level with a recurrence period of 100000 years at 1.7 m below the mean sea level. At low water level, an alarm is received, the power output declines and shutdown occurs before the design values are reached.

Seawater temperature

The event extreme temperature includes frazil ice formation and high and low seawater temperature. The event extreme frazil ice formation is classified by SAR as a fast course and the event belongs to event class H3. The events of extreme water temperatures are classified according to the SAR as slow events.

Design value of the low seawater temperature is +0.5 ° C in front of the intake building. Highest water temperature according to design is +25 °C. Extreme values for highest seawater temperature measured by SMHI are +28 ° C and just below 0 ° C minimum.

In the event of seawater temperatures below 0 ° C, there is a risk of frazil ice formation that can lead to clogging of the intake building. The frequency of frazil ice formation at Forsmark has been estimated at a recurrence interval of 100000 years. The temperature reduction is considered to be a slow event and an alarm will be received at 0.1°C, allowing preventive action to be taken. Forsmark 1, 2 and 3 all have the capability to re-circulate warmer water to the intake canal, which leads to the water temperature being increased by 1 ° C. If there is still a risk of frazil ice formation although the recirculating flow is fully utilized, the flow is reduced by stopping some of the main cooling water pumps.

As with the decreasing seawater temperature, increasing seawater temperature is a slow course and preventive measures will therefore be taken. At the seawater temperature of + 20 °C or higher, the reactor power is reduced and at + 25 ° C the reactor is brought into cold shutdown.

Design value for high and low seawater temperature is then considered to be adequate.

4.A.2 Evaluation of safety margins

4.A.2.1 Estimation of safety margin against extreme weather conditions

Previous calculations of Forsmark 1 and 2 show that there is a resilience against wind loads and missiles estimated for each safety classified building. Generally, there are large margins against wind speeds higher than 80 m/s. In some buildings, the walls and ceilings have lower margins in strength. The corresponding estimate for Forsmark 3 shows that all classified buildings have good margins.

Overall, the high design value of 80 m / s and the margins in strength provide a broad margin for core damage due to extreme winds.

In the event of an extreme tornado and an associated tornado-generated missile, physical separation is credited for cooling systems and diesel generator buildings. The frequency of tornadoes and tornado-generated missiles larger than design values is very low (<1E-6).

Based on a previous calculation, Forsmark 1 and 2 have sufficient strength against snow loads on roofs for each safety classified building. In general, there are large margins. In some buildings, walls and ceilings have smaller margins of strength. In the event of extreme snowfall, there is enough time to remove the snow before the load of snow is too heavy. Therefore, the margins are estimated to be good. The corresponding estimate for Forsmark 3 shows that all safety classified buildings have good load margins.

In extreme rainfall, the capacity of drainage systems, roof well, stormwater systems etc., will be exceeded. This means that water can accumulate on the roof up to the catchments boards without that the dimensional roof load is exceeded. The area around the power plant is designed to drain into ditches and the sea. Rainfall over the design value of 155 mm/day is therefore no further challenge for the buildings and safety functions.

The likelihood of lightning knocking out equipment through a strike in a diesel generator stack is very low according to the study done on Olkiluoto 1 and 2. A loss of safety systems due to a lightning strike beyond design is considered to be of very low probability.

Extreme outdoor air temperatures occur as a slow course case where there is time for preventive manual intervention. At low temperatures, the fans to the ventilation system are shut off to avoid bringing cold air into the plants.

Ageing speed of certain components is accelerated at temperatures over design. Measures for this are covered by the plant maintenance programme. In light of the above, the facilities have good margins against extreme outdoor temperatures.

Consequences of extremely low sea level will lead to the same scenario as the loss of primary heat sink in a sea level brought down to 5 m below mean sea level. This is considered unlikely.

In the extremely low seawater temperatures considered, the plants have good margins seeing that this is a slow process and the plants can be brought down to a safe mode.

Lower seawater temperatures than 0° C result in no greater risk of frazil ice formation than the design basis accounted for. The opportunities available for circulating the water and taking water from the outlet channel give good margins to prevent this occurrence.

4.A.2.2 Measures which can be envisaged to increase robustness of the plants against extreme weather conditions

For the weather phenomenon that is described, the margins are considered to be adequate. In addition, investigations are recommended to evaluate the overall risk of extreme weather events and combined effects of weather phenomena that plants may be exposed to outside design.

The following proposed actions are identified to increase the robustness against extreme weather conditions:

- Complete procedures for the deficiencies found during the "stress test", in particular, to draw up a procedure for external impact on the Forsmark 3 unit.
- Until 31 March 2012, investigate which combinations of events outside the design, together with extreme weather events, that need further attention. Furthermore, the

investigation must state which combinations of weather events that need to be studied. The aim is to evaluate the overall risk of these effects.

4.B. Oskarshamn

4.B.1 Design basis

4.B.1.1 Reassessment of weather conditions used as design basis

One prerequisite applied in the stress tests performed for Oskarshamn 1, 2 and 3 is that "Bad weather" is interpreted as weather conditions with a recurrence of 100 year. Weather conditions with frequencies 10^{-2} annual belongs to event category H2 in accordance with the SAR. Weather conditions classified as event category H2 may indeed compare to "bad weather".

The formal requirement to consider phenomenon such as severe weather conditions are specified in the Swedish regulation SSMFS 2008:17 §14.

"A review of design basis parameters for external events and completeness of any measures due to the review should be performed in accordance with §14 at the latest of 31th of December 2012."

Based on SAR general part chapter 3 and the requirements in SSMFS 2008:17 §14, the weather conditions and weather caused phenomenon that should be analysed within the frame of SAR general part. In accordance with the transition plan for SSMFS 2008:17 a review of design parameters has been performed. A new edition of SAR has not yet been established regarding severe weather conditions. However this is foreseen to take place in the near future, hence the design basis parameters and requirements are considered in this report.

In accordance with SAR severe weather conditions correspond to weather conditions classified as event category H4. The prerequisites applied in the stress tests for severe weather within design base is consistent with the prerequisites used within SAR and may be summarized as follows:

- The denomination of events such as severe weather conditions is in SAR specified as "Extreme external effect".
- The acceptance criterion for events classified "Extreme external effect" are specified in SAR.
- Concurrent with any "Extreme external effect" parameters of other conditions are assumed to correspond to parameters with recurrence frequency 10^{-2} annual, if not analyses shows that some combinations are physical impossible. "Extreme external effect" is assumed to occur with a recurrence frequency of 10^{-5} annual, if not other is stated.

<u>Extreme precipitation</u>

The extreme precipitations are based on available statistics. A precipitation of 150-180 mm/day corresponds to a recurrence frequency of 10^{-4} per year at the Simpevarp site. For values above this the uncertainties will increase significantly. As an extreme value with a recurrence frequency of 10^{-5} – 10^{-6} per year at site a precipitation of 400 mm/day has been regarded.

The design basis parameters concerning extreme precipitation are 400 mm/day and with a maximum intensity of 25 mm during 10 minutes.

Buildings where equipment important to nuclear safety are located, and which have a flat roof construction with surrounding roof ledges, the load effects of extreme precipitation must be considered.

At Oskarshamn 1 and 2 have roofs that are slightly inclined with ledges that are 0.7 m, 0.3 m and some buildings have no ledges.

The roofs are slightly inclined but as a conservative approach the buildings are assumed to have flat roofs where water can be collected.

The buildings with flat roof design and ledges on Oskarshamn 2 have introduced overflow pipes, about 0.1 m above the roof, in the ledges around roofs. This measure was introduced after analysis showed that the roofs could not bear the load of water in case of a blockage in the drainpipes.

The roof of the reactorbuilding of Oskarshamn 1 has ledges of height 0.7 m and the roof can withstand a load of 2 kN/m^2. Under the assumption that no water is drained through the drainpipes the water height will exceed the allowable after 12 hours. This observation may results in more careful calculations of the roof or review and completion of operational and maintenance procedures to prohibit unallowable accumulation of water on the roof.

All buildings at Oskarshamn 3 have a flat roof design with surrounding ledges. The roofs are slightly inclined but as a conservative approach the buildings are assumed to have flat roofs. The ledges vary due to size and inclination but are at most approximately 1 m. The design basis precipitation results in a load of 4 kN/m^2 after 24 hours, with the conservative assumption that no rainwater is drawn off by the drainpipes. The flat roofs are designed to withstand a load of 5 kN/m^2 and the beams are designed against a load of 10 kN/m^2. The ledges surrounding the flat roofs are less than 1 m. Therefore, collapse of the flat roofs due to extreme precipitation can be precluded.

It is assumed that manually measures may be credited after 24 hours to draw of rainwater from the roofs e.g. manual rinsing the gutters or provide mobile drainage pumps.

The reactor building of Oskarshamn 1 is erected in a shaft which is blasted into the rock. Rainwater may enter the area between the reactor building and the surrounding rock in case of excessive precipitation causing an external pressure on the outer walls of the building. The reactor building is verified to withstand an outer pressure corresponding to an increased water level of approximately 12 meters in the shaft. The yard around the plant is situated high compared to the surroundings and therefore the amount of rain water that may entering the shaft is judged to be limited and significant less compared to the capacity of the drainage pumps which have diesel backed power supply. The electric control building and the new pump building are not erected in a shaft and are therefore not affected by a potential outer water pressure.

The reactor building of Oskarshamn 2 is erected in a shaft which is blasted in the rock. Rainwater may enter the rockshaft by a ventilation opening in the southwest corner of the building. The opening is 1x1 meter and has a concrete ledge of 0.2 m. Otherwise the ground is watertight connected to the building. The reactor building is verified to withstand an outer pressure corresponding to a water level 6 meters in the shaft.

The ground is watertight connected to the building and therefore the amount of rainwater entering the shaft through the one ventilation opening is judged to be limited and significant less compared to the capacity of the drainage pumps. The turbine-, screen house- and diesel buildings are also erected in a blast out rock shaft surrounding the buildings. The ground is watertight connected to these buildings and rainwater cannot enter. The other buildings are installed on the ground. The integrity of the buildings important to safety is not jeopardized in case of design basis precipitation.

At Oskarshamn 3 flooding of the yard may occur in case of an assumed precipitation of 0.9 liters/(min, m^2) concurrent as the storm water system is used for emergency cooling with a sea water flow of 2 m^3/s. However, the mass flow that has to be drawn of by the storm water system due to the design basis precipitation is 0.28 liters/(min, m^2). Therefore flooding of the yard is avoided.

Extreme air temperatures

Changes of the air temperatures are a protracted lapse. Thereby it is possible to envisage when safety related limits may be exceeded. For that reason no extreme air temperatures are considered as "Extreme external effect".

The highest and lowest temperatures that are considered possible at site has been calculated based on meteorological data. According to SAR chapter 3 these are as follows:

Location	High temperature [°C]	Low temperature [°C]
Oskarshamn	36.3	-37.2
North cape of Öland	33.4	-27.0
Västervik	37.4	-35.0
Kalmar	37.6	-30.0

Measured temperatures in the vicinity of Oskarshamn.

The concern that is relevant in case of high air temperature is that electrical equipment may be affected. The main consequence of electrical components exposed to ambient temperature that exceeds the temperature for which the component is qualified could be an accelerated ageing. Such a consequence is judged to be of no concern considering the ability to reach and maintain a safe shutdown condition.

However in case of temperatures that are considerable beyond the temperature for which a specific component is qualified malfunction may not be precluded.

At Oskarshamn 1 the majority of the safety electrical components are located in the electrical control building (EKB). The ventilation system for the building is designed to maintain the temperature in the building within allowable limits as long as the outside temperature in the range of -40°C to 40°C. The electrical components in ventilation system have either diesel or battery backed power supply. In case of a prolonged outside air temperature exceeding +40°C the ambient temperature will exceed the design temperature in the same range as the air temperature exceeds +40°C. Engineering judgment is that this only may cause a slightly accelerated ageing of concerned equipment.

The most important building at Oskarshamn 2 is the main electrical building housing them majority of the safety related electrical and I&C equipment. The air in the main electrical building is cooled and ventilated. The ventilation system is designed to maintain the ambient air temperature within allowable limits up to an outside air temperate of 25°C. In case of a prolonged outside air temperature exceeding 25°C the ambient temperature may be assumed to exceed the allowable ambient temperatures in the same range as the air temperature exceeds +25°C. In case of an assumed prolonged outside air temperature of approximately +40°C it is therefore reasonable to assume that the ambient air temperature in the rooms does not exceeds allowable limits with more than 10°C.

The ventilation systems serving the control building, the diesel buildings

and the auxiliary system buildings of Oskarshamn 3 are designed in such a manner that the maximum allowable temperature in areas, were safety classified electrical equipment are located, will not be exceeded in case the outside air temperature is 32°C. In case of an assumed prolonged outside air temperature of approximately +40°C it is reasonable to assume that the ambient air temperature in the rooms does not exceeds allowable limits with more than 5°C. An engineering judgment is that this only may cause accelerated ageing of concerned equipment. The temperature limit derives from the original Technical Requirement of Electrical equipment (TRE). In the TRE it is also stated that the electrical equipment shall not be damaged at an ambient temperature of 55°C

In general for rooms and buildings cooled or ventilated by operational system, which could be postulated inoperable following an initiating event, limitation of the ambient air temperature could be provisional arranged by manual measures e.g. opening of doors.

The concern in case of low air temperature is that some essential process measurement piping could run the risk of freezing and thereby indicate inaccurate values to the reactor protection system. This could be caused in the event of malfunction of the ordinary ventilation systems in the reactor building when the heating system is assumed to fail and when very low outside temperature exists. In case of low temperature alarm in the ventilation system, manual measures shall be performed in accordance with the procedure to ensure that freezing of concerned components are avoided. In this case it is to manually stop the inflow air system and at some building to increase the surveillance.

Snow conditions

Buildings should be designed to withstand snow loads in accordance with the Swedish National Board of Housing, Building and Planning, BSV 97. In BSV 97 the loads to be considered due to snow in Sweden are presented.

The snow loads to be considered for the Simpevarp site are:

- The snow load basic value on ground S0 = 2.5 kN/m^2

- A snow load 2.5 kN/ m^2 corresponds to a snow depth of 0.9 m assuming a snow density of 280 kg/m^3.

The event is protracted in time why manual measures can be carried out to avoid extreme accumulation of snow. For that reason no extreme snow loads, beyond what is prescribed in BSV 97, are considered as "Extreme external effect".

Ridges and roof beams of Oskarshamn 1 and 2 do not fulfill the requirements in BKR 12 when assuming snow loads in accordance with BSV 97. This could be apprehended because the roof of the reactor building of Oskarshamn 1 is designed in accordance with the rules in BABS 1960 (Kungliga Byggnadsstyrelsens Publikationer 1960:1).

The roof of the reactor building of Oskarshamn 2 is designed in accordance with the rules in SBN67 (Svensk Bygg Norm 67 Föreskrifter).

The snow loads in BSV 97 are approximately twice the snow load prescribed in BABS 1960 and in BKR 12. However, it is judged that the snow height applied in the calculations is not probable to occur on the roof because of the roof's design with low ledges and that the roof is exposed to wind.

There is an analysis that show in case of an intense snowfall concurrent with wind snow could accumulate in snowdrifts which may jeopardize the integrity of the roof of the turbine building of Oskarshamn 1. This may cause down falling parts from the roof and endanger components e.g. the main cooling water pumps installed below the roof in the screen house building. However, safety classified sea water cooling pumps are located in the adjacent new pump building and are therefore not affected.

This issue is taken under consideration and will be handled in an appropriate manner in accordance with the transition plan for SSMFS 2008:17 §14.

The reactor building, control building, auxiliary system buildings and diesel buildings of Oskarshamn 3 are verified to withstand snow loads in accordance to BSV 97.

Tornado and tornado induced missiles

The design basis tornado and tornado induced missiles are in accordance with US NRC Regulatory Guide 1.76 Rev 1. Based on the results presented by the Swedish Meteorological and Hydrological Institute (SMHI) of estimated wind velocities due to tornados that may occur in Sweden, RG 1.76 Rev 1 Tornado Region III has been considered as appropriate to assume as design basis.

The frequency of a tornado wind velocity of 25-70 m/s is according to SAR 10^{-5} per year at site. The estimation is based on results presented by SMHI and is presented in SAR.

The maximum wind velocity is 72 m/s in accordance with Regulatory Guide 1.76 Rev 1 for Tornado Region III.

The design basis tornado induced missiles are in accordance with the following:

Missile type	**Dimension**	**Mass**	**Horizontal velocity** [m/s]
Solid steel sphere	Ø25 mm	0,07	6 m/s (for all levels)
Steel pipe	Ø168 mm, L=4580 mm	130	24 m/s (for all levels)
Automobile	4.5 m x 1.7 m x 1.5 m	1178	24 m/s (for levels lower than 13 m above ground level)

The requirement which shall fulfill in case of a design basis tornado and tornado induced missiles is as follows:

Safety functions shall not be affected to such an extent that the ability to bring and maintain the reactor in a safe shut down condition is jeopardized. Buildings or part of buildings where required components are located shall withstand the effect of the design basis tornado and tornado induced missiles.

Oskarshamn 1

It should be noticed that Oskarshamn 1 was originally built without consideration of loads resulting from "Extreme external effects" such as tornado and tornado induced missiles. The applied design rules were mainly BABS 1960. However the EKB building erected within the safety modernization of the unit is designed to withstand tornado and tornado induced missiles.

The outer roof of the reactor building at Oskarshamn 1 does not fulfill the acceptance criteria of the BKR 12, when postulating a wind velocity of 72 m/s. The same concerns the western and eastern outer walls above +137.5 m (+106 m corresponds to ground level) when postulating a wind velocity of 72 m/s. This could cause that beams and/or parts of the wall structure falls down on the reactor hall floor or into the storage pool for spent fuel.

Further it may conservatively be assumed that the auxiliary condenser is affected because some parts are located on elevation 137.5 or higher.

Concerning consequences of down falling objects it is judged that those are covered by the performed analysis of postulated load drop events. With respect to the integrity of the storage pool, analysis for Oskarshamn 2 shows that only minor leakage (about 130 l/hour) may occur in case a lock-gate of 1200 kg is dropped in the storage pool. The design of the storage pool in Oskarshamn 1 is similar the one in Oskarshamn 2. A leakage of such order can be compensated by e. g the de-ionized water system or the fire water system or by using fire trucks.

In case spent fuel elements are mechanically damaged the possible released radioactivity to the environment is judged to be considerable less compared to the amount that could be released in case the shroud head is dropped onto the core grid during refueling outages. In the analysis of a postulated dropped shroud head onto the core grid conservatively 67 fuel elements are assumed to be mechanically damaged, but even then the released radioactivity to the environment are within the prescribed limit.

The reactor building is located in such a way that only solid steel sphere or steel pipe missiles could impact the building. Such missiles could penetrate the outer walls and affect components located in the proximity of the wall. However, this is a very local effect and due to the physical separation of required safety functions such missiles will not jeopardize the ability to reach and maintain a safe shut down condition.

The electrical control building (EKB) is designed with the same prerequisites that were used for Oskarshamn 3, i.e. a maximum wind velocity of 107 m/s and tornado induced missiles according US NRC Regulatory Guide 1.76 Rev 0.

Systems and components installed in EKB are therefore not jeopardized in case of a design basis tornado or tornado induced missiles. The emergency diesel generators in sub division A and B are located in EKB. The diesel generators are cooled by the cooling chain using the Baltic Sea as the ultimate heat sink. Parts of the cooling system are located in the new pump building adjacent to the Screen house. The roof and walls of the new pump building are not verified to withstand penetration of a steel pipe missile impact. However, the stability of the structure is not jeopardized. Further, impact of an automobile may jeopardize the stability but not penetrate the roof and walls. The pumps and cables for power supply are installed below the ground level and are not judged to be jeopardized by the missiles or down falling objects. Therefore the cooling of the emergency diesel generators in sub division A and B are judged to be ensured.

Each diesel generator is provided with a diesel storage tank (installed in EKB) with a capacity for 24 hours full power operation. After 24 hours, manual measures are credited for refilling the diesel storage tanks from e. g. mobile fuel trucks.

The reactivity control systems, emergency core cooling and residual heat removal are not jeopardized by a design basis tornado and tornado induced missiles.

The residual heat from the storage pool for spent fuel has to be removed in the long term. In case the operational cooling system is inoperable the storage pool can manually be connected to the cooling chain for cooling reactor pressure vessel enable residual heat removal from both the spent fuel pool and the reactor pressure vessel. The manual interconnection to the cooling chain is not required within 40 hours in case the initiating event is assumed during power operation. If the event is assumed during refueling and maintenance outage after the full core has been offloaded to the storage pool, the interconnection may conservatively be required approximately 5 hours following the initiating event.

<u>Oskarshamn 2</u>

It should be noticed that Oskarshamn 2 was originally designed in accordance with the design rules in SBN 67. In addition to loads prescribed in SBN 67 loads due to act of war were considered. The roof beams and steel plates of the outer roof fulfill the requirement in BKR 12 when postulating a wind velocity of 72 m/s.

It is assumed that the scenario for tornado induced missiles that hit the reactor building is similar as described for Oskarshamn 1 above.

The spent fuel pool cooling system is not judged to be affected. However, even if the ordinary spent fuel pool cooling systems are postulated to be inoperable the cooling can be accomplished by feed and bleed using the fire water system. This is described in the stress test for loss of UHS, see section 5.

The safety functions, reactivity control, emergency core cooling and residual heat removal are not affected by down falling objects on the reactor hall floor or into the storage pools for spent fuel.

The ridges and steel plates of the roof of the turbine building do not fulfill the requirement in BKR 12 when postulating a wind velocity of 72 m/s. The roof beams and walls fulfill the requirement in BKR 12. Further, it is judged that the outer roof does not withstand impact of steel pipe missile. The high pressure core injection system (HPI) is installed in such a way that it is protected against down falling object above slabs. However, the water storage tank supplying HPI could be affected by a missile. Therefore it may conservatively be assumed that the high pressure core injection is affected. The core cooling can be accomplished with the emergency core cooling system.

The roof beams of the screen house building do not fulfill the requirement in BKR 12 when postulating a wind velocity of 72 m/s. Further, it is judged that the outer roof does not withstand impact of steel pipe missile. Down falling objects from the roof may damage the operability of pumps for residual heat removal and the function could be affected. If the residual heat removal function is postulated inoperable, residual heat removal may be accomplished using the mitigation system. This is described in the stress test for loss of UHS, see section 5. This issue is handled in the ongoing modifications in accordance with the transition plan for SSMFS 2008:17 §14.

The roof and walls of the diesel building fulfill the requirement in BKR 12 when postulating a design basis tornado and tornado induced missiles. The operability of the emergency diesel generators is therefore ensured.

Each diesel generator is provided with a diesel storage tank (installed in the diesel building) with a capacity for 8 hours full power operation. After 8 hours, manual measures are credited for refilling the diesel storage tanks.

The roof and walls of the main electrical building fulfill the requirement in BKR 12 when postulating a design basis tornado and tornado induced missiles. In the main electrical building essential safety electrical and I&C equipment's are located.

The gas turbine building is not judged to withstand a design basis tornado and tornado induced missiles.

The conclusion is that the ability to reach and maintain a safe shutdown condition is not jeopardized in case of a design basis tornado and tornado induced missile.

Down falling objects from the roof of the screen house building may jeopardize the operability of the pumps for residual heat removal. Therefore the residual heat removal function could be affected. If the residual heat removal function is postulated inoperable, residual heat removal may be accomplished using the mitigation system. However, using the mitigation systems for residual heat removal is not formally permissible within design basis. This issue is handled within the ongoing modifications in accordance with the transition plan for SSMFS 2008:17 §14.

<u>Oskarshamn 3</u>

The buildings of Oskarshamn 3 where safety classified structures, systems and components, required to bring and maintain the reactor to a safe shut down condition are located are verified to withstand a design basis tornado and tornado induced missiles. These buildings are the reactor building, control building, auxiliary system buildings and diesel buildings. A design basis tornado and tornado induced missiles will not jeopardize the ability to reach and maintain the reactor in a safe shut down condition.

Extreme lightning due to thunderstorm

The design basis for lightning at Oskarshamn 1 and Oskarshamn 2 is peak current, 110 kA and a peak current derivative, 80 kA/µs.

The design basis for lightning at Oskarshamn 3 is peak current, 300 kA and a peak current derivative, 120 kA/µs

The lightning density at the Simpevarp region is calculated to 0.18 lightning/km^2/year. The average numbers of thunderstorm days in the Simpevarp region are 8 days/year. The positive strokes are harder compared to negative strokes and are therefore those who constitute the base in the design of lightning protection.

Direct lightning strokes are less probably and according to IEC 1024-1 and statistics the estimated frequency for a direct stroke to the buildings are as follows:

- 1 negative stroke every 14 year
- 1 positive stroke every 208 year
- 1 stroke with a current above 100 kA every 1764 year.

According to safety assessment which concerns Ringhals (located about 20 km north of the town Varberg in Sweden), a 200 kA stroke has a frequency of 10^{-5}/year. It is judged that this frequency is also valid for the Simpevarp region.

The external events solar magnetic activity and EMC (electromagnetic compatibility) are enveloped by the event extreme lightning with respect to consequences. EMC is defined as "it neither causes, nor is susceptible to, electromagnetic interference (within the limits of applicable standards)". These disturbances are generated in other electrical equipment and are limited to values for equipment common used in the society. The internal grounding system installed at the plant, to handle the disturbances from lightning, is more than sufficient to handle EMC disturbances from electrical equipment. Solar magnetic activities influence the high voltage system by high "DC-like geomagnetic current" which saturated transformers and a loss of off-site power may occur in a worst case.

However, the on-site power system will not be affected. The lightning protection system protects the on-site power systems from geomagnetic currents.

Plant capabilities to handle design basis parameters for the existing lightning protection at Oskarshamn nuclear power plants are of acceptable status. The I&C equipment and cables are located and installed in such a way that lightning is not judged to affect the nuclear safety in an unwarranted way.

4.B.2 Evaluation of safety margins

4.B.2.1 Estimation of safety margin against extreme weather conditions

Extreme precipitation

Since the design basis has been established with a frequency of 10^{-5}–10^{-6} /year, it can be assumed to be a level that will not be possible to exceed. If the precipitation increases above this conservatively used design basis the water will flow over the ledge of the flat roof buildings to the plant ground level. Even if the ground level gutters should be completely blocked the water flows down to the Baltic Sea. This may cause an increase of water entering the rock shaft around the reactor building. However, the drainage capacity in the rock shaft of Oskarhamn 1 and at Oskarhamn 2 is sufficient to handle such an increase of water entering the rock shaft.

In Oskarshamn 1 reactivity control, core cooling and residual heat removal is thus unaffected even if the ground sewage system is totally clogged and a precipitation is stressed because of:

- EKB building is based on ground level and water tight to 0.3 m above ground level.
- New pump building is under ground and water tight.
- Rock shaft surrounding the reactor building has excessive drain capacity.
- Reactor building roof will not collapse and the needed reactor safety system for hydraulic scram, core cooling and residual heat removal could function.

In Oskarshamn 2 reactivity control, core cooling and residual heat removal is thus unaffected even if the ground sewage system is totally clogged and a precipitation is stressed because of:

- Flat roofs have flooding protection by overflow pipes in the ledges around the roof at about 0.1 m above the roof level.

- Ground inclination will not cause flooding of buildings.
- Rock shaft surrounding the reactor building has excessive drain capacity and only one opening with a 0.2 m high ledge. Even if all the water on the roof of the reactor building is assumed to enter the shaft around the reactor building, the rockshaft could be pumped out because the drainage capacity of three pumps.

The surrounding ground of Oskarshamn 3 is in contrast to Oskarshamn 1 and 2 blasted down to the elevation of 103 m because the twin unit Forsmark 3 had this ground elevation. Excessive precipitation is therefore more sensitive at Oskarshamn 3 than in Oskarshamn 1 and 2.

However, in case of a postulated precipitation beyond three times the design basis this may cause flooding of the yard. If the yard is flooded it is assumed that water will enter safety classified buildings. The scenario of a flooded yard is described in the stress part flooding, see section 3.

The ability to shut down and maintain the unit cold shut down condition is possible even if precipitation is stressed to about three times design basis values because of:

- Flat roofs have flooding protection by overflow pipes in the roof at about 0.1 m above the roof level.
- Flat roofs can take the load from water up to the ledge around the roof without collapsing even if the roof drains are blocked.
- Drainage capacity of the yard around the buildings is about three times design basis for precipitation.
- If rock shafts surrounding buildings are flooded they can withstand the load and are water tight up to the ground elevation.

Extreme air temperatures

High and low temperatures exceeding design limits +40°C/-40°C are slow processes and the unit could be shut down before design limits will occur. Engineering judgment is that the air temperature at site could not physically be more extreme compared to what has been considered in design basis section 4.B.1. Outside temperatures considerable beyond + 40°C or -40°C is therefore not evaluated.

Engineering judgment of temperature below -40°C in combination with multiple failures in the heating systems is handled by alarms and manual shut off normal ventilation and start of emergency ventilation. This situation is within the safety analysis report. An analysis performed in Oskarshamn 2 to study the effects when all room cooling or heating are lost. Two scenarios were studied when the outdoor temperature was +25°C and when the outdoor temperature was -16°C. The result of the analysis was that there is enough time for manual actions. Similar results could be expected for Oskarshamn 1 and 3.

The support systems located outside are fire water tanks, de-ionized water tanks, drinking water tanks and storage tanks for diesel fuel (day tanks for 24 hours operation are inside the building). The outdoor water tanks have electrical heaters while the diesel fuel is heated by an electrical boiler via a heat exchanger. The time until the liquid in the outdoor tanks will be too low for the function is difficult to determine but engineering judgment is that the process is slow and there will be time enough to manual actions.

Extreme snowfall

Extreme snow fall must be considered in two aspects normal heavy snow fall and a combination of snow with strong wind.

The first aspect is snow falling above the design basis event. In this case the snowfall is a slow process and accumulation over 24 hours is of interest. Accumulated snow depth for the south part of Sweden in a 24 hours period is 1 m is considered to be an extreme weather value. If the value is increased to 2 m as an upper limit for stress test only the EKB roof can handle the load at Oskarshamn 1. In such a case manual measures are relevant to consider because of the slow process. Manual measures can be performed in different ways either by manual shovelling of snow or using snow blowers. Each day, at the daily operations meeting, weather forecast is considered and any warnings and risk categorization from Swedish Meteorological and Hydrological Institute (SMHI) is communicated. Actions can therefore be taken to consider severe weather

The second aspect is when snow in combination of strong winds could accumulate in snowdrifts and/or clogging of air intakes. In this aspect it is considered to be extremely difficult to perform manual measures outdoors e.g. manual shovelling of snow. Therefore such manually measures are not credited.

At Oskarshamn 1 snowdrifts cannot accumulate on roofs above the roof's ledges of the emergency control building (EKB), reactor building and reactor control building building, because those buildings are freestanding. Other buildings might be jeopardized by snow drifts above design limits. Core cooling is maintained by the emergency core cooling system and residual heat removal is performed by safety relief valves and water blowing valves transferring the residual heat to the condensation pool. The situation will essentially be the same as after an earthquake, see section 2. The residual heat from the storage pool for spent fuel has to be removed in the long term. In case the operational cooling system is inoperable the storage pool can manually be connected to the reactor cooling chain.

During prolonged periods of snow fall there is a risk that the filter may also be clogged by large amount of snow accumulating in the outside air intake ducts. In such a case the ventilation system must be manually switched over to recirculation mode. This is a scenario within SAR and the unit can safely shutdown. The conclusion can be withdrawn for all units at Oskarshamn.

Oskarshamn 1 can be safely shut down and maintained in a safe condition in the worst case by sub division A or B.

At Oskarshamn 2 snowdrifts cannot accumulate on roofs above the roof's ledges of the reactor building, the independent electrical building for high pressure injection, the diesel generator building or the scrubber building because the buildings are freestanding. Other buildings might be jeopardized by snow drifts above design limits such as screen house building for cooling water, the turbine building and the electrical system building.

A conservative approach would be that roof collapse will damage the electrical buildings so the functions in that building are completely lost. It is also assumed the gas turbines and the outer grid cannot withstand stress test snowstorms.

Ongoing modification of Oskarshamn 2 will eliminate the situation because the roof of the screen house building will be reinforced to handle excessive loads. Furthermore, a new residual heat removal system taking water from the outlet will be installed in a separate building on the north side of the unit. This is according to the transition ruled for SSMFS 2008:17 §§ 10 and 14. The modifications will be finalized in the spring of 2013.

Because there is no residual heat removal the scenario will be the same as described in the stress test for loss of ultimate heat sink, see section 5. Core cooling will be performed by the high pressure injection system powered from diesel generators and controlled from the independent electrical building. The independent electrical building was erected to handle core cooling when the whole main electrical buildings were assumed to be lost. The residual heat will be removed by the boiling in the condensation pool and boiling in the scrubber pool. There will be no release of radioactivity to the environment.

The residual heat from the storage pool for spent fuel has to be removed in the long term. In the case all sea water cooling is lost by the collapse of the screen house building roof the situation will be the same as described in the stress test for loss of ultimate heat sink, see section 5. Water could be added to compensate the boiled off water in the refuelling pool from the fire system. All systems which are able to cool the spent fuel pool are dependent on the main electrical building.

For Oskarsham 3 the conclusion for the stress test of snow and snowstorm is that it is a relatively slow process. Weather forecast is communicated within the organization and actions are established. Stressing the weather situation beyond plant design basis has demonstrated the Oskarshamn 3 can be safely shut down and maintained in a safe condition. In the worst case the ultimate heat sink and diesel power is lost in two trains (out of four) but this will not affect the safe shut down and maintaining the unit in a safe shut down condition. Normal ventilation, gas turbines and off site power are lost. Oskarshamn 3 has a robust design even at stress test beyond design basis for snow, snowstorm and snowdrift.

Tornado and tornado induced missiles

When postulating tornado and tornado induced missiles beyond design basis it is reasonable to assume that the consequences will not differ significantly compared to what is described within design basis. The effect of a tornado induced missiles are of its nature a local effect. Therefore it is reasonable to assume that the consequences with respect to structures, systems and components roughly are the same as discussed for design basis missiles.

At Oskarshamn 1 the probability that concerned structures, systems and components actually would be affected will increase. In case the tornado wind velocity is postulated to exceed the design basis it is reasonable to assume that the consequences with respect to structures, systems and components roughly are the same as discussed for design basis. Within design basis it is assumed that structures/buildings that either not fulfill acceptance criteria in the BKR or are unanalysed may be damaged. This is a conservative approach. When assuming tornado wind velocities beyond design basis the probability that concerned structures, systems and components actually would be affected will increase.

The EKB building is designed to withstand a tornado wind velocity of 107 m/s. This is considered as a robust margin against a hypothetic tornado of extreme magnitude. The design basis tornado wind velocity is 72 m/s.

The conclusion for Oskarshamn 1 is that the ability to reach and maintain a safe shutdown condition is not deteriorated in a significant way in case of a beyond design basis tornado or tornado induced missile.

At Oskarshamn 2 the identified structures that do not fulfill the requirements in BKR 12 for design basis tornado wind are the roofs of the reactor-, turbine and screen house buildings. The consequences with respect to systems and components that may be affected in case of beyond design basis tornado winds is judged to be roughly the same as discussed for design basis tornado wind.

The structures that fulfil the requirements in BKR 12 for design basis tornado and tornado induced missiles will maintain their integrity also when assuming beyond design basis tornado. This is not judged reasonable to perform considering the ongoing modifications within projects to fulfil requirements of SSMFS 2008:17.

Within design basis refilling of the diesel storage tanks are credited after 8 hours by manual measures.

The conclusion is that the ability for Oskarshamn 2 to reach and maintain a safe shutdown condition is not deteriorated in a significant way in case of a beyond design basis tornado or tornado induced missile.

For Oskarshamn 1 and 2 the design basis refilling of the diesel storage tanks are credited after 8 hours by manual measures. Refilling of the diesel storage tanks is normally performed from the gas turbine storage tanks. It is also possible to refill the diesel storage tanks from e. g. mobile fuel trucks using locally pipe connections to the tanks. A tornado induced missile may impact and cause leakage from a gas turbine storage tank. However, it may not be precluded that such impact could cause a fire that might jeopardize also the adjacent gas turbine storage tank. Further, a tornado wind may affect the ability for fuel mobile trucks to reach the site. Therefore it is essential that there is a sufficient storage of diesel oil in the large oil storage tanks.

Oskarshamn 3 was originally designed against tornado and tornado induced missiles in accordance with US NRC Regulatory Guide 1.76 Rev 0, Tornado region III. The maximum wind velocity is 107 m/s according to RG 1.76 Rev 0, Tornado Region III. Further, the assumed missiles are severer in RG 1.76 Rev 0.

The frequency of a tornado with a maximum wind velocity of 107 m/s is probably less than 10^{-7} per year at site. Because of the magnitude of the original design basis tornado the robustness against tornado and tornado induced missiles is high. Further, the effects of a tornado induced missile are of its nature a local effect. Considering the redundancy and physical separation of required systems and components an extreme severe beyond design basis missile is not judged to jeopardize the possibility to bring and maintain the reactor in a safe shut down condition.

Extreme lightning due to thunderstorm

In case a stronger lightning stroke than 110 kA hits the EKB-building at Oskarshamn 1 with a direct stroke, the induced voltage in cables and electronics can cause some damage in the logics of pumps, valves and protection equipment. In EKB-building the physical separation is fully implemented between the sub divisions. Therefore only one sub division could be affected.

If a stronger lightning stroke than 110 kA hits the main electrical building at Oskarshamn 2 with a direct stroke, the induced voltage in cables and electronics can cause some damage in the logics of pumps, valves and protection equipment. In the main electrical building the physical separation is fully implemented between the sub divisions A/C and B/D. Therefore only two sub divisions could be affected. Two sub divisions are enough to shut down the plant and maintain it in cold condition.

If a stronger lightning stroke than 300 kA hits the main electrical building at Oskarshamn 3 with a direct stroke, the induced voltage in cables and electronics can cause some damage in the logics of pumps, valves and protection equipment. In Oskarshamn 3 the physical separation is fully implemented between sub divisions A, B, C and D. One sub divisions is enough to shut down the plant and maintain it in cold condition. In general Oskarshamn 3 has installed electrical equipment important to safety with a distance of 2-3 m from the outside wall which gives a safe distance to induced electrical disturbances

Ringhals 1 has made a study together with Uppsala University about their lightning protection system. In the investigation it assumed that a 400kA stroke hits the main stack. The main stack function can be seen as an umbrella for the buildings surrounding the stack. The report shows how much the stack, the grounding under the stack and the surface of the buildings with steel, reduces the current peak in the building containing electrical and I&C equipment.

The results from the studies is that even with a stroke more severe than ever known in the world (the most powerful stroke ever been registered was 386 kA in Japan), the stack and other lightning protection precautions reduces the current peak to a level lower than the design levels. It is considered extremely improbable that a stroke of such magnitude or higher could hit the main stack of any unit in Oskarshamn. The frequency of such an event is probably less than 10^{-7}/year.

4.B.2.2 Measures which can be envisaged to increase robustness of the plants against extreme weather conditions

During the stress tests the licensee holder OKG has found the following measures that can increase the robustness of the plants against extreme weather conditions.

Review and completion of operational and maintenance procedures to prohibit unallowable accumulation of water and snow on some identified roofs.

To secure long time operation of safety diesel generators at Oskarshamn 1 and 2, a review and completion of procedures shall be performed to ensure that a sufficient storage of diesel oil is stored in the large oil storage tanks. A tornado induced missile may impact and affect the gas turbine storage tanks.

Investigate how a provisional air cooling/ventilation arrangement could be carried out to obtain as high cooling efficiency as possible for the EKB at Oskarshamn 1, assuming all ventilation and/or room cooling systems are inoperable.

At Oskarshamn 2 the steel plates and frame protecting of the air intakes to emergency diesel generators in sub division A and B should be of a more robust design to increase the robustness to withstand any tornadoes, tornado induced missiles and snow storms.

An investigation at Oskarshamn 3 can show how a provisional air cooling/ventilation arrangement could be carried out to obtain as high cooling efficiency as possible for the Control building, assuming all ventilation and/or room cooling systems are inoperable.

It is planned and there are many ongoing activities in Oskarshamn to increase the robustness of the plants to maintain safety during bad weather conditions. The measures are planned to modernise the power plant and to fulfil new requirements such as SSMFS 2008:17.

4.C. Ringhals

4.C.1 Design basis

4.C.1.1 Reassessment of weather conditions used as design basis

The present postulated severe weather conditions within the areas temperature, high air density, subcooled fog, saltstorm, ground frost, sea water temperature, wind, rain, hail, lightning and ice clogging has been evaluated. The present postulated limits of the respective parameters represent an occurrence frequency of 10^{-5} /year. The postulated limits, shown in table below, are based on new regulations, and the resulting backfitting work is in progress.

Condition	Classification	Loading	
		H2	H4
High outdoor temperature	Slow	35,4 °C	40 °C
Low outdoor temperature	Slow	-29,2 °C	-45 °C
High air density	Slow	Bounded by low outdoor temperature. The density is dependent on the temperature and pressure of the air. A low temperature results in a higher density of the air. The air pressure variations however has a marginal effect. Hence, this event is bounded by the low temperature event.	
Snowfall	Slow	1,5 kN/m²	3,0 kN/m²
Rime, subcooled fog	Slow	$T_{luft} < 2$ °C Relative humidity > 99 %	
Saltstorm	Slow	No loading specified	
Ground frost	Slow	No loading specified	
High seawater temperature	Slow	Occurence frequencies of seawater temperatures are not available	
Strong wind	Fast	25 m/s (1)	35 m/s (1)
Hurricane	Fast	--	72 m/s (2)
Hurricane missile	Fast	--	Missiles according to Regulatory guide 1.76
Rain	Fast	19 mm/10 min, 82 mm/ 24 h	37 mm/10 min, 145 mm/24 h
Hail	Fast	Bounded by snowfall	
Lightning	Fast	100 kA, 80 kA/µs	400 kA, 500 kA/µs, 50 MJ/Ω
Ice clogging	Fast	No loading specified	

1. The values are reference wind speeds as defined in "Boverkets handbok om Snö- och vindlast" (BSV 97).
2. The value is the actual maximum wind speed.

Postulated severe weather conditions at Ringhals.

The areas of severe weather conditions are judged to be complete, with the exception of the phenomena ice storm, which needs to be addressed.

The evaluation shows that the plant design can withstand the postulated severe conditions, with a few identified shortcomings, being addressed within the backfitting process to adapt to the new regulations.

The stressing of the weather parameters beyond the postulated levels has not revealed further shortcomings; rather it has emphasized the identified ones.

The assessment is made against the following postulated severe weather conditions

4.C.2 Evaluation of safety margins

4.C.2.1 Estimation of safety margin against extreme weather conditions

Outdoor temperature

Low/high outdoor temperatures are slow events, mainly affecting the ventilation systems, potentially causing environmental effects on exposed equipment. High temperature may e.g. cause accelerated ageing of components, while low temperature may cause ice clogging of air inlets to ventilation systems and diesel generators.

The assessment of plant performance with regard to low/high outdoor temperature according to table above shows that the main shortcoming is the guidance/instructions for appropriate measures in the case of severe outdoor temperature excursions. This is taken care of within the program to adapt to the new regulations SSMFS 2008:17. Since the low/high outdoor temperatures are slow events, there is time for reflection available, which is why the present situation is judged to be acceptable.

The stressing of the temperature issue is made by applying +40°C and -40°C for a period of several days. The difference between the stressing temperature level and the temperatures in the table above is that the temperatures in table represent one hour values, while the applied stressing level lasts for several days.

The stress test assessment is focused on the temperatures in rooms containing equipment related to residual heat removal, spent fuel cooling and power supply by the diesels. The temperature in the identified rooms has been evaluated by applying +40°C and -40°C as outdoor temperature. The diesel – and lube oil properties has also been evaluated.

The result of the stressed temperature level evaluation is that the ambient temperatures will not jeopardize the operation of equipment related to residual heat removal, spent fuel cooling and power supply by the diesels at extreme temperatures.

Snowfall

Snowfall is a slow event, affecting building integrity, ventilation systems, air intake to the diesels, grid availability and the general access to the site. The evaluation has been made against the H4 snowfall loading.

Regarding building integrity, the result of the evaluation is that some buildings needs plans for snow removal from their roofs, which is an ongoing work. These buildings are:

- Turbine buildings Ringhals 1 – Ringhals 4
- Reactor building Ringhals 1
- Fuel building Ringhals 2
- Intermediate building Ringhals 1
- Chamber for connections (Kopplingskammaren) Ringhals 1- Ringhals 2
- Diesel building Ringhals 1- Ringhals 2

- Service building Ringhals 3- Ringhals 4

Ventilation systems and diesel air intakes are generally installed above the expected snow depth.

Potential isolation of the site and loss of offsite power due to snowfall has the same consequences as isolation and loss of power for other reasons. This is discussed in section 5 of this report.

In summary, snowfall is a slow event with time for measures available, which is why the present situation is judged to be acceptable.

The stressing of the snowfall issue is made by assuming that no snow removal will take place. This will result in an assumed collapse of the roofs mentioned above. The maximum consequences in terms of fuel damage in case of a collapse are:

- Turbine buildings Ringhals 1 – Ringhals 4: No consequences
- Reactor building Ringhals 1: Damage to the fuel in the fuel pools
- Fuel building Ringhals 2: Damage to the fuel in the fuel pools
- Intermediate building Ringhals 1: No consequences
- Chamber for connections (Kopplingskammaren) Ringhals 1-Ringhals 2: No consequences

Collapse of the diesel building Ringhals 1-Ringhals 2 or the service building Ringhals 3-Ringhals 4 threaten to cause damage to the fuel in the core and in the fuel pools if the grid also is lost.

The stressed level snowfall margins have not been possible to decide, due to inexact knowledge of the properties of the roofs, and the snow distribution to be assumed. Therefore, it is appropriate to further address the roofs exposed to heavy loadings at large precipitation.

Rime, subcooled fog

Rime/subcooled fog is a slow event, potentially affecting air intakes by ice covering and thereby preventing their function.

The evaluation shows that a few ventilation air intakes are supplied with heating, thereby preventing ice coverage. For other buildings the consequences of interruption of the ventilation has been calculated, showing that the temperature increase is a slow process allowing time for consideration and manual measures. For buildings at the PWR units these calculations (and instructions developing) are an ongoing part of the program to adapt to the new regulations.

The air intakes for the diesels are designed to cope with the effects of rime/subcooled fog and associated ice coverage.

In summary, rime/subcooled fog is a slow event with time for reflection available enabling manual measures, which is why the present situation is judged to be acceptable.

Stressing of the rime/subcooled fog issue is not made, since more severe conditions than the postulated conditions cannot occur.

Salt storm

Salt storm is a slow event, consisting of a storm combined with salt coverage from the sea. The main issue of a salt storm is the impact on the switchyard equipment, potentially causing a loss of offsite power event.

The loss of offsite power is discussed in section 5 of this report. The plant is equipped with fresh water flushing systems to avoid salt storms affecting the operation.

The evaluation shows that the salt storm event does not jeopardize the safe operation of the plant.

Stressing of the salt storm issue is not made, since the consequences of a saltstorm are bounded by the strong wind issue and loss of offsite power.

Ground frost

Ground frost is a slow event, potentially causing buried piping to freeze. This would affect drinking water, raw water and sewage piping. However, this would take a very long time. Historically, this has not been an issue. The buried parts fire water system is equipped with heat tracing.

The evaluation shows that the ground frost event does not jeopardize the safe operation of the plant. Stressing of the ground frost issue is not made, since it is not that kind of phenomena. Either it is ground frost or not.

High seawater temperature

High seawater temperature is a slow event, affecting the cooling capacity of the plant cooling chain. Appropriate limits for the allowable seawater temperature are defined in the plant tech specs. At the defined temperatures measures are prescribed either to reduce the power or to close down the plant.

High seawater temperature will hence not affect the safe operation of the plant.

Stressing of high seawater temperature is not made, since harmful temperatures can not originate from weather conditions, only from blockage of canals which is discussed in section 3 of this report.

Strong wind

Strong wind is a fast event, potentially affecting the integrity of buildings by pressure differences or missiles. The applied windspeed of 35 m/s is a value with the occurrence frequency 10^{-5}/year, developed by SMHI. It is judged to be sufficiently conservative.

The evaluation shows that all buildings containing safety related equipment will withstand this wind.

Stressing of the strong wind is made by an assessment of the windspeeds correlating to the present design of the buildings. The result is that the H4 windspeed of 35 m/s could be increased to 40 m/s.

Hurricane and hurricane induced missiles

Hurricane is a fast event, potentially affecting the integrity of buildings by pressure differences or missiles.

The presence of hurricanes in Sweden is limited. The applied hurricane is therefore derived from RG 1.76, resulting in a hurricane windspeed of 72 m/s. The conservatism in this application is judged to be acceptable, based on information provided by SMHI.

The evaluation shows that two buildings will require measures to withstand hurricanes or hurricane missiles. The Ringhals 2 diesel building requires measures to withstand hurricanes or hurricane missiles. Ringhals 1 reactor building roof requires reinforcements to withstand hurricane missiles.

Measures at the diesel building are being taken. The consequences of a missile impact on the roof of Ringhals 1 are judged to be within acceptable limits. The situation is judged to be acceptable, regarding the 10^{-6}/year probability for the occurrence of the hurricane.

Stressing of the hurricane issue is made by converting the building wall design base load of 5 kN/m^2 to a windspeed. The resulting windspeed is 79 m/s, to be compared with the H4 hurricane windspeed of 72 m/s. This resulting windspeed will however affect some of the roofs beyond allowable limits. The affected roofs are the same as the ones affected by snowfall as described above.

Rain

Rain is a fast event, potentially affecting buildings by insufficient roof drainage. Rain may also cause flooding events. All roofs will withstand the 10 min rain loading according to table above. A few roofs may be overloaded during the H4 24 h rain loading if the rainfall exceeds the drainage capacity.

These are the same roofs as described in the snowfall section above. The drainage capacities of these roofs need to be checked in case of severe rainfalls. Further, the overflow drainage capacity needs to be evaluated.

Flooding of the yards caused by severe raining is prevented by sloping of the yards. The drainage provisions have large capacities.

In summary, a few deficiencies regarding the handling of severe rainfall have been identified. None of these is considered as a safety deficiency, but rather a potential for increased robustness of the plant.

Stressing of the rain issue is made by assuming an even heavier rainfall. The outcome is the same as for the severe rain description above. Another possibility is a flooding event caused by rain, which however is bounded by the seawater level generated flooding discussed in section 3 of this report.

Hail

Hail is a fast event. The loading generated by the hail is bounded by the loadings generated by the snowfall case.

Stressing of the hail issue is made by assuming a large size of the hailstones, causing missile effects on outdoor equipment. Heavier hailfall is bounded by the heavier snowfall as discussed above.

Missile effects of hailstones may cause the switchyard equipment to fail, causing a loss of offsite power event. This event is discussed in section 5 of this report. Building damages due to large hailstones are not expected.

Lightning

Lightning is a fast event. The plant may be affected thermally, mechanically and electrically when struck by lightning.

The plant has been evaluated against the lightning properties described in table above. The result is that Ringhals 1 will withstand both the H2 and H4 size of lightning. Ringhals 2-4 will withstand the H2 size of lightning. This has earlier been identified within the process to adapt to the new regulations, and is an ongoing work.

Stressing of the lightning issue is made by applying larger energy content than specified in above. The resulting temperature increase in the lightning rod is calculated. The result is that for the increased energy contents 60 and 70 MJ/Ω, the temperature rise is well below the melting temperature of the aluminium in the lightning rod (181°C vs.660°C).

Frazil ice formation

Frazil ice formation in the intake canal is a fast event. It is caused by low seawater temperatures creating ice crystals accumulating on equipment below the surface. This is prevented by a recirculation flow in the cooling water canals.

The evaluations show that frazil ice formation does not affect the safe operation of the plant.

Stressing of the frazil ice formation issue is not made, since the ultimate consequence is blockage of the intake canal, which is discussed in section 5 of this report.

4.C.2.2 Measures which can be envisaged to increase robustness of the plants against extreme weather conditions

The identified main areas for improvement are

- Administrative measures (instructions/guidelines) to cope with low/high temperature conditions, snow removal and severe rainfall at identified roofs.
- Measures to cope with the H4 lightning for Ringhals 2-4.
- Further addressing of the properties of certain roofs at heavy precipitation.

4.2 Assessments and conclusions

4.2.1. Licencees assessment and conclusions

The assessments according to design requirements are carried out completely by the licensee holders. Swedish regulation says that nuclear reactors must be dimensioned to withstand natural phenomena and other events that occur outside or inside the facility that can lead to a radiological accident.

All Swedish NPPs are required to analyze how bad weather can affect the plant and the Swedish regulations SSMFS 2008:17 §14 have more stringent requirements to analyses for severe weather conditions that the licence holders are adapting the plants to, where they have found shortfalls. The extreme weather conditions are based on statistics from the last 100 years as a maximum. An estimation of more improbable extreme weather events is done by the licensee holders assisted by SMHI (Swedish Meteorological and Hydrological Institute).

Extreme air temperatures are considered as a slow event where there is time to take manual measures.

At low air temperature the concerns for the nuclear power plants are that some essential process measurement piping could run the risk of freezing and thereby indicate inaccurate values to the reactor protection system. In case of low temperature alarm in the ventilation system, manual measures shall be performed in accordance with the procedure to ensure that freezing of concerned components are avoided. In this case the measure is to manually stop the inflow air system and at some building to increase the surveillance.

In case of a prolonged high outside air temperature exceeding dimensional values the ambient temperature may be assumed to exceed the allowable ambient temperatures in the same range as the air temperature are exceeded. In case of an assumed prolonged outside air temperature of approximately +40°C it is therefore reasonable to assume that the ambient air temperature in the rooms does not exceeds allowable limits. The engineering judgment is that this only may cause accelerated ageing of concerned equipment and not instant malfunctioning.

Extreme rainfall and snowfall are considered as slow events were there is time to take manual steps if the ordinary equipment doesn't operate as designed.

The main identified risk connected to rainfall is the load of roofs because many roofs are flat or slightly inclined with surrounding ledges. If no drain is drawn off by the drainpipes etc. the load can be too heavy. Some buildings with a safety task are not constructed for a load of water to the top of the ledges and some neither have a diversified drain system.

Leakage of a roof is not considered to be a significant risk since there is good separation between redundant safety trains.

Another identified risk is if rainwater from the buildings and from the ground is leaking into the rock shaft. The drainage pumps in the rock shaft have a good margin in capacity to avoid flooding of the rock shaft. For more description of flooding, see section 3.

Assessment by some licensee holder is that proactive actions must be made within 12 – 24 hours for some buildings in case the drainage pipes are clogged during a heavy rainfall.

At intensive snowfall the roofs can be overloaded with snow and in case of a collapse for some buildings, there is a risk that the collapsed roof can destroy safety systems or fuel in the basin for storage of spent fuel.

A majority of the buildings are dimensioned to receive a very intensive snowfall during 1-2 days before the snow must be removed by shuffle etc. All licensee holders claim that manual measures such as shuffling of snow must be done in case of a very intensive snowfall.

The event snow storm is handled differently by the licensee holders. Forsmark and Oskarshamn describes that there will be problem to perform any work outside during a snowstorm. Snow is expected to reach the top of the ledges at all buildings without any treat to the building. But there are locations were snowdrifts jeopardize to become higher and too heavy for the building and in some cases a risk for the safety systems. For newer NPPs there is expected to be redundant safety trains, in other wind directions, that can shut down the plant and maintain residual heat removal. Ringhals has not reported snow storm separately but consequences of heavy snowfall without snow removal.

All licensee holders have reported that the sites are constructed for strong wind. At much stronger winds there are some shortfalls reported for elder plants such as risks that beams and parts of the walls can fall down and some roofs may be affected.

For tornado missiles all licensee holders have used the missiles described in the NRC Regulatory Guide 1.76. There are some buildings were shortfalls have been reported for the described missiles. The consequences are though limited such as damage of a limited number of fuel elements, damage of one safety train or damage of diesel tanks. The consequences and potential emissions are expected to be limited and are equivalent with other events in the SAR. The effects of tornado missiles are considered as a fast event.

The assessment of strong wind, hurricane and tornado induced missiles show that the licensee holders have analyzed the strong wind and they have found some shortfalls. The shortfalls are validated reasonable by SSM since it concerns very improbable weather conditions according to reported statistics.

High seawater temperature is considered as a slow event where all licensee holders have procedures to reduce the reactor power and to shut down the reactor at sea water temperatures about +25°C. After shut down there are not expected to be any shortfalls for the reactor.

Frazil ice formation in the intake is a fast event. It is caused by low seawater temperatures creating ice crystals accumulating on equipment below the surface. This is prevented by a recirculation flow in the cooling water canals. The evaluations show that frazil ice formation is not expected because of the recirculation flow. Stressing of the frazil ice formation issue, results in worst case, in the ultimate consequence blockage of the intake, which is discussed in section 5 of this report.

Assessment of high and low seawater temperature is not a weather condition were there are obvious risks for the safety of the nuclear power plant since there are prepared procedures and limitations for operation. The exception is extreme frazil ice formation or ice clogging which is very improbable.

Lightning is a fast event. The plant may be affected thermally, mechanically and electrically when struck by lightning. All licence holders have made analysis for Boiling Water Reactors of the consequences of a severe lightning and the consequences are limited. Ringhals 2-4 has earlier identified that they need measures to handle a H4 lightning.

4.2.2. Licensees recommendations for potential improvements

The licensees have identified the following recommendations for further evaluations and reassessments.

- Complete procedures for the deficiencies found during the "stress test", in particular, to draw up a procedure for external impact on the Forsmark 3 unit.
- Review and completion of operational and maintenance procedures to prohibit unallowable accumulation of water and snow on some identified roofs.
- To secure long time operation of safety diesel generators at Oskarshamn 1 and 2, a review and completion of procedures shall be performed to ensure that a sufficient storage of diesel oil is stored in the large oil storage tanks. A tornado induced missile may impact and affect the gas turbine storage tanks.
- Investigate how a provisional air cooling/ventilation arrangement could be carried out to obtain as high cooling efficiency as possible for the EKB at Oskarshamn 1, assuming all ventilation and/or room cooling systems are inoperable.
- At Oskarshamn 2 the steel plates and frame protecting of the air intakes to emergency diesel generators in sub division A and B should be of a more robust design to increase the robustness to withstand any tornadoes, tornado induced missiles and snow storms.
- An investigation at Oskarshamn 3 can show how a provisional air cooling/ventilation arrangement could be carried out to obtain as high cooling efficiency as possible for the Control building, assuming all ventilation and/or room cooling systems are inoperable.
- Administrative measures (instructions/guidelines) to cope with low/high temperature conditions, snow removal and severe rainfall at identified roofs.
- Measures to cope with the H4 lightning for Ringhals 2-4.
- Further addressing of the properties of certain roofs at heavy precipitation.

4.2.3. SSM's assessment and conclusion

SSM is looking into which requirements, according to SSMFS 2008:17, those are appropriate for severe weather and there may be further assessments of SSM in the future. The assessment in this report is evaluated of severe weather for the Post-Fukushima Stress tests of European Nuclear Power Plants.

SSMs assessment of the reported extreme weather conditions is that it covers expected weather conditions which are reported in the safety analysis reports. There are some conditions related to extreme weather that is not reported or only reported by one licensee holder such as, extreme sea waves, extreme seaweed growth and extreme water level. The consequence of these conditions is handled in section 3 flooding and in section 5 loss of ultimate heat sink. Some reported weather conditions are not presented in the conclusion since it is only reported by one licensee holder and there is not discovered any serious or different consequence. Ice storm is a weather condition where SSM consider that new knowledge leads to new requirements of analysis.

The licensee holders have not identified any shortfalls that they consider to address for extreme air temperature. An assessment of SSM is that the analysed temperatures are reasonable because there are basically long-term air temperatures that affect a nuclear site. Licensee holders should establish levels for extreme outdoor air temperatures in the Safety Analysis Report (SAR).

Extreme rain and snowfall is a slow event, according to design basis, where there is time to take proactive actions in 12-24 hours. The reported levels of extreme precipitation are reasonable according to reported weather statistics. SSM expects that strength reinforcements are performed to meet new requirements where it is demanded.

For high and low seawater temperatures which is considered as a slow event there are determined procedures how the operation will shut down at high seawater temperature and start recirculation at cold seawater temperature. However, frazil ice formation is a fast event where there is a risk for clogging of the intake grating. The event is described in SAR and the consequences are described in section 5 for stop in the intake. Therefore SSMs assessment is that there is no need for determination of more extreme seawater temperatures.

To evaluate tornados and tornado induced missiles the licensee holders' applies US NRC Regulatory Guide 1.76 Rev 1. SSM finds the assessment of strong wind and tornados reasonable although there are buildings that the licence holders' reports require reinforcements. SSM expects that strength reinforcements are performed to meet new requirements.

For lightning the assumed level are different between the plants. SSMs assessment is that for the stress tests the set values are sufficient and investigated extreme lightning from the licensee holders show no shortfalls.

Additional, the following items have been identified by SSM in addition to the measures identified by the licensees during the licensee assessments. The licensee holders must undertake more analysis and consider potential measures to increase robustness of the plants for the following aspects:

- None of the licensee holders have dealt with how severe weather will affect the access to the plant for staff, heavy equipment and supplies as an initiating event. In section 6, bad weather is described as a part of severe accident management. There is not described any procedures to prevent shortcomings due to blockage of access roads connected to extreme weather.
- The condition and plausible consequences of an ice storm is not described by the licensee holders.
- Oskarshamn reports that weather forecasts are communicated within the organization. The other licensee holders should report how weather forecasts are handled.

According to the conditions in ENSREG the licensee holders are requested to search for cliff edge effects of different weather conditions. Of course it is impossible to analyze the consequences for the nuclear power plants of all cases of extreme weather. SSM considers there are few cliff edge effects reported outside original classified events for extreme weather.

5 Loss of electrical power and loss of ultimate heat sink

In this chapter an introduction to the Swedish "stress test" assessments regarding loss of electrical power and loss of ultimate heat sink, will be given in section 5.1. In the site specific sections following the introduction, section 5.A, 5.B respective 5.C, a summary of the licensee assessments of the sites' and the units' protection against loss of electrical power and loss of ultimate heat sink, will be presented. The site specific sections include results and identified measures which have been foreseen by the licensees to increase robustness of the sites and units respectively. The last section in this chapter, section 5.2, contains the Swedish Radiation Safety Authority's overall assessments and conclusions of the Swedish sites' and units' protection against loss of electrical power and loss of ultimate heat sink.

5.1 Introduction

This chapter is intended to cover a broader spectrum of initiating events or indirect initiating events (in addition to earthquake and flooding) that may result in loss of electrical power and/or loss of ultimate heat sink for the Swedish NPPs. Considered events are for instance bad weather conditions and situation provoked by indirect initiating events, such as large disturbance from the electrical power grid impacting AC power distribution systems or forest fire, airplane crash, etc.

Transport difficulties to the sites are considered in these assessments by assuming that the sites are isolated from delivery of heavy material for 72 hours by road, rail or waterways. Portable light equipment can arrive to the sites from other locations after the first 24 hours. Transportation difficulties within the site should also been considered.

According to Swedish Radiation Safety Authority regulation SSMFS 2008:17, all Swedish NPPs are required to withstand natural phenomena and other events that arise outside or inside the facility and which can lead to a nuclear accident. Natural phenomena and other events to be considered are for example, extreme winds, extreme precipitation, extreme icing, extreme temperature, extreme sea waves, extreme seaweed growth or other biological conditions that can affect the cooling water intake, extreme water level, earthquake, etc. All of these phenomena and events could potentially lead to loss of electrical power and loss of ultimate heats sink.

5.1.1. Loss of electrical power

The electric power systems for all Swedish units are designed with two power inputs from the transmission system and independent back-up power generators that will ensure power to the plant even if power from the external grid should fail. In addition, the Swedish NPP units, with exception for Oskarshamn 2, have alternate means via automatic or manual actions, to provide the units with electrical power from gas turbines (GT) on-site or in the vicinity of the site.

The purpose of the electrical power systems is to provide the units with the required electrical power under normal operating conditions and to support the plant safety and protection features so that they can perform their respective functions as required in abnormal and accident conditions. The electrical power systems should be design so it will be maintained even when power from the external grid is not available.

Basic safety requirements relating to the plant's power supply are formulated in the Swedish Radiation Safety Authority regulations SSMFS 2008:1 and SSMFS 2008:17. When designing and building the electrical power system, the following general conditions have been followed; the safety systems including the power systems are divided in separate and redundant divisions to be able to secure the safety functions in a reliable manner. The divisions of generators and batteries used as power supply sources, should therefore be physically and functionally separated and as far as reasonable achievable independent of one other.

House load operation

In case of disturbance on the off-site external power grid it should be possible to supply the unit power supply systems from the main generators, i.e. house load operation. At house load operation electric power is supplied to the plant exactly like during normal operation. In this mode of operation, on-site power back-up sources are not required from a power supply perspective. No diesel or battery capacity needs to be utilized. House load operation provides an extension of time until on-site power sources are needed (i.e. emergency diesel generators, gas turbines and batteries). If loss of off-site power occurs, units are automatically disconnected from external grid and reactor power is reduced to a level required for house load operation. The steam flow will quickly be reduced to the turbine plant so that the turbine or generator is not tripped due to exceeded limit values in any protective function. The excess steam is dumped into the condenser. Also, the non-safety power supply system remain energised, i.e. the same situation as for normal operation of the unit.

A minimum of 12 hours house load operation is a non-safety requirement given by the Nordic grid codes. A verification of the plant's capability to be able to transfer to house load operation from full load is required by the Swedish Radiation Safety Authority and will be performed.

Significant events in Forsmark (Sweden) and Olkiluoto (Finland)

Loss of offsite power always leads into a dynamic response, with the risk of creating transients in the electric power system. An unexpected high transient caused a loss of power in the Forsmark 1, in 2006. Similarly a high transient on the main generator of Olkiluoto 1 was experienced year 2008. The main cause to these events was the design of the electric power system. The Swedish NPPs have used the result from the analysis of those events to improve the safety of the on-site power systems in the plants. Other types of large disturbance from the electric grid are covered by the three postulates of the stress test.

5.1.2. Loss of ultimate heat sink

All Swedish NPPs are located on the coastline of Sweden and the primary ultimate heat sink for all units is sea water. The Forsmark and Oskarshamn sites are situated on the east coast of Sweden and the Baltic Sea forms the primary ultimate heat sink for the plants. The Ringhals site is situated on the west coast of Sweden where the sea area Kattegat forms the primary ultimate heat sink for the plant.

5.1.3. Severe accident mitigation systems

In the analyses of loss of electrical power and loss of ultimate heat sink and the severe accident mitigation systems are discussed. All Swedish units are provided with equipment for filtered containment pressure release, the Multiple Venturi Scrubber system (MVSS). The primary MVSS functions are passive. However, battery back-up power supplies are required for providing process information to plant operators. The battery backed-up power supplies system for consequence-mitigating systems has been designed to maintain operation during the first initiating hours in case of a severe accident. There are two options for connection of mobile integrated drive generators in each plant with the aim of supplying the consequence mitigating systems after the battery backed-up power supply has been discharged.

For the BWR designs the assessments provided by licensees have identified that the MVSS could be used in emergency situations for residual heat removal from the reactor core. If loss of core cooling occurs in a BWR design the temperature of the water inventory and the pressure in the reactor pressure vessels will increase and at a certain pressure level the safety relief valves will automatically open and transfer steam to the suppression pools. The temperature in the suppression pools will increase and eventually the water inventory in the suppression pool will start to boil. The only available alternative at this point is to open the valves to the MVSS. If valves to the MVSS are not open manually by operators, valves will open automatically by the rupture disc when the containment pressure has increased to the rupture disc limit (which is set to a level that will give a reasonable margin to the containment design pressure). The released steam will be condensed in the MVSS water pool and the temperature in the MVSS will increase until it starts to boil. Steam is then evaporated through the stack of the MVSS to the atmosphere.

Water inventory in the suppression pool could be maintained by using demineralised water for make-up water according to the normal procedures. In an emergency situation, water could also be added from the independent containment spray system. This means that in conditions when all containment cooling is lost, fire trucks will be able to provide containment spray through prepared arrangements in the ordinary containment spray system.

5.A. Forsmark

5.A.1 Loss of electrical power

The power supply systems

The Forsmark site is connected to the external 400 kV grid and the external 70 kV grid.

During normal operation all units at the site are connected to the 400 kV grid. Forsmark 1 and Forsmark 2 have a common transmission system interface and Forsmark 3 has a separate transmission system interface (however, if needed, Forsmark 3 can be connected to the 400 kV grid via the other units' interface.

The 70 kV grid has two separate incoming lines and is also connected to gas turbine (GT). The 70 kV grid has one common line to Forsmark 1 and Forsmark 2, and one connection to Forsmark 3.

The GT located in the vicinity of the site are included in the SBO case. The system itself is classified as non-safety system but will be initiated via the Forsmark automatic safe supply system if at least two of the main auxiliary power supplies in any of the units are unavailable.

The main auxiliary power supplies for Forsmark 1 and Forsmark 2 are 6 kV and 10kV for Forsmark 3. Connections are in place between the non-safety auxiliary power supply system, the stations transformers and main generators. All safety classed electrical systems are normally supplied via the non-safety auxiliary power supply system. The non-safety auxiliary power system is supplied via the generators during normal operation and house load operation if the off-site power supply system is unavailable. If the main generator is unavailable the non-safety auxiliary power system is supplied via the external grid through the main transformer.

12 hours of house load operation is required, but experience shows that only about 50 % of the transitions to house-load operation with stable operations for more than 2 hours, are successful.

Failed house load operation will cause under voltage in the units non-safety and safety classed power supply systems, which results in an emergency diesel generators (EDG) initiation that will cause the EDGs to start and run idle. If the voltage from the 70 kV is available, the supply from the 70 kV grid through to the safety classed power supply system is engaged via automatic changeover. Objects start via a starting sequence. The EDG will in this case remain idling.

The ordinary back-up AC power supply system

If 70 kV supply is not available, the EDGs energise the safety classed AC power supply system.

Each unit has four physically and electrically separated EDGs which all have sea water as their ultimate heat sink. Each EDG are design to provide a 50% power supply capacity for the safety systems and components. However, some important systems can only be supplied from two out of the four EDGs at each unit.

The alternate back-up AC power supply

In case of a disturbance in the off-site power grid an automatic function connects the power supply systems for each units to the GT and initiates start of the GT. The GT can also be started manually locally or from the main control room at Forsmark 1.

The GT does not have the capacity to supply all three units with simultaneous full start sequences. In the start sequence for GT, equipment directly connected to the normal power supply system is therefore blocked. The start sequence for the EDG backed-up power supply system is not affected.

The battery backed-up AC and DC power supply systems

The batter back-up power supply systems are divided into redundant physically separated divisions.

Batteries are designed to maintain 2 hours of operation without charging.

The power supply system for the severe accident mitigation systems

The accident mitigation systems have two redundant divisions of battery back-up power supplies and are designed to maintain 24 hours of operation without charging.

At the Forsmark site arrangements are in place for connection of small external mobile generators to the accident mitigation system if the ordinary power supply fails. There are three small external mobile generators available on site.

5.A.1.1 Loss of off-site power

Loss of off-site power is in the design basis for all the units and presented in the safety analysis report. No consequences for the fuel in the reactor core or in the spent fuel pools have been identified as long as the ordinary backed-up AC power supply systems are functioning.

The EDG fuel storages

The diesel fuel supply for the EDGs consists of day tanks and an additionally outdoor storage tank.

Forsmark 1 and Forsmark 2 have a common outdoor storage tank and Forsmark 3 has a separate outdoor storage tank. The separate outdoor storage tanks together with the separate indoor tanks are sufficient for 7 days of operation for four EDGs. For Forsmark 1 and

Forsmark 2, the total diesel fuel storage is sufficient for approximately 4 days of operations if all eight diesels are in operation at the same time.

However, for all units only two out of four EDGs are required to operate during accident conditions according to the safety analysis report. Estimations shows however, that for most scenarios, only one out of four EDGs would be needed to reach a cold shut down state for the reactors and maintain a safe state for the unit in long time term conditions. Furthermore, the power demand for the units is also expected to decrease over time. Hence, the fuel supply for all units is expected to be enough for much longer time than 7 days.

The lubricating oil storage

The lubricating oil top ups for Forsmark 1 and Forsmark 2 is sufficient for more than 7 days of operation and for Forsmark 3 it is expected to be sufficient for 7 days.

Analysis

As long as the EDGs are available there are no expected consequences for safety of the plants (including the spent fuel pools). For this situation fuel damage is not expected to occur, neither in the reactor cores nor in the spent fuel pools.

Weather conditions that can adversely affect functions and manual actions have been considered in the analysis. However, no such effects have been identified in this analysis.

5.A.1.1.1 Results from the licensee assessments of protection against loss of off-site power

The units are designed for loss of off-site power and fulfil requirements within the design basis.

Electrical power to safety related functions can be provided by house load operation and by the EDGs at loss of off-site power.

The on-site power source can be maintained for 7 days even under extreme weather conditions as stated in the safety analysis report.

5.A.1.1.2 Measures which can be envisaged to increase robustness of the plants in case of loss of loss of off-site power

There are some on-going projects at the Forsmark NPP that are expected to increase the availability of the power supply for the Forsmark units. For example, there is one project to replace and upgrade the 70 kV transmission system interface in 2013, including upgrading the battery backed-up power supply for the 70 kV switchgear auxiliaries. The goal is to secure a battery back-up power supply for at least 15 hours of operating without charging. Also, it should be possible to charge the battery backed-up power supply from an external power supply source in long-term conditions (>12 days).

Aside from the on-going projects the following potential measures have been identified during the analysis and should be evaluated:

- Ensure that requirements from IEEE 765 are met with respect to separation between the different external supply routes.
- Supplementary arrangements with connections for mobile equipment to power the off-site power supplies auxiliary as well as the GT auxiliary should be implemented. The mobile units should also be included in the emergency preparedness instructions and facilities.

- Implementations of instructions for reconstruction of the power supply systems after battery back-up power supplies for off-site switchgear and GT auxiliary have been discharged (including the fact that ordinary control and relay protection functions might have to be restored).
- Consider the possibilities and if necessary procure contracts with the grid system owner, for direct connections to nearby power production facilities, e.g. hydro plants. The aim should be to obtain priority for supplying the Forsmark NPP in the event of extreme situations.
- All EDGs are dependent of the sea-water cooling and at loss of sea-water cooling systems all EDGs are expected to quickly fail. By considering the possibility to rebuild the cooling system for some of the existing EDGs or complement the existing EDGs with additional EDGs that are air- or groundwater cooled, the likelihood of severe consequences for the safety of the plants in case of loss of off-site power could be limited.

5.A.1.2 Loss of off-site power and loss of the ordinary back-up AC power source

At loss of off-site power and loss of on-site back-up power sources, power from the external grids is lost and the EDGs are assumed to fail. The GT will however, still be available.

<u>The GT fuel storage</u>

The GT has fuel for 4 days at full power operation. The lubricating oil system is a closed system where the lubricating oil must be changed within 7 days of operation time.

<u>Analysis</u>

At loss of off-site power and loss of the ordinary back-up AC power source a reactor trip will automatically be initiated.

The normal start up time for the GT is approximately 1.5 minutes. The time available for the GT to start before core damage is unavoidable is approximately 30 minutes.

The powering of safety systems from GT is automatic and as long as the GT is in operation and able to supply the units there will be no consequences for the safety of the plants (including the spent fuel pools) and there will be no fuel damage occurring in either the reactor cores or in the spent fuel pools.

5.A.1.2.1 Results from the licensee assessments of protection against loss of off-site power and loss of the ordinary back-up ac power source

At a loss of off-site power and loss of ordinary back-up AC power, all units will be supplied by the GT.

The GT can provide power for at least 4 days without external support.

The GT capacity is enough to supply and maintain operation of the safety systems at all units simultaneous, including core cooling and cooling of spent fuel pools.

5.A.1.2.2 Measures which can be envisaged to increase robustness of the plants in case of loss of off-site power and loss of the ordinary back-up ac power source

There are a number of on-going activities to update and improve existing limitations in relation to station block out, for example, the following activities are being reviewed:

- Zero voltage protection with automatic reconnection.

- Blocking of start sequence in connection with weak systems as well as with simultaneous restart of several reactors.
- Improve independence in relation to the 400 kV power supply.
- Measures with the aim of increasing availability for GT (renewal, modernisation, testing, etc.).
- Reduce risk of common cause failures (CCF) in diesel system (diesels currently have the same heat sink/sea water, fuel and protection).

5.A.1.3 Loss of off-site power and loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power sources

At loss of off-site power, loss of the ordinary back-up AC power sources, and loss of permanently installed alternate back-up AC power, only battery back-up power systems will be available.

The battery backed-up power supply

The battery backed-up AC and DC power supplies have been designed to maintain 2 hours of operation. However, estimations by considering load profiles during normal operation and abnormal conditions and the discharge profiles indicates that the battery backed-up power supplies could be maintained for more than 12 hours. Assumptions are based on 30 years of operation and conservative design principles. Also, replenishing of the batteries after a long term discharge is not a problem.

Analysis

The battery backed-up power supply systems' powers all the safety classed instrumentation and control functions as well as some valve actuations.

At loss of all EDG backed-up AC-power no core cooling functions are available. For all units a reactor trip will be automatically initiated as a result of the initial event.

For Forsmark 1 and Forsmark 2 and by crediting the MVSS capability to remove residual heat, damage to the reactor fuel is expected to be unavoidable after approximately 35 minutes. For Forsmark 3, damage to the fuel will be unavoidable after approximately 1 hour.

The spent fuel pools

Worst case scenario in terms of time to significant fuel damage in the reactor core, occurs during power operation, e.g. during outage when fuel is present in either in the reactor pressure vessel and/or in the spent fuel pools the time to fuel damage is much longer.

The fire-water system could be used for manual cooling of the spent fuel pools. The fire-water system has two separate diesel driven fire pumps and is assumed to be available during these conditions. By manually connect hoses to the fire-water system, make-up water could be maintained for the spent fuel pools at each unit. If the fire-water system is not available, hoses could be connected to fire trucks. However, there is only one fire truck available at the site which means that only one unit at the time could be supplied by the fire truck before external fire trucks from the neighbouring area would be able to reach the site.

If no manual actions to cool the spent fuel pools have been successfully completed damage to fuel will be unavoidable after about 1 day for the most limiting case (when all the fuel is located in the fuel storage pool during refueling).

5.A.1.3.1 Results from the licensee assessments of protection against loss of off-site power, loss of the ordinary back-up ac power sources, and loss of permanently installed diverse back-up ac power sources

Battery capacities are designed for 2 hours of operation without charging.

Worst case scenario in terms of time before significant fuel damage in the reactor core is unavailable, occurs during power operation.

For Forsmark 1 and Forsmark 2, damage to the fuel is calculated be unavoidable after approximately 35 minutes. For Forsmark 3, damage to the fuel is calculated to be unavoidable after approximately at 1 hour.

Activity releases to the environment is limited by the MVSS.

5.A.1.3.2 Measures which can be envisaged to increase robustness of the plants in case of loss of off-site power, loss of the ordinary back-up ac power sources, and loss of permanently installed diverse back-up ac power sources

If the off-site grid in the vicinity of the site could be repaired within a few days, it might be possible to introduce power to the site from a nearby power station by implementing temporary arrangements. As for today, personnel from the electricity supplier are required to repair the grid as soon as possible. It should also be possible to use the NPP personnel and resources for temporary installation of necessary cables to enable these arrangements.

Increased robustness could be achieved by extending the capacity of the battery backed-up power supplies to certain important loads during long term operation, e g. by disconnections of less important loads.

- Another mean to increase robustness and safety could be to:
- Introduce new independent system for emergency cooling (diversified in relation to existing emergency cooling systems).
- Implement manageable connections for the high pressure safety injection systems and the containment spray system to external alternate emergency power supply system.
- Introduce automatic diversified power supply systems for the safety systems (e.g. diversified operation/station blackout diesel units or gas turbine on site with prepared fixed station blackout installation).
- Diversified residual heat removal, without using the sea
- Introduce instructions for energising after protracted loss of electric power

Also, the battery backed-up power supplies are limited to 2 hours of operation. Extending the capacity of the battery backed-up power supplies to certain important loads during long term operation, e g. by disconnections of less important loads, would increase the robustness of the units.

5.A.2 Loss of the ultimate heat sink

The Forsmark site is situated on the coastline to the Baltic Sea and the ultimate heat sink for all units at the site is sea water from the Baltic Sea.

There is no alternate ultimate heat sink available for the Forsmark units.

5.A.2.1 Design provisions to prevent the loss of the primary ultimate heat sink

The sea-water cooling systems

The surface water cooling water inlet channel is common for all units. It is channelled through natural pools and excavated channels, to each intake. The main channel section has a foam barrier to prevent any floating objects from entering further.

Forsmark 1 and Forsmark 2 have a common cooling water intake and Forsmark 3 has a separate cooling water intake. Each intake contains machine-cleaned coarse gratings and basket strainers to remove impurities, plus hatches to close and empty the inner waterways.

The inlets for all units have recirculation capabilities. This means that during cold winter days heated cooling water could be recirculated to the intake, to prevent ice to form and potentially affect the cooling water system.

From the intake of Forsmark 1 and Forsmark 2, the cooling water system separates into inflow culverts (two per unit). Each inflow culverts partly provides water for both the condenser cooling pumps and the auxiliary cooling pumps.

From the intake of Forsmark 3, the cooling water is carried in a cast concrete coated rock tunnel to associated systems. The auxiliary cooling water branches off in the main cooling water pump sump, and goes into a separate culvert to the auxiliary cooling water.

The cooling water outlet from Forsmark 1 and Forsmark 2 is common. The cooling water outlet for Forsmark 1 and Forsmark 2 consists of a tunnel that has a large overflow pool designed to reduce the effects of water surges.

The cooling water outlet for Forsmark 3 is separate from the other units cooling water outlet and consists of waterways up to an overflow shaft and a gate chamber. From the overflow shaft and the gate chamber, the cooling water flows into the outlet tunnel. In the outlet tunnel there is a large overflow pool to reduce the effects of water surges. The outlet tunnel is also equipped with an auxiliary discharge in case of any extreme events that cause a blockage (due to e.g. pack ice) in the main outlet to the sea. If the intake for Forsmark 3 is blocked, the redundant auxiliary cooling water intake in the outlet tunnel is automatically opened, and cooling water from the outlet tunnel flows into the cooling water pumps. At the same time, the auxiliary cooling water discharge hatches will close at the overflow shaft/gate chamber. This forces the auxiliary cooling water to discharge via the spillway and prevents circulation of increasingly warmer auxiliary cooling water.

5.A.2.1.1 The spent fuel pools

Forsmark 1, 2 and 3 have two spent fuel pools each with similar designs. The reactor pool and the pools for internal parts are located in between the two spent fuel pools. All the pools are connected by a manoeuvrable gate. The gate will automatically close if the water level decreases to about 1.5 meters respectively 2 meters below normal level in the pools for Forsmark 1 and Forsmark 2 respectively Forsmark 3. If this happens there will still be more than 7 meters respectively 5 meters of water above the fuel assemblies Forsmark 1 and Forsmark 2 respectively Forsmark 3. One of the spent fuel pools are also connected by a manoeuvrable gate to the fuel transport cask pool, which is a specific pool used for loading of the transport cask for spent fuel.

The ultimate heat sink for all cooling trains connected to the spent fuel pool is sea water. Normal cooling of the fuel pools is performed by the spent fuel pool cooling system which has two redundant trains (one in standby during normal operation). The levels in the pools are maintained by the demineralised water system or the liquid radwaste system.

Reactor pools are temporary emptied during refueling outages in order to remove the containment head and the reactor pressure vessel head. After the heads are removed the reactor pool is refilled. There are different cooling systems used to for the residual heat removal during a refueling outage. In the beginning of the outage the residual heat removal system is used. When all fuel assemblies have been moved from the reactor pressure vessel to the spent fuel pool the residual heat removal system can be shut off and the spent fuel pool cooling system will be able to provide sufficient cooling for residual heat removal.

The spent fuel pools levels and temperatures are monitored in the main control room and will initiate an alarm at low level or high temperature. The design temperature for the spent fuel pools is 60°C.

5.A.2.1.2 Design provisions to mitigate loss of primary ultimate heat sink

The fresh-water supply

The fresh-water supply system has storage of 19000 m^3 of water available for all units.

The demineralised water system

At Forsmark 1 and Forsmark 2 the demineralising water system can be passively supplied by gravity from the fresh-water system. At Forsmark 3 this capability does not exist.

At Forsmark 1 and Forsmark 2 the high pressure safety injection system is supplied by water from a separate demineralised water storage tank. For Forsmark 3 the high pressure safety injection system is normally connected to the suppression pool, but in emergency situations the high pressure safety injection system could also be supplied by a separate non-safety demineralised water storage tank.

At Forsmark 1 and Forsmark 2, the refilling of the demineralised water storage tanks is normally provided by the system for demineralised water. The separate non-safety demineralised water storage tank at Forsmark 3 does not have the same capability available.

If refilling of the demineralised water system tank by the demineralised water system is unavailable (as for Forsmark 3) or insufficient, the demineralised water storage tank may also be manually refilled by the fire-water system. This can be achieved by connecting hoses to the vent pipe on the demineralised water storage tank or by connecting hoses directly to the demineralised water system.

The capacity of the system for demineralised water at Forsmark 1 and Forsmark 2 should be able to deliver sufficient flow to the demineralised water storage tanks in each unit to avoid core damage, if the right flow is obtained. Operational procedure for refilling exists and environmental effects such as severe weather conditions are not expected to prevent/delay these manual actions. However, to establishing the specific required flow, manual actions have to be performed and the availability of both operating procedures and personnel for this specific task is questionable.

Since the flow from the system for demineralised water is small, it may be necessary to monitor the level of the demineralised water storage tank and to stop the flow in high pressure safety injection system low levels to avoid air ingestion in the high pressure safety injection pumps. There is an operating procedure in place that states that high pressure safety injection system should be put in recirculation mode when the level of the demineralised water storage tank reaches low level.

The fire-water system

The fire-water system has two independent diesel driven fire-water pumps.

The fire-water storage tank contains 1500 m^3 at Forsmark 1 and Forsmark 2 and 3600 m^3 at Forsmark 3.

At all units the fire-water system can passively be supplied by gravity from the fresh-water system. However, the connection between the fresh-water reservoir and the fire-water system must be manually obtained. There are operational procedures in place describing this and operators to perform this task should always be available at each unit. However, environmental effects such as severe weather conditions may lead to prevention/delay of these manual measures.

5.A.2.2 Loss of the primary ultimate heat sink (e.g., loss of access to cooling water from the river, lake or sea, or loss of the main cooling tower)

In the event of loss of primary ultimate heat sink, blockages are assumed to occur both in the sea-water intake and the cooling water outlet. Additionally, recirculation is assumed to be unavailable.

If only blockage in the sea-water intakes occurs for Forsmark 3 the redundant auxiliary cooling water intake from the outlet tunnel is automatically opened and cooling water from the discharge tunnel will provide water to the cooling water pumps. This situation is within the design basis for Forsmark 3 and all acceptance criteria are met for this situation (the unit can be taken to a cold shut down state and maintained in a safe state).

Forsmark 1 and Forsmark 2 do not have the option to take water from the outlet.

<u>Analysis</u>

The Forsmark 1, 2 and 3 are of similar design and the initiating sequence will have equivalent impact on all units.

At loss of primary ultimate heat sink, water level in the intake basin will quickly drop and the main condenser cooling pumps will stop or start to cavitate. When the main cooling pumps stops, pressure will increase in the main condensers and initiate a reactor trip. The reactor trip system is of a fail-safe design and has a hydraulic control unit and a diversified electrical control rod drive mechanism.

Off-site power is postulated to be lost at the same time as the reactor trip is initiated. The main cooling pumps are eventually lost due to the blockage. This means amongst other things that the main feedwater system will trip.

The water level in the reactor pressure vessel is maintained by the high pressure safety injection system (which consists of four physically and functionally separated trains) powered by the EDGs.

Steam from the reactor pressure vessel is transferred by the safety relief valves to the suppression pool. Temperature will increase for the sea water remaining in the cooling water system (i.e. water that was already in the cooling water system when blockage of the intake occurred), which eventually will cause all the EDGs to overheat and automatically stop. The GT starts automatically at loss of off-site power.

The temperature in the suppression pool will rise and when it reaches 93 °C around 2.5 hours after initiation event, two out of four trains in the high pressure safety injection system at Forsmark 3 must be manually switched to the non-safety demineralised water storage tank and the remaining two trains will stop. At Forsmark 1 and Forsmark 2 the high pressure safety injection system is already connected to a separate demineralised water storage tank and no actions will be necessary.

Temperature in the suppression pool for all units will continue to rise and eventually start to boil. The only available alternative at this point is to open the valves to the MVSS and transfer residual heat from the reactor core to the atmosphere through the MVSS as described in the introduction to this chapter.

The suppression pool water inventory could be maintained by the demineralised water system according to normal procedures, or if this would not be available, by the independent containment spray system as described in the introduction to this chapter. Operators to perform these manual actions are always available at each unit. Environmental effects such as severe weather conditions are not expected to prevent/delay these manual interventions.

Core cooling with the high pressure safety injection system is maintained as long as there is either water in the demineralised water storage tank, there is fuel for the GT or as long as the temperature of the rooms that contains pumps for the high pressure safety injection system, the low pressure safety injection system and the containment spray system, are within limits.

After a certain time (the exact time depends on the operating mode of the high pressure safety injection system) the demineralised water storage tank must be refilled. If the high pressure safety injection system at Forsmark 1 and Forsmark 2 is operating in a reactor water level control mode, the water inventory in the demineralised water storage tank are expected to last for about 30 hours (assuming a nominal initial water volume). If two trains of the high pressure safety injection system are operating continuously at Forsmark 1 and Forsmark 2 , the demineralised water storage tank are expected to last for about 13 hours and if four trains are operating continuously, the demineralised water are expected to last for about 7 hours. If two trains in the high pressure safety injection system at Forsmark 3 are successfully switched to the non-safety demineralised water storage tank, the water inventory in the non-safety demineralised water storage tank for Forsmark 3, are expected to last about 8 hours.

Assuming that the high pressure safety injection system at Forsmark 1 and Forsmark 2 is operating in a reactor water level control mode, refilling of the demineralised water storage tank are expected to be necessary after 30 hours for Forsmark 1 and Forsmark 2 . For Forsmark 3, refilling of the demineralised water storage tank are expected to be necessary after 10 hours if two trains in the high pressure safety injection system are successfully switched to the non-safety demineralised water storage tank within 2.5 hours However, since the demineralised water system storage tank for Forsmark 1 and Forsmark 2 is subdivided, it would be a great advantage if refilling starts already within 24 hours for these units. If so, it will prevent the sub-volumes relating to the two operating trains from emptying, and prevent air ingestion from damaging related pumps.

High water level of the MVSS may occur during these events. This could be prevented by manually connecting hoses to the MVSS and discharge water inventory in a controlled manner.

Core damage is not expected to occur as long as the high pressure safety injection system is able to maintain water level in the reactor core and the residual heat can be removed to the atmosphere through the MVSS. For this, the following manual actions are required:

- For Forsmark 1 and Forsmark 2 , refilling of the demineralised water system storage tank must be manually performed within 30 hours.
- Additionally, manual actions must be in place within 24 hours to avoid air ingestion into high pressure safety injection system.
- For Forsmark 3, refilling of the demineralised water system storage tank from the fire-water system must be in place within 10 hours.
- For all units, the temperature of the rooms that contains pumps for the high pressure safety injection system, the low pressure safety injection system, and the containment

spray system, must be manually maintained within limits. This means that manual actions must be in place within 2.5 hours at all units.

- For all units, the fuel for the GT is sufficient for 4 days of full power operation. However, since the load on the GT is considerably smaller the full power operation, refilling of the fuel to the GT is estimated to be necessary within 16 days.

The spent fuel pool

If blockage in the sea-water intakes occurs, the spent fuel pool cooling systems will be lost.

The loss of primary ultimate heat sink is assumed to occur during refueling with a completely discharged core (i.e. all the fuel assemblies are placed in the spent fuel pools). The GT is assumed to be in operation.

The temperature in the spent fuel pool increase by approx. 10°C/hour without any cooling, and hence boiling will occur if no action is taken.

In order to maintain the water level in the spent fuel pools, refilling may be manually performed by connecting hoses to the fire-water system and manually provide make-up water to the pools. Operating procedures and staff to implement the measures are always expected to be available at all units. Environmental effects such as severe weather conditions are not expected to prevent/delay these manual actions.

After a few days the fire-water storage tanks would need to be refilled from the fresh-water system. Manual actions are required in order to connect the fresh-water system to the fire-water storage tank. Operating procedure for this are in place and staff to implement the measures are always expected to be available at all units. Environmental effects such as severe weather conditions may prevent/delay these manual interventions.

Analysis

If make-up water is not provided, fuel in the spent fuel pools are expected to be uncovered within 23 hours, for all units

For Forsmark 1 and Forsmark 2 the fire-water system storage tank must be refilled from the fresh-water storage, within 3 days. If the fire-water system could not be refilled, fuel in the spent fuel pools is expected to be uncovered within 4 days.

For Forsmark 3, the fire-water system storage tank must be refilled from the fresh-water storage, within 7 days. If the fire-water system could not be refilled, fuel in the spent fuel pools is expected to be uncovered within 8 days.

5.A.2.2.1 Results from the licensee assessments of protection against loss of ultimate heat sink

In case of loss of primary ultimate heat sink all units are expected to be shut down and maintained in a safe shut down condition without any damage to the fuel in the reactor core or in the spent fuel pools.

For all units, damage to the fuel in the reactor core is not expected to occur as long as the high pressure safety injection system is available and the residual heat in the reactor core can be removed via the suppression pool and transferred to the atmosphere through the MVSS.

The suppression pool level is expected to be maintained by the independent containment spray system.

For all units, damage to fuel in the spent fuel pools is not expected to occur as long as make-up water from the fire-water system can be manually provided. If make-up water for the spent fuel pools is not available, fuel in the spent fuel pools is expected to be uncovered within 23 hours.

If the event affects all units at the same time the situation is within the design basis for each unit and is expected to be handled without requiring any external water source.

5.A.2.2.2 Measures which can be envisaged to increase robustness of the plants in case of loss of ultimate heat sink

There are a number of measures, both simple and more complex, that are envisaged to improve the robustness of the units:

- Introducing improved operating procedures.
- Introducing comprehensive training of unit operators and the fire brigade personal.
- Increase the number of personnel available during night.
- Implement an improved automatic connection to the GT. (This is an on-going activity that are planned to be completed in 2013.)
- Implement the possibility to recirculate discharge water for Forsmark 1 and Forsmark 2 in the same way as already exist for Forsmark 3. (This is an on-going project.)
- Implement a new independent core cooling system.
- Introduce improved capabilities to provide make-up water (and emergency cooling) to the spent fuel pools without using hoses.

5.A.2.3 Loss of the primary ultimate heat sink and the alternate heat sink

No alternate ultimate heat sink exists for any unit at the Forsmark site.

5.A.2.3.1 Results from the licensee assessments of protection against loss of ultimate heat sink and loss of alternate heat sink

No alternate ultimate heat sink exists for any unit at the Forsmark site.

5.A.2.3.2 Measures which can be envisaged to increase robustness of the plants in case of loss of ultimate heat sink and loss of alternate heat sink

No alternate ultimate heat sink exists for any unit at the Forsmark site.

5.A.3 Loss of the primary ultimate heat sink, combined with station black out

At loss of the primary ultimate heat sink, combined with station black out the off-site power as well as the EDG backed-up power supplies for each unit are assumed to be lost.

<u>Analysis</u>

If the GT is assumed to function the scenario will be equivalent to the case of only loss of ultimate heat sink.

If the GT is assumed to fail the scenario will be the same for the loss of off-site power, loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power. The reason for this is that loss of off-site power, loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power, leads to loss of core cooling and loss of residual heat removal. (See subsection 5.A.1.3.)

5.A.3.1 Time of autonomy of the site before loss of normal cooling condition of the reactor core and spent fuel pool (e.g., start of water loss from the primary circuit).

Results are presented in subsection 5.A.1.3.

5.A.3.2 External actions foreseen to prevent fuel degradation

Results are presented in subsection 5.A.1.3.

5.A.3.3 Measures, which can be envisaged to increase robustness of the plants in case of loss of primary ultimate heat sink, combined with station black out

Results are presented in section 5.A.1.3.

5.B. Oskarshamn

5.B.1 Loss of electrical power

<u>The power supply systems</u>

The Oskarshamn site is connected to the external 400 kV grid and the external 130 kV grid. All units have both a 400 kV and a 130 kV transmission system interface.

There are two gas turbines (GT) at Oskarshamn 2 which can directly connect to the 130 kV switchyard. Because the130 kV switchyard and the two gas turbines are located within the site, power distribution to other units is available even if the site connections to the external grid are unavailable.

Although 12 hours house load operation is required by the grid codes, none of the units currently are in a state where this ability has been verified.

Unavailability of the normal grid connection will cause low voltage on the units' non-safety and safety classed power supply systems. Low voltage in the non-safety power supply system at all units will initiate an automatic reactor trip for all units. Low voltage in the emergency diesel generator (EDG) backed-up power supply systems, or in the GT backed-up power supply systems, will initiate an automatic start of the EDGs or the GTs respectively, to provide corresponding systems with power.

The main generator at Oskarshamn 1 is connected to the external 130 kV grid and to the non-safety and the safety classed power supply systems at Oskarshamn 1. The main generators at Oskarshamn 2 and Oskarshamn 3 are connected to the external 400 kV grid and to the non-safety and safety classed power supply systems at respectively unit. There are also connections from the 130 kV and from the GTs to the non-safety and safety classed power supply systems for each unit.

The normal power supplies for Oskarshamn 1 and Oskarshamn 2 are 6 kV and 10kV for Oskarshamn 3. Connections are in place between the non-safety power supply system and the stations transformers and the main generators. Units can be supplied with power from the main generators during normal operation and also from the external grid through the main transformer when the main generators are unavailable. The non-safety and safety classed power supply system at Oskarshamn 1 can also get power directly from the GTs. It is also possible to manually connect the power supply systems at Oskarshamn 3 to the GTs. However, this manual action will require some time before completion.

The emergency power supply systems are designed such that manual actions are not required during the first 30 minutes from initial event.

The ordinary back-up AC power supply system

The ordinary back-up AC power supply systems at Oskarshamn 1 and Oskarshamn 3 consist of four EDGs per unit, connected to the safety classed power supply systems at each unit. The ordinary back-up AC Power systems at Oskarshamn 2 consist of two EDGs and two GTs, both connected to the safety classed power supply system.

For Oskarshamn 1, two EDGs are full independence and are seismically qualified, and two EDGs have limited physical separation and are not seismically qualified. For Oskarshamn 2 the EDGs and the GTs are not formally seismically qualified. However, most of the support system has been evaluated by the SMA-method and the majority of the equipment was judged to be seismically adequate. For Oskarshamn 3 all four EDGs and the safety classed power supply system are seismically qualified.

For Oskarshamn 1 the safety classed power supply system and the EDG backed-up power supply systems have the same limitations as the corresponding EDGs regarding physical separation and seismic qualification

Oskarshamn 2 has in general a limited separation between divisions. The safety classed power supply system is based on a two main division structure with physical and functional separation between these two main divisions.

For Oskarshamn 1 some diversification exists between the EDGs, where two types of manufactures (two EDGs of each design) and different cooling principles are applied. Two EDGs are cooled by sea water and two EDGs are cooled by air. Also, the automation in the Oskarshamn 1 connection to the GT back-up power supply system, use a diversified technology.

For Oskarshamn 2 some diversification exists between the EDGs and the GT. The GT are independent of the sea-water cooling system and would therefore not be affected in the event of a loss of ultimate heat sink.

The alternate back-up AC power supply

In case of simultaneous failure of all EDGs at the site, the GTs may be credited as an alternate AC power source for Oskarshamn 1 and Oskarshamn 3. The GTs capacity is enough for simultaneous required safety supply to all units, including operation of core cooling functions and cooling of spent fuel pools.

For Oskarshamn 1 the EDG backed-up power supply system has alternate connections to the GT backed-up power supply. This is to secure automatic power supply to process functions in case of loss of off-site power in combination with a common cause failure (CCF) within the EDG backed-up power supply system. At a loss of off-site power and loss of the ordinary EDG backed-up power supply system, Oskarshamn 1 is provided with battery backed-up power supply systems and provisions for GT back-up power.

For Oskarshamn 3 manual action can be performed to enable the safety classed power supply systems to be connected to the GT backed-up power supply. Hence, at a loss of off-site power and loss of the ordinary EDG backed-up power Oskarshamn 3 is provided with battery backed-up power supply systems and provisions for GT back-up power if manual actions are successfully performed.

The battery backed-up AC and DC power supply systems

The battery backed-up AC and DC power supply systems for instrumentation, control and valve actuations at Oskarshamn 1 and Oskarshamn 2 , consists of separate battery backed-up AC and DC power supplies, for each of the units four divisions. These supplies can independently provide uninterruptible power to the reactor protection system and operator information through safety control panels and the safety desk in the main control room. The battery backed-up AC power supply system supplies mainly safety classed motor operated valves and the instrumentation and control system (I&C). The battery backed-up DC power supply system supplies mainly safety classed pneumatic valves, I&C equipment for reactor protection system, and the EDGs switchgear functions and relay protections, in each division.

For Oskarshamn 1, the safety classed battery backed-up AC and DC power supply systems are separated in four divisions with functional and physical independence and are seismically qualified. However, part of the cable routes at Oskarshamn 1, are only separated by distance or, in some cases, by fire-resistant lining.

For Oskarshamn 2, the battery backed-up AC and DC power supply system is based on a two main division structure with physical and functional separation between these two main divisions.

Oskarshamn 3 is in general based on a four division structure for the safety classed I&C and the emergency power supply systems. The safety classed battery backed-up AC and DC power supply systems are separated into four, seismically qualified, functional, and physically independent divisions.

For all units there are diversified automatic functions to supply the battery backed-up AC and DC power supply systems from the EDG backed-up power supply systems in case of failure within the uninterruptable power supply.

At Oskarshamn 1 all divisions of the battery backed-up AC and DC power supply systems have manually operated connections from the redundant EDG backed-up power supply system. These functions are used in case of a seismic event because two out of four EDGs are not seismically qualified.

The power supply system for the severe accident mitigation systems

All units are provided with equipment for filtered containment pressure release as described in the introduction to this chapter. For Oskarshamn 1 and 2 the MVSS is shared, and Oskarshamn 3 has a separate MVSS.

The power supply systems for the MVSS consist of two divisions. The MVSS power supply systems for Oskarshamn 1 and Oskarshamn 2 are non-safety and battery backed-up with a battery backed-up power source designed for a 24 hour condition. The MVSS power supply system for Oskarshamn 3 is safety classed, seismically qualified, and is battery backed-up with a battery backed-up power supply designed for a 24 hour condition.

For all units, there are prepared connections for external power generators to enable long term operation of the MVSS and enable transport of contaminated water from the MVSS, to the containments, in long term accident conditions.

Mobile power generators

Mobile power generators are located on site. A mobile diesel power generator is located at the site, and a petroleum-driven power generator is available from the MVSS (one at each MVSS).

During loss of off-site power, loss of the ordinary back-up AC power sources and loss of permanently installed alternate back-up AC power sources, both MVSS on site and the site emergency response centre will needs the site mobile diesel generator simultaneously.

5.B.1.1 Loss of off-site power

The loss of off-site power event is within the design basis for all units and presented in the safety analysis reports. There are no consequences expected for the fuel in neither the reactor cores, nor the spent fuel pools, as long as the ordinary back-up AC power supply systems are functioning at each unit.

Weather conditions could adversely affect functions and manual actions, which has been included in the analysis. However, no such effects have been identified in this analysis.

Fuel storages for the ordinary back-up AC power sources

At Oskarshamn 1, two EDGs are individually supported with a seismically qualified safety classed fuel system for 24 hours of operation. After 24 hours of operation, the safety classed fuel system is automatically refilled from a non-safety, and non-seismically qualified, on-site, fuel storage system common for the Oskarshamn 1 EDGs, and the Oskarshamn 2 EDGs and GTs. If EDG fuel is only required for Oskarshamn 1, the actual fuel storage would be sufficient for more than 7 days of EDG operation. The safety classed fuel system at Oskarshamn 1 can be manually refilled by external delivery via prepared connection points. Additionally, the common fuel storage tanks for Oskarshamn 1 and Oskarshamn 2 can be manually refilled by external delivery via prepared connections points.

For Oskarshamn 2, each GT and each EDG is individually supported by safety classed fuel systems for 8 hours of operation. After 8 hours of operation the safety classed fuel system is automatically refilled from a non-safety, and non-seismically qualified, on-site, fuel storage system, common for Oskarshamn 1 and Oskarshamn 2 . If the EDG fuel in the storage tanks is only required for Oskarshamn 2, the actual storage capacity would be sufficient for more than 7 days of EDGs and GTs operation. The safety classed EDG fuel tanks and the GT fuel tanks can also be manually refilled by external delivery via prepared connection point. Additionally, the common fuel storage tanks for Oskarshamn 1 and Oskarshamn 2 can be manually refilled by external delivery via prepared connection points.

Each EDG at Oskarshamn 3 is individually supported with a safety classed fuel system for 7 days of operation. After 7 days of operation, manual actions must be taken to refill the safety classed fuel system by external delivery via prepared connection points.

The lubricating oil storages

For Oskarshamn 1 lubricating oil must be manually refilled after approximately 2 days of operation of two EDGs, and after approximately 1 day of operation by the other two EDGs. The lubricating oil capacity on site is secured for more than 7 days of operation for two EDGs respective almost 72 hours of operation for the remaining two EDGs. Hence, external delivery of lubricating oil might be necessary within 72 hours.

For Oskarshamn 2 lubricating oil must be manually refilled for the EDGs after 8 hours of operation and the lubricating oil storage at site will be sufficient for almost 7 days of operation. Lubricating oil capacity for the GTs is however unknown, but historically refilling has only been needed during service (once a year).

For Oskarshamn 3, lubricating oil must be manually refilled after 2.5 days of operation and the lubricating oil storage on site is secured for more than 7 days of operation.

5.B.1.1.1 Results from the licensee assessments of protection against loss of off-site power

The plant is designed for loss of off-site power and fulfils requirements within its design basis. Weather conditions that can adversely affect functions and manual actions have been considered in the analysis. However, no such effects have been identified in this analysis.

On-site power sources can be maintained for at least 7 days under extreme weather conditions as stated in the safety analysis report. However, for Oskarshamn 1 external delivery of lubricating oil might be necessary within 72 hours.

5.B.1.1.2 Measures which can be envisaged to increase robustness of the plants in case of loss of loss of off-site power

The EDGs are equipped with dual lubricating oil and/or fuel filters. These two filters are connected in parallel via three-way valves so that the medium passes through both filters. By changing the initial position of the valves so that only one filter is used at a certain time, there would always be an unused filter available.

Existing instructions for the EDGs indicate that lubricating oil level should be greater than or equal to its minimum permissible level. The EDGs operability could therefore possibly be extended, by requiring that the lubricating oil level in the standby mode should be at its maximum permissible level instead of minimum.

For Oskarshamn 2 implementation of planned measures in the on-going modernization project will increase the robustness of the plant. After completion of the modernization project, Oskarshamn 2 will have four diesel generators by two different manufacturers (two of each kind) and with a fuel storage sufficient for 24 hours of operation.

Based on the operation and maintenance instructions, it must be ensured that required amount of consumables (engine oil, oil filters, etc.) are accessible in from the EDGs (and the GTs) location to withstand at least 72 hours of operation. Additionally, the central storage should ensure consumption for EDGs for a total of at least seven days of operation. In particular, the available lubricating oil volumes for two EDGs have to be increased.

Increased robustness may also be achieved by performing analyses to verify that it is possible to use lubricating oil from other units in case of emergency.

At loss of off-site power, the EDGs and/or the GTs may be needed in a long term condition. Existing instructions are developed with the assumption that the external grid return within a few hours. Review of existing relevant instructions and training/coaching of staff could be implemented to improve the ability to maintain the power supply function in a long term process. The conditions for long-term operation can be improved by improving the instructions in the following ways:

- Instruction based routines for regular drainage of condensate should be implemented because during loss of off-site power it is essential that the EDGs can utilize all the available fuel, without risk of disruption.
- Two EDGs at Oskarshamn 1 are cooled by a roof-mounted air-cooling system. During large amounts of snow, it is possible that the cooling system could be covered in snow and the capacity of the system might be reduced.
- Snow, ice and pollution can affect the combustion air supply, purity of the air, and cooling system capacity, and thus affect the EDGs output. Such effects are slow and can be prevented or mitigated by manual actions.
- Fuel supply in relation to long-term operation depends on the availability of the storage tanks and associated electric powered heaters, cables, and pumps. Reviewing of systems and equipment to ensure fuel supply for at least 7 days without any manual

actions, and without being adversely influenced by the initial event, or external weather conditions, could improve the robustness the site.

The following planned actions have been identified to increase the availability and reliability of the power system in a long term condition, (above 7 days of operation):

- Increase the possibilities to refill the diesel tanks at diesel units for Oskarshamn 1 and Oskarshamn 2 by ensuring that valves and pumps are fed by power from safety classed equipment.
- Perform analysis to verify that it is possible to use lubricating oil from other units in case of emergency without jeopardizing the operation.
- Investigate the possibilities to use diesel engines which today are used for the physical protection systems (security functions).
- Increase the number of mobile diesel generators at site (each facility should get its own mobile diesel unit).
- Future site configuration regarding the GTs and the 130kV switchyard should cover the need of alternate on-site power supply to all units, individually and simultaneously.
- Adoption of the U.S.NRC regulations 10 CFR 50.63 and Regulatory Guide (RG) 1.155 for all units.

5.B.1.2 Loss of off-site power and loss of the ordinary back-up AC power source

For Oskarshamn 1 and Oskarshamn 3, loss of off-site power and loss of ordinary back-up AC power source means failure of all four EDGs. For Oskarshamn 2 loss of off-site power and loss of ordinary back-up AC power source means that both the EDGs and the GT will fail and only the battery backed-up power supply system will be available. Sequences where only battery backed-up power supply systems are available will be further described in section 5.B.1.3.

Weather conditions could adversely affect functions and manual actions, which has been included in the analysis. However, no such effects have been identified in this analysis.

<u>The GT fuel storage</u>

Each GT is individually supported with a safety classed fuel system for 8 hours of operation. After 8 hours of operation, the safety classed fuel system is automatically refilled from a non-safety fuel storage system common for Oskarshamn 1 and Oskarshamn 2 . The total capacity of the fuel storage tanks will enable more than 7 days of GT operation.

<u>Analysis</u>

For Oskarshamn 1, failed house load operation function will cause low voltage in the unit's non-safety and safety classed power supply systems. Low voltage in the non-safety and safety classed power supply systems will initiate an automatic reactor trip, and an automatic function will connect the power supply systems to the GTs. Also, failures in two or more divisions of the EDG backed-up safety classed power supply system, automatically initiates an automatic function that connects the power supply systems to the GTs. The power supply systems can also be manually connected to GT backed-up power supply.

For Oskarshamn 3 failure of all EDGs will initiate manual actions to supply the unit from the GT backed-up power supply. If manual actions are successfully performed there will be no further consequences for the fuel located in the reactor core or in the spent fuel pools. Failed manual connection between Oskarshamn 3 and the GTs will results in a plant condition where only battery backed-up power is available. This case will be further discussed in section 5.B.1.3.

For Oskarshamn 2 the analysis will be further discussed in section 5.B.1.3.

5.B.1.2.1 Results from the licensee assessments of protection against loss of off-site power and loss of the ordinary back-up ac power source

At loss of off-site power and loss of the EDG backed-up power supply systems the Oskarshamn 1 and Oskarshamn 3 are provided with battery backed-up power supply systems and provisions for GT backed-up power supply. Operation of the GTs can be maintained for at least 7 days.

The GTs capacity is enough for simultaneous required safety supply to all units, including the core cooling functions and cooling of spent fuel pools.

5.B.1.2.2 Measures which can be envisaged to increase robustness of the plants in case of loss of off-site power and loss of the ordinary back-up ac power source

For Oskarshamn 2, implementation of planned measures in the on-going modernisation project is expected to increase the robustness of the plant by, for example, introducing four EDGs, by two different manufacturers (two of each kind). After completion of the modernisation project, the existing GTs will be excluded as an ordinary back-up AC power source, and can be credited as an alternate back-up AC power source (i.e. similar solution as for Oskarshamn 1 and Oskarshamn 3).

By introducing an automatic function, the reliability of the connection between the Oskarshamn 3 power supply system, and the GTs, could be improved.

5.B.1.3 Loss of off-site power and loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power sources

All units have safety-classed battery backed-up AC and DC power supply systems. Battery capacities for the safety-classed battery backed-up AC and DC power supply systems are designed for 2 hours of operation without charging.

Weather conditions could adversely affect functions and manual actions, which has been included in the analysis. However, no such effects have been identified in this analysis.

Analysis

For Forsmark 1 and Forsmark 2 and by crediting the MVSS capability to remove residual heat, damage to the reactor fuel is expected to be unavoidable after approximately 35 minutes. For Forsmark 3, damage to the fuel will be unavoidable after approximately 1 hour.

Oskarshamn 1 is equipped with a non-seismically qualified emergency condenser. The present design of the emergency condenser requires battery backed-up power supply for the isolation valves. Because the batteries are designed for 2 hours of operation, the available function during loss of off-site power, loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power sources, is limited. If the emergency condenser is assumed to be available, and by crediting the MVSS capability to remove residual heat, damage to the reactor fuel is expected to be unavoidable after approximately within 3 hours. However, if earthquake is considered, i.e. the emergency condenser is assumed to be unavailable, and by crediting the MVSS capability to remove residual heat, damage to the reactor fuel is expected to be unavoidable within 1 hour.

For Oskarshamn 2, with only battery backed-up power supply systems available, and by crediting the MVSS capability to remove residual heat, damage to the reactor fuel is expected to be unavoidable after approximately 2 hours.

For Oskarshamn 3, with only battery backed-up power supply systems available, by crediting the MVSS capability to remove residual heat, damage to the reactor fuel is expected to be unavoidable after approximately 1 hour.

The spent fuel pools

During loss of off-site power, loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power sources, the scenario will be the same as the scenario described in subsection 5.B.2.3, Loss of the primary ultimate heat sink (e.g., loss of access to cooling water from the river, lake or sea, or loss of the main cooling tower).

5.B.1.3.1 Results from the licensee assessments of protection against loss of off-site power, loss of the ordinary back-up ac power sources, and loss of permanently installed diverse back-up ac power sources

Battery capacities are designed for 2 hours of operation without charging.

For Oskarshamn 1, and if the emergency condenser is assumed to be available, damage to the fuel is calculated to begin within 3 hours. This assumes that battery power is available for the emergency condenser during the first 2 hours. However, if earthquake is considered, i.e. the emergency condenser is assumed to be unavailable, damage to the fuel is calculated to begin within 1 hour.

For Oskarshamn 2 damage to the fuel is calculated to begin after approximately 2 hours.

For Oskarshamn 3, damage to the fuel is calculated to begin after approximately 1 hour.

Release of activity is mitigated at sever accidents by means of the MVSS-function. However, if all units are affected simultaneously, both MVSS on site and the site emergency response centre requires the site mobile diesel generator.

5.B.1.3.2 Measures which can be envisaged to increase robustness of the plants in case of loss of off-site power, loss of the ordinary back-up ac power sources, and loss of permanently installed diverse back-up ac power sources

The primary MVSS functions are independent of power supply. However, battery back-up power supplies are required for providing process information to plant operators. Also, for long term conditions power supplies are required to enable transport of contaminated water from the MVSS and back to the containments. Therefore, increased robustness should be achieved by providing each MVSS with its own back-up power generator.

Increased robustness should also be achieved by extending the capacity of the battery backed-up power supplies to certain important loads during long term operation, e g. by disconnections of less important loads.

Battery back-up power supplies are limited to 2 hours of operation. Extending the capacity of the battery backed-up power supplies to certain important loads during long term operation, e g. by disconnections of less important loads, would increase the robustness of the plats.

An on-going project is supposed to provide the emergency response centre with its own permanently installed emergency power supply. This will decrease the demand of the site mobile diesel generator.

5.B.2 Loss of the ultimate heat sink

The Oskarshamn site is situated on a peninsula in the Baltic Sea and the ultimate heat sink for the Oskarshamn units is sea water from the Baltic Sea.

The alternate heat sink for Oskarshamn 1 is the atmosphere provided by the emergency condenser. In the emergency condenser the residual heat is removed to the atmosphere. Pumps providing demineralised makeup water to the emergency condenser are powered from air cooled diesel generators.

The alternate heat sink for Oskarshamn 2 is the sea water provided by the alternate suppression pool cooling capabilities. Fire water at Oskarshamn 2 can be connected to the secondary side of the cooling system for the suppression pool and discharged to the Baltic Sea trough either the outlet basin, the outlet tunnel or separate outlet pipe on the south side of the peninsula, to establishing alternate suppression pool cooling capabilities.

5.B.2.1 Design provisions to prevent the loss of the primary ultimate heat sink

The sea-water cooling systems

The cooling water inlets to Oskarshamn 1 and Oskarshamn 2 are connected. During 2010 and 2011 a deep water inlet has been constructed to Oskarshamn 1 and Oskarshamn 2 and is currently undergoing commissioning. The new deep water inlet is at present connected to only Oskarshamn 2 while the old surface water inlet provides cooling water to Oskarshamn 1. Later on, the deep water inlet will additionally be connected to Oskarshamn 1. Both inlets, the old surface water inlet and the deep water inlet, individually have sufficient capacity to provide cooling water to both Oskarshamn 1 and 2. The inlets could be shifted manually.

The deep water inlets are designed to eliminate problems with icing due to sub-cooled water, drifting ice, heavy oil, fish, sea weed and other biological material. The deep water inlet for Oskarshamn 1 and Oskarshamn 2 is approximately 18 m below the normal sea level. The corresponding deep water rock tunnel has a capacity of 60 m^3/s. For the Oskarshamn 1 and Oskarshamn 2 a rock tunnel leads the water from the deep water inlet to an intake pool.

In case of large oil spill in the Baltic Sea, the deep water inlets and the surface water inlets for Oskarshamn 1 and Oskarshamn 2 , can if needed, be switch off, and recirculation can established via the return tunnel from the outlet. This means that if both inlets are unavailable, the units can still be shut down to a safe state using the recirculation channel from the outlet, which is manually opened. During these situations the safety classed sea-water system will use the outlet for cooling. The recirculation tunnel for Oskarshamn 1 and Oskarshamn 2 is designed to provide cooling water for both units at the same time.

The surface water inlets for Oskarshamn 1 and Oskarshamn 2 are divided by a man made dam between the shore and an island. The dam separates the intake pool of the deep water inlet from the surface water inlet. Outside the surface water inlet there is a barrier to prevent any floating objects from entering further and to prevent any oil from reaching the inlet. There are wreck screens at the intake structure for both surface and deep water inlets to stop things larger than 100 mm, for example driftwood or ice floe.

Cooling water is conveyed from the Oskarshamn 1 and Oskarshamn 2 inlet structure through separate rock tunnels to a screening house at Oskarshamn 1 and Oskarshamn 2 , respectively.

The cooling water inlet to Oskarshamn 3 is a deep water inlet. The inlet concrete structure is shaped to enable horizontal flow. The horizontal flow in deep water was designed to eliminate problems with icing due to subcooled water, heavy oil, fish and other biological material such as sea weed and jelly fish. In case of a large oil spill in the Baltic Sea, the deep water inlet can be shut off and recirculation established via the return tunnels from the outlet. The water is conveyed from the inlet structure in the Baltic Sea through a rock tunnel to a water intake basin.

All units have four separated sea-water channels with rinsing equipment and individual basins upstream and downstream of the sea-water channels. The water is rinsed in the screening house by trash racks (20 mm distance between bars) and travelling basket screens to be cleansed of particles and ice particle smaller than 2 mm. Icing (by subcooled water) of the travelling basket screens is prevented by warm spray water. Clogging is prevented by monitoring differential pressure over the trash racks and the travelling basket screens and monitoring of the sea-water level. Automatic rinsing starts by differential level or pressure drop over each of the four rinsing channels. Each safety sea-water systems also have filter downstream of each sea-water safety pump in case a breakthrough of the travelling basket screens takes place. These filters are backwashed automatically at high differential pressure or on a set time interval.

The cooling water outlet goes to an outlet basin for each unit and is then conveyed through rock tunnels to the bay on the other side of the peninsula. There is a rock tunnel outlet from each unit.

Oskarshamn 2 is currently being modernised mainly to fulfil new requirements from the Swedish Radiation Safety Authority. One on-going modification is to add a new diversified residual heat removal cooling chain (two trains) which will take water from the outlet basin of Oskarshamn 2 and return the water to the outlet water basin for Oskarshamn 1. The new cooling system will enable residual heat removal from full reactor power down to cold shut down. The new systems will be taken into operation in the when the on-going modernisation project has been completed.

The total normal cooling water flow is 22 m^3/s at Oskarshamn 1, 26 m^3/s at Oskarshamn 2 (after modernisation flow will increase to 35 m^3/s) and 55 m^3/s at Oskarshamn 3. The safety systems require only a fraction of this flow, which for Oskarshamn 1 would be a total sea-water cooling flow of around 450 kg/s, for Oskarshamn 2 a total sea-water cooling flow of around 400 kg/s, and for Oskarshamn 3 a total sea-water cooling flow of around 760 kg/s .

The required minimum sea-water level for the safety sea-water cooling pumps is 4 m at Oskarshamn 1, 8 m at Oskarshamn 2 and about 4 m at Oskarshamn 3, below normal water level in the sea.

The safety sea-water cooling pumps and their filters are installed in rooms that can withstand a sea-water level of approximately 6 m at Oskarshamn 1, at 3 m at Oskarshamn 2 and 4 m at Oskarshamn 3, above normal sea-water level.

The maximum sea-water temperature for operating the plants is 25-26°C according to the operating limits and conditions.

The sea-water cooling pumps for the Oskarshamn 2 EDGs are designed for underwater functions and will not be affected by flooding.

5.B.2.1.1 The spent fuel pools

Oskarshamn 1 has one spent fuel pool located next to the reactor pool. The spent fuel pool and the reactor pools are connected by a manoeuvrable gate. The gate will automatically close if the water level decreases to about 2 meters below normal level in the pool. If this happens there will still be more than 5 meters of water above the fuel assemblies. The spent fuel pool is also connected by a manoeuvrable gate to the fuel transport cask pool, which is a specific pool used for loading of the transport cask for spent fuel. The water volume of the spent fuel pool is around 1300 m^3. The water volume of the reactor pool is 950 m^3 during refueling and around 800 m^3 during power operation, and the water volume of the Fuel Transport Cask Pool is around 140 m^3. There are no pipe connections at the lower part of the spent fuel pool. The pipe connections are in the upper part of the pools to maintain at least 5 m of water on top of the fuel assemblies in case of a break in connecting pipes. The spent fuel pool is cooled by two physically and functionally separated trains. Each spent fuel pool cooling train has one pump common to the containment spray system. This cooling system is the same as for the suppression pool. If one train is lost, the capacity in the remaining train is sufficient to cool both the spent fuel pool and the suppression pool. In this case, the cooling is alternated between the spent fuel pool the suppression pool cooling. There is also an alternate cooling system that has the same capacity as the spent fuel pool cooling system trains and could therefore be used as a back-up cooling system. This cooling system is normally used to cool the emergency condenser.

Oskarshamn 2 has two spent fuel pools. The reactor pool and the pool for internal parts are located next to the spent fuel pools. The pools are connected by a manoeuvrable gate. The gate will automatically close if the water level decreases to about 2 meters below normal level in the pool. If this happens there will still be more than 5 meters of water above the fuel assemblies. One of the spent fuel pools are also connected by a manoeuvrable gate to the fuel transport cask pool, which is a specific pool used for loading of the transport cask for spent fuel. The water volumes in the spent fuel pools are 500 m^3 respectively 920 m^3. The water volume of the pool for internal parts is about 1350 m^3 and the water volume of the reactor pool is around 990 m^3, both during refueling. The water volume of the fuel transport cask pool is around 140 m^3. There are no pipe connections at the lower part of the spent fuel pools. The pipe connections are in the upper part of the pools to maintain at least 5 m of water on top of the fuel assemblies in case of a break in connecting pipes. In the pool for reactor internal parts there is a strainer and a pipe connection to the core spray system with the centreline at around 3 m above the bottom of the pool. This pipe connection could also be used to backwash the strainers in the suppression pool (one of three different backwashing possibilities) and/or to provide demineralised water to the core spray pumps if the suppression pool is not available for the same reason. Normal cooling of the spent fuel pools is performed by the fuel pool cooling system, which is powered by the GTs. Because the spent fuel pool cooling system is not fully safety classed, there are additional connections to the safety classed cooling systems, which are powered from the EDGs.

Oskarshamn 3 has two spent fuel pools. The reactor pool and the pools for internal parts are located in between the two spent fuel pools. All the pools are connected by a manoeuvrable gate. The gate will automatically close if the water level decreases to about 2 meters below normal level in the pool. If this happens there will still be more than 5 meters of water above the fuel assemblies. One of the spent fuel pools are also connected by a manoeuvrable gate to the fuel transport cask pool, which is a specific pool used for loading of the transport cask for spent fuel. The water volumes in the spent fuel pools are 590 m^3 respectively 790 m^3. The water volumes of the pools for internal parts are 630 m^3 each and volume of the reactor pool is 850 m^3. The water volume of the fuel transport cask pool is 250 m^3. There are no pipe connections at the lower part of the spent fuel pool. The pipe connections are in the upper part of the pools to maintain at least 5 m of water on top of the fuel assemblies in case of a break in connecting pipes. Normal cooling of the fuel pools is performed by the fuel pool cooling system which has two redundant trains (one in standby during normal operation). The levels in the pools are maintained by the demineralised water system or the radwaste system.

All reactor pools are temporary emptied during refueling outages in order to remove the containment head and the reactor pressure vessel head. After the heads are removed the reactor pool is refilled. There are different cooling systems used to for the residual heat removal during a refueling outage. In the beginning of the outage the residual heat removal system is used, and for Oskarshamn 2, this system is accompanied with the spent fuel pool cooling system. When all fuel assemblies have been moved from the reactor pressure vessel to the spent fuel pool the residual heat removal system can be shut off and the spent fuel pool cooling system will be able to provide sufficient cooling for residual heat removal. The ultimate heat sink for all cooling trains connected to the spent fuel pool is sea water.

The spent fuel pools levels and temperatures are monitored in the main control room and will initiate an alarm at low level or high temperature. The design temperature for the spent fuel pools is 60°C.

5.B.2.1.2 Design provisions to mitigate loss of primary ultimate heat sink

All important water storage tanks on site can withstand all bad weather conditions within design.

The fresh-water supply systems

The normal fresh-water supply is taken from a nearby lake. The lake is situated within 10 km of the site. Water in the lake is very clean because it has no environmentally negative inlets. For drinking water purposes, it does not need any chlorine to be added. The fresh-water storage in the lake is expected to be enough for more than one year of fresh-water supply to one unit. The maximum pumping capacity is around 40 kg/s. The pump station has redundant pumps powered from the normal power grid.

The back-up fresh-water reservoir is situated on the site. It was created when the plant was built by enclosing one part of the bay by a man-made dam. The bay was emptied after the dam was built and filled with fresh water from a creek. The water reservoir contains approximately 120 000 m^3 of fresh water. The fresh-water storage in the reservoir is expected to be enough for at least five months of normal operation of all three units without any refilling from the creek. The pumping station consists of two trains with one pump in each train. Each train has a capacity of around 30 kg/s. If both trains are coupled together to one train, the combined capacity is reduced to less than 40 kg/s. The pump station is powered from the GTs at Oskarshamn 2 and operation of the pump stations is handled by the Oskarshamn 2.

Fresh water from the fresh-water supplies is treated in the raw water system. The system consists of two trains and produces drinking water. Maximum capacity is around 40 kg/s. The system has two tanks with drinking water. Each tank has a capacity of around 320 m^3. Production of drinking water starts when the volume in any of the tanks is at around 170 m^3. The capacity is determined by the requirement to fill the fire-water tanks at the units.

There is also a drinking water reservoir that is filled from the drinking water storage. The capacity from the drinking water storage to this system is around 10 kg/s. The total volume in the system with drinking water is 970 m^3 (three different storage tanks) Water in the system is distributed to consumers through the drinking water distribution system.

The fire-water systems

Fresh water is also stored in the fire-water tanks at the site.

Oskarshamn 1 and Oskarshamn 2 have two common fire-water tanks. The minimum volume in these fire-water tanks according to operational limits is 1200 m^3 in each tank. The tanks are normally filled to 1500 m^3 each. The fire-water tanks are located outdoors on ground level (6 m above normal sea water) in close connection to the water treatment plant where also the fire-water pumps are installed. The common fire-water pumps for Oskarshamn 1 and Oskarshamn 2 consist of one electrical driven pump and one diesel driven pump. There are also two small pumps for small consumers, and also to maintain pressure in the system. The electrical pumps are GT backed-up. The diesel driven pump and the main electrical driven pump each have a capacity of 200 kg/s.

Oskarshamn 3 has two separate fire-water tanks. The minimum volume in these fire-water tanks according to operational limits and is 1200 m^3 in each tank. The tanks are normally filled to 1500 m^3 each. The tanks are normally filled with drinking water from the fresh-water plant. The fire-water pumps consist of two direct driven diesel pumps and two electrical driven pumps. The diesel driven pumps each have a capacity of 150 kg/s at and the electrical pumps each have a capacity of 75 kg/s. There is also a small pump for small consumers and to maintain pressure in the system. The electrical pumps can be manually supplied by the GTs if off-site power is lost.

The fresh-water supply has a capacity of around 40 kg/s to fill the fire-water tanks.

For back-up purposes, pipe connections for the fire-water systems exist between the units.

The demineralised water supply systems

Demineralised water is produced in the water treatment plant. The water treatment plant is common for the site but operation lies within Oskarshamn 2. The demineralised water plant has two EDG back-up trains with a capacity of 60 m^3/h in each train. Both trains can be operated together to produce 120 m^3/h. Production starts automatically when the demineralised water storage tanks in any of the units needs refilling. The capacity is such that at least 24 hours of production can occur without need for regeneration. Water from the fresh-water supply is treated in the demineralised water plant and finally refills the demineralised water storage tanks in each unit. Production starts at low level in any one of the site tanks and the normal refilling rate is 30 m^3/h per train. If an additional low level signal comes from any other tanks, the refilling rate is automatically increased to 60 m^3/h per train.

The liquid radwaste system

In the liquid radwaste systems there are tanks containing purified system drainage for the purified water train. The purpose of the purified water trains is to form a closed loop for the process. This system is powered from the non-safety power supply system but could possibly be supplied by the GTs.

The ordinary back-up AC power source cooling systems

At Oskarshamn 1 the ordinary back-up AC power source consists of four separated EDGs. Two of the EDGs are cooled by sea-water and two EDGs are cooled by an intermediate water system which is air-cooled by forced-circulation cooling towers.

The two EDGs at Oskarshamn 2 are sea-water cooled and the two GTs are independent of the sea-water systems. The GTs are initiated at loss of off-site power and is automatically connected to the unit safety classed power supply systems. All safety systems at Oskarshamn 2 will be powered from the GTs if the EDGs fails. After manual release the GTs will also be able to provide power to the demineralised water plant, the fire-water pumps, the fresh-water plant and the fresh-water system.

The four EDGs at Oskarshamn 3 are and functionally separated and cooled by sea water.

The emergency condenser

Oskarshamn 1 is design to withstand loss of primary ultimate heat sink. The protection system for this consists of an emergency safety condenser that uses the atmosphere as the heat sink. The emergency condenser has two redundant trains. The system is connected to the steam lines and the condensate returns to the reactor main recirculation system. The connections to the steam lines are normally open during power operation while the condensate return lines are closed. Once the condensate return valves to the main recirculation lines are opened, the emergency condenser operation will be passive. Make-up water to compensate for the steam boiling off in the emergency condenser tank is supplied from the demineralised water system. In addition, the water in the spent fuel pool could also serve as an alternate emergency water source. By opening a valve the emergency condenser can be refilled from the spent fuel pool by the hydrostatic head difference.

The independent suppression pool cooling system

Oskarshamn 2 has a specific provision to handle loss of ultimate heat sink. The suppression pool is normally cooled by the cooling system for the containment spray system. During loss of ultimate heat sink the fire-water system could be connected to the secondary side of the cooling system for the suppression pool to provide residual heat removal. The fire-water will pass through the secondary side of the cooling system and then be discharged to the discharge basin (the same basin as the discharge from the sea-water cooling systems). The discharge basin has piped connection both to the outlet water basin and to the deep water inlet pool. Pumps in the cooling system for the suppression pool are powered from the GTs. Manual actions according to the emergency procedures are required to enable this function. The fire-water pumps are both independent of the sea-water cooling systems.

The accident mitigation system for containment filtered pressure release

When nothing else is available, the containment filtered pressure release system, MVSS, could be used to maintain the residual heat removal of the reactors. For Oskarshamn 1 and 2 the MVSS is shared. Oskarshamn 3 has a separate MVSS. The capacity of the MVSS for Oskarshamn 1 and Oskarshamn 2 is the same as Oskarshamn 3 and are expected to be able to handle the residual heat from both units at the same time. Also, because of difference in design, analyses shows that actuation of the MVSS will not occur at the same time in both units. According to calculations, Oskarshamn 1 will require the MVSS later than Oskarshamn 2.

Water level in the MVSSs could be maintained by manually initiating recirculation of the condensed water from the MVSSs to the containments by using an electrical operated pump. The MVSSs could be cooled by connecting external fire water to the MVSS cooling systems. To not overfill the MVSSs, the recirculation mode where the MVSS water is pumped back to the containments, must be in operation.

The power supply for the MVSS is described further in subsection 5.B.1.

Manual opening of valves in the MVSS could be performed from a location outside the main control room, and the activity release from the MVSS stack could be monitored from the same location.

The independent containment spray system

For scenarios where the suppression pool is boiling water inventories could be maintained by replacing boiled off water with sprayed water from the independent part of the containment spray systems. These systems are connected to the fire-water systems.

For Oskarshamn 1 and Oskarshamn 2 this function contains one direct driven fire diesel pump (capacity 200 kg/s). The main water source for Oskarshamn 1 and Oskarshamn 2 is two 1500 m^3 water tanks with drinking water (minimum allowable volume of 1200 m^3 each). By opening a valve in one of the two trains for Oskarshamn 1 and Oskarshamn 2 , a rupture disc will open and the water will be sprayed in the primary containment and conveyed to the suppression pool.

Manual opening of valves will be done from a location outside the main control room. The temperature and level in the containment as well as in the suppression pool could also be monitored from this location.

Instrumentation

Level measurements in the suppression pool and the MVSS are performed by measuring differential pressure. The instrumentation lines are outside the containment and will not be affected by boiling in the suppression pool.

The level and pressure instrumentation in the reactor pressure vessel has instrumentation lines inside the containment and is powered from batteries with 2 hours design capacity.

Preventive measures for hydrogen detonation

At high temperature in the reactor core, hydrogen from radiolysis in the reactor water is transported to the suppression pool by the safety relief valves. The hydrogen is then transported to the drywell through the vacuum breakers in the diaphragm floor and vented to the MVSS. The containment, the pipes to the MVSS and the MVSS itself are all nitrogen filled during power operation to prevent hydrogen detonation.

5.B.2.2 Loss of the primary ultimate heat sink (e.g., loss of access to cooling water from the river, lake or sea, or loss of the main cooling tower)

In an event of loss of primary ultimate heat sink, blockages occur both in the intake and the outlet. Additionally, recirculation is assumed to be unavailable.

If blockage in the sea-water intake occurs for any unit at site, the water level in the intake basin will quickly drop and the main condenser cooling pumps will stop or start to cavitate. The reactors are tripped after a few seconds because the main cooling pumps stop which will cause a pressure increase in the main condensers. The reactor trip system is of a fail-safe design and has a hydraulic control unit and a diversified electrical control rod drive mechanism. If only the intake is blocked the safety sea-water pumps will continue to operate. By manually opening valves in the return channel the outlet on the other side of the peninsula can be used for emergency cooling of the main sea-water cooling system. This situation is within the design basis and all acceptance criteria are met. Units are taken to a safe state and maintained in a safe state.

Analysis

When loss of primary ultimate heat sink occurs for Oskarshamn 1, blockages of both the intake and the outlet have taken place. The blockage occurs so fast that manual actions cannot be credited. In this event the reactor is automatically tripped at loss of main cooling pumps due to high pressure in the main condenser. Off-site power is postulated to be lost at the same time as the reactor trip is initiated. The safety sea-water pumps are eventually lost due to the blockage. This means, amongst other things, that the feedwater system is lost. The emergency condenser and the auxiliary feedwater system at Oskarshamn 1are initiated at low level in the reactor. The emergency condenser will automatically keep the reactor pressure at constant pressure. However, during the first minute the capacity of the emergency condenser is not enough to keep the reactor pressure at constant. Therefore, four relief valves will open in the pressure relief systems and transfer steam to the suppression pool. This will cause a small temperature increase in the suppression pool. The condensed steam in the emergency condenser is returned to the reactor by the condensate lines to the main recirculation system. To make up for losses from the reactor pressure vessel and to compensate for the density difference at shut down, the high pressure injection pump provide water to the reactor pressure vessel and maintains constant reactor level. The demineralised water is taken from the demineralised water storage tank. The water reservoir in the emergency condenser is enough for residual heat removal for approximately 6 hours without any refilling. Emergency refilling from the spent fuel pool will add another 12 hours for residual heat removal with the emergency condenser. Demineralised make-up water is provided to the emergency condenser by air-cooled EDGs powered makeup pumps. Only one pump is required to operate at these conditions.

At loss of primary ultimate heat sink for Oskarshamn 2, the alternative circulation of water from the Oskarshamn 1 outlet return channel to the Oskarshamn 2 inlet is not working because the intake system is assumed to be totally blocked. The reactor is automatically tripped at loss of main condenser cooling pumps, due to high pressure in the main condenser. Off-site power is assumed to be lost at the same time as the reactor trip is initiated. Cooling of the suppression pool is imitated and the emergency feedwater system starts pumping water to the reactor. The cooling water system will be available as long as the cooling water pumps can take water from the intake basin down the streams to the travelling baskets screens which last at least 10 minutes. When the cooling systems are lost the EDGs will overheat and automatically stop. However, the GTs will automatically start at loss of off-site power. Isolation of the reactor containment spray system will be initiated after 70 minutes due to the temperature increase that follows a the loss of containment cooling. This will automatically initiate the containment spray system. Since the ventilation systems are cooled by the sea-water cooling systems, the containment spray system is assumed to be manually stopped after 3 hours due to high temperatures in the pump motors. Reactor cool down is assumed to start after 75 minutes when the containment cooling is lost. After 4 - 5 hours the suppression pool temperature will reach 93°C, which is the upper limit to enable steam to condense in the suppression pool. At this time the alternate suppression pool cooling system is assumed to be manually taken into operation, and fire water will start to cool the suppression pool. If the fire-water system is assumed to fail, the suppression pool could be cooled either by the demineralised water system or by water from the reactor pool. This situation is within the design basis and all acceptance criteria are met. The unit is taken to cold shut down condition and can be kept there in a safe manner.

When loss of primary ultimate heat sink occurs for Oskarshamn 3, blockages of both the intake and the outlet have taken place. The blockage occurs so fast that manual actions cannot be credited. In this event the reactor is automatically tripped at loss of main cooling pumps due to high pressure in the main condenser. Off-site power is postulated to be lost at the same time as the reactor trip is initiated. The safety sea-water pumps are eventually lost due to the blockage. This means, amongst other things, that the feedwater system is lost. The water level in the reactor pressure vessel is maintained by the high pressure injection system (four physically and functionally separated trains) which is powered from the EDGs. The high pressure injection system takes water from the suppression pool and injects the water into the reactor pressure vessel. Steam is dumped to the suppression pool by safety relief valves. Temperature will increase for the sea-water remaining in the system (i.e. water that was already assumed to be in the system before the sea-water blockage of the intake occurred), which eventually will cause all the EDGs to overheat and automatically stop. The GTs starts automatically at loss of off-site power or on a reactor trip at Oskarshamn 2, but there is no automatic connection between Oskarshamn 3 and the GTs. Hence, manual actions in the main control room must be performed by the operators to connect Oskarshamn 3 to the

GTs. The time needed for connecting the GTs to Oskarshamn 3 is approximately 1 hour. During this time the temperature in the suppression pool will rise and at high temperature (95°C) in the suppression pool, the operator in the main control room will manually initiate the reactor protection system. The water supply for the high pressure injection systems is then automatically transferred to the demineralised water system which takes water from the demineralised water storage tank. The demineralised water system will also perform cooling in the rooms where pumps for the high pressure injection system are installed. One train of the demineralised water system and the high pressure injection system is enough to maintain the water level in the reactor pressure vessel during this event. The demineralised water storage tank is filled from the fresh-water system in the radwaste building and/or from the demineralised water plant. However, the temperature in the suppression pool will continue to rise and eventually start to boil. The only available alternative at this point is to open the valves to the MVSS. The water level in the MVSS is stabilized at about 1 m above normal level. The suppression pool water inventory could be maintained by using demineralised water for make-up water according to the normal procedures for maintaining water level in the suppression pool. In an emergency situation, water could also be added from the fire-water system.

The spent fuel pool

The design basis postulated for Oskarshamn 1 in the safety analysis report during power operation is a total loss of all spent fuel pool cooling system. In this case one train of the safety classed cooling system will be assumed to function and there will be no further consequences.

The design basis postulated for Oskarshamn 2 in the safety analysis report during power operation is a total loss of fuel pool cooling system by a crack in the system main pipe lines. In this case, there is enough time to perform the repair or to start so called "feed-and-bleed" (adding cool water and removing heated water). The case is handled by emergency procedures.

The design basis postulated for Oskarshamn 3 in the safety analysis report during power operation is a total loss of all spent fuel pool cooling system. Additionally, the back-up cooling systems and the redundant cooling systems are all postulated to fail. The result shows that it takes 1 day to reach the design temperature 60°C.

The design basis postulated for Oskarshamn 1and Oskarshamn 2 in the safety analysis report during refueling outages is separated in to different cases by considering different stages during the outage. The most limiting case is total loss of cooling systems immediately after the outage, when all spent fuel has been removed from the reactor core to the spent fuel pool, which also contains spent fuel from previous outages. If the safety classed cooling system is assumed to fail, the temperature will increase in the spent fuel pool. For Oskarshamn 1 and in the most limiting case, the design temperature (60°C) is reached within 10 hours. Cooling will be performed by so called "feed-and-bleed", where cold demineralised water is added to the pool and at the same time heated water from the pool is removed to the fuel transport cask pool. A flow rate of 40 kg/s is initially needed for sufficient cooling (keep water temperature below design temperature) which requires one pump to operate. For Oskarshamn 2 and in the most limiting case, the design temperature (60°C) is reached within 70 hours if the gates between the reactor pool and the spent fuel pools are open and after about 1 day if gates are closed. Cooling will be performed by so called "feed-and-bleed", where cold demineralised water is added to the pool using the containment spray system or the reactor core spray system supplied by the suppression pool, and at the same time heated water from the pools are removed to the to the suppression pool or to the liquid radwaste systems via the fuel transport cask pool. These scenarios are described in emergency procedures.

The design basis postulated for Oskarshamn 3 in the safety analysis report during refueling outages is separated in to different cases by considering different stages during the outage. The most limiting case is total loss of cooling systems immediately after the outage, when all spent fuel has been removed from the reactor core to the spent fuel pool, which also contains spent fuel from previous outages. The temperature will increase in the spent fuel pool. Cooling will be performed by so called "feed-and-bleed", where cold demineralised water is added to the pool and at the same time heated water from the pool is removed. A flow rate of around 20 kg/s is initially needed for sufficient cooling (keep water temperature below the design temperature). Cooling by so called "feed-and-bleed", can be performed by using the high pressure injection system, the low pressure injection system or the demineralised water system to add water, and at the same time, the heated water from the pools are removed to the liquid radwaste systems. These scenarios are described in emergency procedures. Results show that temperature in the spent fuel pool will not exceed the design limit. Since the event is in the design basis for the plant design, loss of off-site power has been postulated in the analysis.

Analysis

Loss of ultimate heat sink according to the stress test specification will causes all cooling systems for the spent fuel pools to fail. Temperatures in the spent fuel pools will increase until boiling take place. Engineering judgment indicates that the integrity of the spent fuel pool is likely to be preserved at a water temperature of 100°C.

In a situation where all cooling systems for the spent fuel pools have failed, hoses must be manually carried to the spent fuel pools. This is done according to the procedure for abnormal events.

If the fire-water pumps fail at one of the units, the neighbouring units' fire-water system could probably be used. If all fire-water systems are assumed to be unavailable, the fire truck at the site could be used. The capacity of pumps installed on fire trucks is usually around 40 kg/s.

The ventilation systems in the spent fuel pool area are assumed to function and the steam is vented out to the main stack. Some condensed water will drain to the pools and might also accumulate in other parts of the building. This will however not affect the safety core cooling systems because they are installed in water tight compartments that can withstand water up to ground level. If so called "bleeding" is needed from the reactor pool and the spent fuel pool, this could be performed manually by opening valves to the radwaste system tanks. Water could then be purified and reused or discharged into the Baltic Sea without any significant release. The make-up water capacity in the fire-water systems is enough to compensate for the boiling in the spent fuel pools for at least 90 - 250 days for Oskarshamn 1 and Oskarshamn 2 , and 18-66 days for Oskarshamn 3, depending on when the event occurs.

The residual heat during a refueling outage is much less than immediately after the reactor trip during power operation. The capacity of several of core cooling systems (e.g. the core spray system, the high pressure injection system/emergency feedwater system, the containment spray system, and the demineralised water system) is however enough to provide make-up water for the spent fuel pool and the reactor pool. Also, the water volumes in the suppression pools are enough to maintain water level in the spent fuel pools for approximately 6-7 days without refilling.

5.B.2.2.1 Results from the licensee assessments of protection against loss of ultimate heat sink

In case of total blockage of sea-water intake, all units can be shut down and maintained in a safe shut down conditions without any damage to the fuel in the reactor cores or in the spent fuel pools.

If both sea-water intake and outlet is assumed to be blocked, core cooling and residual heat removal will be maintained, and all units are expected to be shut down and maintained in a safe state.

For Oskarshamn 1 there has been no cliff edge effects identified as long as the emergency condenser together with the air-cooled EDGs are operational.

For Oskarshamn 2 there has been no cliff edge effects identified as long as the GTs and the alternate suppression pool cooling system are available.

For Oskarshamn 3 there has been no fuel damage in the reactor core identified as long as one train of the high pressure injection system is operational, and the connection between Oskarshamn 3 and the GTs will be manually accomplished within 1 hour. This situation requires also that the residual heat from the reactor core will be transferred to the atmosphere via the suppression pool, and the MVSS. Suppression pool level will then be maintained by the independent containment spray system and the MVSS will be cooled by the external cooling system.

If the event affects all units at the same time the situation is within the design basis for each unit and will be handled without requiring any external water source.

For the spent fuel pools, the make-up water capacities in the fire-water systems are expected to be enough to prevent fuel damage in the spent fuel pools for at least 90-250 days for Oskarshamn 1 and Oskarshamn 2 , and 18-66 days for Oskarshamn 3, depending on when the event occurs.

5.B.2.2.2 Measures which can be envisaged to increase robustness of the plants in case of loss of ultimate heat sink

A possible improvement would be to install new pipes to provide fire water to the spent fuel pools without entering the spent fuel pool areas. As a suggestion, these pipes should be able to be supplied by fire water in the same manner as the independent containment spray system. Such an installation would minimize manual actions required to establish "feed-and-bleed" of the spent fuel pool during an accident.

Another possible improvement at Oskarshamn 1 and Oskarshamn 3 would be to install a connection from the independent containment spray system to the suppression pool cooling system, in order to enable cooling of the containments and suppression pools in the same manner as has already been implemented at Oskarshamn 2.

Also, by implementing an automatic connection between Oskarshamn 3 and the GTs, the unit's capability to handle these situations would be improved.

5.B.2.3 Loss of the primary ultimate heat sink and the alternate heat sink

Analysis

In case of loss of the primary ultimate heat sink and the alternate heat sink, at Oskarshamn 1, two scenarios have been identified. The first scenario assumes only the emergency condenser to fail. In this case the emergency feedwater system powered from the air-cooled EDGs will restore the reactor water level and maintain a normal water level in the reactor. No fuel damage will take place. However, because the emergency condenser is assumed to fail, reactor steam is dumped to the suppression pool by relief valves. The suppression pool temperature, which normally is controlled by the sea-water cooling system, will thereby increase and eventually water in the suppression pool will start to boil. The suppression pool cooling system is powered by sea-water cooled EDGs and will therefore also be lost in this scenario. Failed cooling of the suppression pool will lead to a containment pressure increase. The only available alternative at this point is to open the valves to the MVSS and transfer residual heat from the reactor core to the atmosphere via the suppression pool and the MVSS as described in the introduction to this chapter.

The second scenario for Oskarshamn 1 assumes both the emergency condenser and the air-cooled EDGs to fail. In this scenario, low water level in the reactor and a flow of less than 10 kg/s in the feedwater system will occur simultaneously. When this happen, the emergency feedwater system, which in this case is controlled by the diversified protection system and powered by the GTs, will restore and maintain reactor water level. The emergency feedwater system is automatically powered from the GTs when the EDGs are lost, and provided with water from the demineralised water supply. However, because the emergency condenser is assumed to fail, reactor steam is assumed to be transferred to the suppression pool by the safety relief valves. The suppression pool temperature, which is controlled by the sea-water cooling system, will thereby increase and eventually start to boil. The suppression pool cooling system is powered by the sea-water cooled EDGs and will therefore also be lost in this scenario. Failed cooling of the suppression pool will lead an increase in containment pressure. The only available alternative at this point is to open the valves to the MVSS and transfer residual heat from the reactor core to the atmosphere via the suppression pool and the MVSS as described in the introduction to this chapter.

In case of loss of the primary ultimate heat sink and the alternate heat sink, the scenario for Oskarshamn 2 will be similar to the event of loss of primary ultimate heat sink, except for the postulation that the alternate suppression pool cooling system initially fails. In this scenario, the water level in the reactor pressure vessel is assumed to be maintained by the emergency feedwater system which is powered from the EDGs or the GTs. The emergency feedwater system is provided with water from the demineralised water storage tank. The volume in this storage tank is enough to maintain the reactor level and enable the reactor to be taken to a safe state. However, there will be no cooling of the suppression pool because it assumed that the alternate suppression pool cooling system has failed. This means that when steam is dumped to the suppression pool from the Safety Relief Valves, the temperature in the suppression pool will continue to rise and eventually start to boil. The only available alternative at this point is to open the valves to the MVSS and transfer residual heat from the reactor core to the atmosphere via the suppression pool and the MVSS as described in the introduction to this chapter.

In case of loss of the primary ultimate heat sink and the alternate heat sink, the scenario for Oskarshamn 3 will be equivalent to the scenario described in subsection 5.B.2.3.

As mentioned above, the GTs starts at loss of off-site. The GTs will provide power to the demineralised water plant, the fresh-water plant and the fresh-water pumps from fresh-water reservoir at the site.

There has been no cliff edge effect identified for Oskarshamn 1 for cases where the GTs are operational, because core cooling will provided by emergency feedwater system. The core cooling capacity of emergency feedwater system is such that the Oskarshamn 1 could be shut down and maintained in a safe state for unlimited time. However, the GTs have limited fuel storages at the site, which will be a limiting factor for this scenario as described in the subsection 5.B.1.2.

Specific analyses have been performed for Oskarshamn 1 and Oskarshamn 2 , to study the case where no manual actions are performed. Two different scenarios have been considered. The first case assumes that the demineralised water storage tank contains 700 m^3 of water and no actions are performed. The second case assumes automatic refilling of the demineralised water storage tank. Results show that for Oskarshamn 1, the temperature in the suppression pool will reach 95°C after approximately 10 hours, and the rector water level can be maintained for almost 50 hours without any manual actions. For Oskarshamn 2, results show that the reactor water level can be maintained for more than 30 hours without any manual actions. If refilling of the demineralised water storage tank is assumed, core cooling at both units will be available for unlimited time. However, the GTs which are assumed to be available in these scenarios have limited fuel storages at the site, which will be a limiting factor for the scenario as described in subsection 5.B.1.2.

If the event affects all units at the same time, all units will require cooling water for long term conditions. Cooling water could initially be produced in the demineralised water plant. However, in the long term condition, the water quality will not fulfil the specifications for demineralised water. If the demineralised water plant has failed, the drinking water reservoir and, the site fresh-water reservoir have sufficient capacity to prevent any damage to the fuel in the reactor core and to the spent fuel in the fuel pools. Engineering judgment indicates that the lower quality of the water will not affect the safe shut down of the units and the ability to maintain them in a safe shut down condition.

The Spent Fuel pool

For all units, loss of the primary ultimate heat sink and loss of the alternate heat sink, will lead to a scenarios equivalent to the ones described in subsection 5.B.2.3.

5.B.2.3.1 Results from the licensee assessments of protection against loss of ultimate heat sink and loss of alternate heat sink

For Oskarshamn 1 and Oskarshamn 2 , there will be no damage to the fuel as long as the emergency feedwater system is operating. For Oskarshamn 3 the scenario will be equivalent to the scenario described in subsection 5.B.2.3.

Residual heat is removed from the suppression pool and transferred to the MVSS. Suppression pool level will be maintained by the independent containment spray system and the MVSS will be cooled by the external cooling system. The residual heat removal to the atmosphere via MVSS is passive and can go on for a long time without any radioactive release.

No releases of activity to the atmosphere are expected in analysed scenarios.

5.B.2.3.2 Measures which can be envisaged to increase robustness of the plants in case of loss of ultimate heat sink and loss of alternate heat sink

For Oskarshamn 1, the success of maintaining the level measurement in the reactor pressure vessel in long term conditions depends on the availability of the GTs. The instrumentation of the reactor pressure vessel is maintained as long as either the GTs can power Oskarshamn 1 or the battery backed-up power supply systems are available. By introducing alternate power supply functions to the GTs the reliability of the level measurements could be strengthen.

5.B.3 Loss of the primary ultimate heat sink, combined with station black out

For Oskarshamn 1 all EDGs are assumed to be lost. An automatic function will connect the GTs to Oskarshamn 1. The scenario will then be equivalent to the scenario described in subsection 5.B.2.3, and no fuel damage in the reactor core is expected to occur. If the GTs are assumed to fail the scenario will be equivalent to the scenario described in the subsection 5.B.1.3.

For Oskarshamn 2 the EDGs will fail. If GTs is assumed to be operating the scenario will be equivalent to the scenario described subsection 5.B.2.3. In the latter case, all core cooling systems and residual heat removal systems are assumed to fail, due to loss of the power. Damage to the fuel is calculated to begin after around 2 hours. The rupture disc to the MVSS is expected to open after around 24 hours.

For Oskarshamn 3 all EDGs are assumed to be lost. The GTs will manually be connected to Oskarshamn 3 after approximately 1 hour. The scenario will be equivalent to the scenario described in subsection 5.B.2.3, and no fuel damage in the reactor core is expected to occur. If the GTs are assumed to fail the scenario will be equivalent to the scenario described in subsection 5.B.1.2.

If an event affects all units simultaneously, and the GTs are assumed to be unavailable, fuel in the reactor cores are assumed to be unavailable within a few hours. However, even when the reactor core is damaged, the severe accident mitigation systems will ensure the effect on the environment is small.

The spent fuel pool

The scenario will be similar to the scenario described in subsection 5.B.2.3, for Oskarshamn 1 and Oskarshamn 2 and in the loss of the primary ultimate heat sink and the alternate heat sink for Oskarshamn 3.

For these scenarios, the spent fuel pool water temperatures will increase and eventually water in the pools will start to boil. Make-up water will be provided by fire water from either the direct driven fire-water pumps or by the fire truck. Because no power is available, the so call "bleeding" will eventually lead to overfilling of the radwaste system tanks that will cause water to be released to the environment (to the site and/or to the Baltic Sea). However, the activity release directly to the Baltic Sea in these circumstances is estimated to be equivalent to the discharge activity during one year of normal operation for the units.

The fire-water reservoir during power operation is expected to last for about 90 days for Oskarshamn 1, 50 days for Oskarshamn 2, and 18 days for Oskarshamn 3. If the reservoir at Oskarshamn 3 is available for Oskarshamn 1 and Oskarshamn 2 , or vice-versa, water will last twice as long. During refuling the fire-water reservoir will last about 10-13 days for Oskarshamn 1, 6-9 days for Oskarshamn 2 and 18-66 days for Oskarshamn 3, depending on when the event occurs.

During these events, it is reasonable to assume that there might be external assistance available that can provide temporary pumps for supplying the fire-water system with water from the fresh-water storage facilities. Also, when all three units are affected, the inventory of fresh water is sufficient for all three units. If the situation occurs during refueling outages, the time frame for manual actions is enough to handle all three units at the same time.

At power operation fuel damage in the spent fuel pool during these conditions are calculated to occur after about 46 days for Oskarshamn 1, 35 days for Oskarshamn 2 and 13 days for Oskarshamn 3. If the unit is in a refueling outage the time to fuel damage is calculated to occur after about 5 days for Oskarshamn 1, 6 days for Oskarshamn 2 and 3 days for Oskarshamn 3.

If boiling in the spent fuel pools progresses without any make-up water available, the fuel assemblies will be uncovered, and hydrogen will soon start to form. The analysis shows that the released hydrogen will be accumulated in the refueling floor area and that manual actions will needed. There are 12 openings at Oskarshamn 1and 10 openings at Oskarshamn 2 to the atmosphere from the buildings and these are all accessible and can be manually opened. By opening these relief paths, hydrogen accumulated in the building could be vented by natural draft. For Oskarshamn 3 several of doors are expected to be opened in these situations and ventilation by natural draft will be accomplished before fuel in the spent fuel pools is uncovered.

Make-up water from the fire-water system can manually be provided to the spent fuel pool. However, to avoid difficulties due to heavy moisture in the room, hoses must be manually rolled out before the temperature in the spent fuel pools reaches approximately 80°C. In power operation, available time before the spent fuel pool water temperatures reaches 80°C is expected to be around 6 days for Oskarshamn 1, 5 days for Oskarshamn 2 and 2 days for Oskarshamn 3, if the event occurs right after start up. If the event occurs right before end of cycle (shut down), available time is expected to be around 17 days for Oskarshamn 1, 13 days for Oskarshamn 2 and 7 days for Oskarshamn 3. In a refueling outage and at the most limiting conditions there will be approximately 24 hours for Oskarshamn 1, 18 hours for Oskarshamn 2 and 8 hours for Oskarshamn 3 before the spent fuel pool water temperatures reaches 80°C. Additionally, during this time frame, valves to the radwaste tanks must manually be opened, to prevent radioactive water from the spent fuel pools to enter and accumulate in the lower part of the building.

5.B.3.1 Time of autonomy of the site before loss of normal cooling condition of the reactor core and spent fuel pool (e.g., start of water loss from the primary circuit).

For Oskarshamn 1, the analyses show the following:

- If the GTs are assumed to be available, there will be no fuel damage in the reactor core.
- If GTs are assumed to fail, the water reservoir in the emergency condenser will be enough to perform residual heat removal as long as the battery backed-up power supply systems are available. Damage to the fuel in the reactor core is however calculated to begin within 3 hours.
- If the GTs and the emergency condenser are assumed to fail, fuel damage in the reactor core is calculated to occur within 50 minutes.
- If the GTs are available, cooling of the spent fuel pool can be maintain for at least 90 days during power operation and for at least 10 days during refueling.
- If boiling in the spent fuel pool will continue without any make-up water available, fuel damage in the spent fuel pool is calculated to occur after 46 days during power operation and after 5 days during refueling.

For Oskarshamn 2, the analyses show the following:

- If all electrical power supplies are assumed to fail, fuel damage in the reactor core is calculated to occur after around 2 hours.
- Cooling of the spent fuel pool can be maintain for at least 50 days during power operation and for at least 6 days during refueling.
- If boiling in the spent fuel pools will continue without any make-up water available, fuel damage in the spent fuel pool is calculated to occur after 35 days during power operation and after 6 days during refueling.

For Oskarshamn 3, the analyses show the following:

- If the GTs are assumed to be available and successfully manually connected to Oskarshamn 3, there will be no fuel damage.

- If GTs are assumed to fail. Damage to the fuel in the reactor core is calculated to begin after about 1 hour.
- Cooling of the spent fuel pool can be maintain for at least 18 days during power operation and for at least 3 days during refueling.
- If boiling in the spent fuel pools will continue without any make-up water available, fuel damage in the spent fuel pool is calculated to occur after 13 days during power operation and after 3 days during refueling.

5.B.3.2 External actions foreseen to prevent fuel degradation

There have been no external actions identified to prevent fuel degradation in the reactors.

For the Spent Fuel, arrangements from the fire brigade at site or from outside of the plant must be in place within 6 days to supply the fire-water tanks with fresh water from the fresh-water facilities.

5.B.3.3 Measures, which can be envisaged to increase robustness of the plants in case of loss of primary ultimate heat sink, combined with station black out

Batteries are designed for 2 hours of operation, which means that the availability of the emergency condenser at Oskarshamn 1 is limited. Increasing the capacity of the batteries to certain important loads during long term operation, (e g. by disconnections of less important loads), would increase the robustness of the plats.

The manual operation of the emergency condenser system could be improved to make it possible to keep the emergency condenser in operation during station black out. Alternatively, redesigning the system so that total loss of power would cause at least one train of the emergency condenser system to operate without any manual actions being required, would improve the safety of Oskarshamn 1.

Implementing emergency procedures to provide make-up water to the spent fuel pools during loss of all electrical power sources would increase the robustness of the units.

5.C. Ringhals

5.C.1 Loss of electrical power

The power supply systems

The Ringhals site is connected to the external 400 kV grid and the external 130 kV grid. All units have both a 400 kV and a 130 kV transmission system interface. There is also a back-up gas turbine plant common for the site and two mobile units. Additionally, the fire-water systems are powered from separate diesel driven pumps.

If loss of off-site power occurs, power will be supplied by dedicated emergency diesel generators (EDG) or by the two dedicated gas turbines (GT) common for the site. All units have individually four redundant water cooled EDGs. Ringhals 1 has additionally two air-cooled EDGs installed which will supply the diversified plant section (DPS).

All units can be supplied with power from the main generators during normal operation and also from the external grid through the main transformers when the generators are not available.

In case of disturbance on the off-site external power grid it may be possible to supply the unit power supply systems from the main generators, i.e. house load operation. 12 hours of house load operation is required but operational experience from Ringhals demonstrates only a successful transition to house-load operation with stable operations for more than 6 hours.

If needed the ordinary power supply source can be switched from the normal supply path to an alternate supply path. At Ringhals 1 this connection requires manual operations. For the other units this is a fully automatic feature, but a permissive to activate this is that the generator breaker fails, otherwise a manual operation is required.

An additional full capacity mobile diesel generator is available at the site.

The power supply to the filtered vent systems has independent battery capacity and a dedicated mobile generator set.

All units were originally designed with two main divisions of safety functions. The supporting power system was divided into four redundant divisions, each one with a dedicated EDG. Two divisions are supporting each main division. The four redundant divisions may in some cases perform the intended functions by different methods such as steam driven and motor driven pumps. Also, the two main divisions have during the years been reinforced. Measures to strengthen the physical and functional separation of the trains have been taken. Additional, diversified reactor trip systems have been installed in all units.

At Ringhals 1 an entire diverse plant section (DPS) has been installed in addition to the original plant section (OPS). OPS and DPS are each sufficient to maintain all safety functions. DPS does however, not include cooling of the spent fuel pools.

All units are equipped with steam driven systems to provide core cooling capabilities, either directly to the reactor pressure vessel (BWR) or via the steam generators (PWR).

The normal power supplies for all units are 6 kV.

<u>The ordinary back-up AC power supply system</u>

The ordinary back-up AC power supply systems for all units are supplied by separate EDGs. All units have individually four redundant water cooled EDGs. Ringhals 1 has additionally two air-cooled EDGs dedicated for the diversified plant section.

All EDGs have alarms indicating low lubricating oil pressure and low level in both the day tanks and storage tanks for fuel to ensure that operators can take measures to refill them before they are depleted.

There is also a mobile diesel generator available at Ringhals, which can replace any of the ordinary EDGs.

<u>The alternate back-up AC power supply system</u>

If loss of off-site power occurs, and the EDGs fails, the two dedicated gas turbines GTs located in the vicinity of the site, will provide power to the site power supply systems.

In case of loss of off-site power, an automatic function, which will be manually initiated from the Ringhals 1 main control room (or locally at the GTs), disconnects all other consumers and connects a link between the GTs and the site. The automatic function also gives a starting signal to the GTs.

Rated power for one GT unit is greater than the rated power of all EDGs together.

The Ringhals 1 DPS is designed to cope with events leading to unavailability of the Ringhals 1 OPS. The DPS contain a reactor protection system, which is independent of the original reactor protection system. The DPS is designed to initiate safety functions for reactivity control, pressure relief, emergency core cooling (including small line breaks) and residual heat removal, based on independent and dedicated input signals. Power supply in case of loss of off-site power for the DPS consists of two dedicated air-cooled EDGs and a dedicated battery back-up power supply. The DPS is in many respects built diversified to the OPS, with extra care for, earthquake, lightning, air cooled emergency diesel generators and so on and is assumed to be operable in a case where ordinary on- site back-up power sources are lost.

The battery backed-up power supply

The batter back-up power supply systems are divided into redundant physically separated divisions in accordance with the safety classed power supply systems design basis.

Batteries are designed to maintain between 1-4 hours of operation without charging.

The accident mitigation system for filtered containment pressure release

The power supply systems for the MVSS are supplied by the safety classed power supply systems for the units or by the EDGs. The battery backed-up power supply for the MVSS is designed for an 8 hour condition.

For all units, there are also prepared connections for external power generators to enable long term operation of the MVSS. There are two mobile units on-site that can be used for all units. Connecting the mobile unit takes normally approximately 1-2 hours. The mobile diesel generator systems can run about 1 day at full power, in case the fuel tank is filled at the start. Display and alarm for low level of fuel or lubricating oil takes place locally. Group alarms are also obtained at a local panel for the MVSS.

For Ringhals 1 the DPS can also supply the MVSS with power.

Valves to the MVSS can be open manually by operators, or automatically by a rupture disc when the containment pressure has increased to the rupture disc limit which has been set to a level that will give a reasonable margin to the containment design pressure.

5.C.1.1 Loss of off-site power

At loss of off-site power the external grids are lost and only the EDGs and the GTs will be available.

Loss of the off-site power is included in the design basis for the units and presented in the safety analysis report, either as an initial event or as a secondary event after other initial events (if it is conservative in the analysis to assume that the off-site power is lost).

The EDG fuel storages

For Ringhals 1and Ringhals 2 each EDG has its own day tank. These day tanks can be supplied with diesel oil from units' diesel storage tank. Ringhals 1and Ringhals 2 also have an additional common diesel storage tank. Ringhals 1 does not have any verifying documentation that indicates how long time the diesel oil lasts in a loss of off-site power situation. However, engineering judgments indicates that Ringhals 1 storage capacity will last longer than the Ringhals 2. For Ringhals 2, the diesel storage is expected to last for about 4-5 days. If possible measures to reduce the fuel consumption are performed, the estimation is that it might be possible to reach about 7 days of operation with the fuel stored in Ringhals 1 respectively Ringhals 2.

For Ringhals 3 and Ringhals 4, each EDG has its own day tank. These day tanks can be supplied with diesel oil from the diesel storage tanks. Each unit has its own storage tank. It is also possible to line-up the system so that it is possible to transfer diesel oil between the units' storage tanks. An analysis indicates that the amount of diesel fuel makes it possible to keep the EDG in operation for more than 7 days.

The operation of the EDGs can also be prolonged by for example manually disconnecting loads, pumping fuel between units, and/or transportation of fuel between units using mobile tanks. This transport requires a forklift or similar equipment which should be available at the site if needed.

The supplier would also be able to deliver approved diesel fuel to the site within 24 hours under the assumption that roads can be used. Delivery can be made from the main supplier in Gothenburg, Helsingborg or Lysekil. There are also several local suppliers who can deliver fuel to the Ringhals site if needed.

<u>The lubricating oil supply</u>

Ringhals 1 does not have any explicit verifying documentation regarding the EDGs consumption of lubricating oil. However, the lubricating oil supply for Ringhals 1and Ringhals 2 at the site is estimated to be sufficient for more than 7 days of operation of all EDGs at both units. Calculations indicate also, that the lubricating oil storage should be sufficient for more than 11 days of EDGs operation if only the Ringhals 2 EDGs are assumed to be operating. If lubricating oil in the oil sump is included and operation below rated power is considered, calculations indicates that the on-site lubricating oil storage should be enough to enable more than 7 days of operation for all EDGs at both units.

Ringhals 3 and Ringhals 4 are using a different type of lubricating oil than the other two units at the site. The volume of lubricating oil in the EDG oil sumps for Ringhals 3 and Ringhals 4, is expected to be sufficient for more than 7 days of operation.

5.C.1.1.1 Results from the licensee assessments of protection against loss of off-site power

As long as the EDGs are available, no fuel damage in the reactor cores or in the spent fuel pools will occur.

All units are judged to have a sufficient amount of consumables (lubricating oil and fuel) stored on-site to maintain the EDGs in operation until the infrastructure are assumed to be restored and heavy transports can reach the site.

5.C.1.1.2 Measures which can be envisaged to increase robustness of the plants in case of loss of loss of off-site power

The EDGs for Ringhals 2 have the lowest endurance, and by increasing the unit's diesel fuel storage, increased robustness could be achieved.

Ringhals 1 does not have any verifying documentation that indicates for how long time, the available diesel oil and the available lubricating oil, will last. By establishing verifying documents the reliability of the EDGs in these situations should be improved.

5.C.1.2 Loss of off-site power and loss of the ordinary back-up AC power source

At loss of off-site power and loss of on-site back-up power sources, power from the external grids is assumed to be lost and the EDGs are assumed to fail. The GTs will however, still be available and there will be no further consequences for fuel in the reactor core or in the spent fuel pools.

<u>The alternate back-up AC power supply systems</u>

According to plant requirements, the GTs shall always have sufficient fuel for at least 7 days of operation. Additionally, an engineering judgement indicates that the fuel supply for the

GTs is sufficient for several weeks of operation if one GT is in operation. The GTs does not consume much lubricating oil, according to operating experience, and the lubrication oil is not expected to be a limiting factor. The battery backed-up power supply system for the GTs is assumed to be available for more than 20 hours.

The Ringhals 1 the day tanks for the air-cooled EDGs (designated for the DPS) are designed for operation during at least 7 days without being refilled from any external source. The lubricating oil in the oil sump will approximately last for 6 days of operation and the lubricating oil in the on-site storage will approximately last for 12 days of operation.

A mobile diesel generator is available at the site to replace any EDG, in case of loss of all ordinary on-site back-up power sources. In a scenario where all units are affected, it would be beneficial if the mobile emergency diesel generator remains in its normal parking space, where it should be able to supply power to both Ringhals 1and Ringhals 2 .

5.C.1.2.1 Results from the licensee assessments of protection against loss of off-site power and loss of the ordinary back-up ac power source

As long as the GTs are available, no fuel damage in the reactor cores or in the spent fuel pools will occur.

5.C.1.2.2 Measures which can be envisaged to increase robustness of the plants in case of loss of off-site power and loss of the ordinary back-up ac power source

There is an on-going project at Ringhals to install two new stationary EDGs (one for Ringhals 1and Ringhals 2, and one for Ringhals 3 and Ringhals 4). The new stationary EDGs shall independently of the existing EDGs be connected to maintain the operability during modernization of the existent EDGs, and also during future maintenance. The new EDGs will be of a different design (modern technology) from the existing EDGs and will be possible to connect to each of the existing safety power supply system. When this has been implemented it is assumed to improve the reliability of the site emergency power supply and also the units' protection against common cause failures (CCF).

5.C.1.3 Loss of off-site power and loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power sources

At loss of off-site power, loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power sources, two scenarios have been evaluated. The first one assumes that mobile equipment will be available. The second scenario assumes that only the battery backed-up power supply systems will be available.

All units are equipped with steam driven systems, which can be monitored as long as the batteries are functioning.

At the site there are two mobile units for water and power supply.

The Ringhals 2, Ringhals 3 and Ringhals 4, are all PWR and the considered scenarios will be equivalent for all three units.

The battery backed-up power supply

The design basis for the battery back-up power supply at Ringhals 1and Ringhals 2 is according to the safety analysis report, 2 hours, with the exception from the uninterruptable power supply system where 3.5 – 4 hours have been applied. The design basis for the battery backed-up power supply for the DPS at Ringhals 1 is according to the safety analysis report 4 hours.

The design basis for the battery backed-up power supply at Ringhals 2 is according to the safety analysis report, 2 hours, with the exception from the batteries supplying the trip circuit supply system where 4 hours have been applied.

The design basis for the battery backed-up power supply at Ringhals 3 and Ringhals 4 is according to the safety analysis report, 1 hour. For Ringhals 3 and Ringhals 4 there is also one common mobile battery for each voltage level (normally used for maintenance purposes and discharge tests) that can be used to obtain an extended battery life time.

Evaluations of battery capacities at all units according to existing supporting documents for Ringhals 1and Ringhals 2 and engineering judgments for Ringhals 3 and Ringhals 4, indicates that the battery capacities for all units are greater than the specified design basis according to the safety analysis reports.

Ringhals 1 has one steam-driven auxiliary feedwater pump. The steam-driven auxiliary feedwater pump can give a water flow corresponding to the steam that is transferred to the suppression pool, by the safety relief valves. The steam-driven auxiliary feedwater pump is dependent of control valves, which can operate as long as battery power is available. Engineering judgements indicates that the battery backed-up power supply for the steam-driven auxiliary feedwater pump will be available for 12 hours. Therefore, in the analysis of loss of off-site power, loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power, 12 hours of operation for the steam-driven auxiliary feedwater pump has been assumed.

Analysis

Scenario1:

a) For Ringhals 1, the mobile unit is assumed to be connected. No detail analysis is found for this scenario. However, the scenario is assumed to initially be equivalent to scenario 2 and it is likely that the mobile unit would improve the time before damage to fuel becomes unavoidable.
b) For Ringhals 2, Ringhals 3 and Ringhals 4, the mobile unit is assumed to be connected for each unit respectively and will enables relevant process information to be obtained. Additionally, by energising certain control equipment the information required to use the operating procedures for accidents would be made available to the operator. It will also provide the possibility to manoeuvre certain pneumatic control valves, like the relief valves in the pressurizer. One steam generator and a steam-driven auxiliary feedwater pump will be able to remove residual heat from the reactor core to the atmosphere.

- The residual heat in the core will decrease over time, and the required feedwater flow to the steam generator, will also decrease. After approximately 24 hours of operation with a feedwater flow-rate of about 6-8 kg/s is sufficient for removing the residual heat. Feedwater for the steam generator would be supplied by the fresh-water supply system.
- Based on the assumptions made in the analysis, it will take about 70 hours before fuel damage may occur.

Scenario 2:

a) For Ringhals 1, the steam-driven auxiliary feedwater pump is assumed to be operating for 12 hours. After 12 hours, and when the steam-driven auxiliary feedwater pump has failed, water level in the reactor pressure vessel will start to decrease since there is no make-up water available. For the most limiting case the fuel in the reactor pressure vessel starts to uncover, and fuel damaged becomes unavoidable after approximately 16 hours.

b) For Ringhals 2, Ringhals 3 and Ringhals 4, the steam-driven auxiliary feedwater pump is assumed to stop when the battery backed-up power supply is lost, and the possibility to remove the residual heat from the reactor core to atmosphere with the steam generators, will be lost. For the most limiting case fuel damaged becomes unavoidable after approximately 9 hours.

The spent fuel pool

The spent fuel pools have no prepared connection for the mobile units. Thus, it has been assumed that only batteries will be available for the spent fuel pool during these scenarios. (Hydrogen related issues will be discussed in chapter 6 to this report.)

When only battery power remains the fire-water system could manually provide water to the spent fuel pools. The time needed to get the fire-water system operating is estimated to be a few hours. Analysis shows that:

- For Ringhals 1, fuel damage becomes unavoidable after 2 weeks if the event occurs during refueling and after 4 weeks if the event occurs during normal operation.
- For Ringhals 2, Ringhals 3 and Ringhals 4, fuel damage becomes unavoidable after 1 week if the event occurs during refueling and after 5 weeks if the event occurs during normal operation.
- For an event that affects all units at the site, fuel damage becomes unavoidable after 5 days if the event occurs during refueling.

If the fire-water systems are not available the time until top of fuel (if only one reactor is affected) is about 3-5 days during refueling and about 13-26 days during normal operation.

Improvised functions, such as providing make-up water to the spent fuel pools by fire trucks and/or by the mobile unit, will most likely take several hours to complete.

For the case when all reactors are affected, calculations have only been made for refueling mode which is the most limiting case. Ringhals 3 has the highest thermal power and thus will be the limiting reactor. Since all reactors are equally affected the fire-water volumes in fresh-water supply system will be shared between the units. Time until top of fuel is uncovered when all units are affected, is approximately 5 days.

5.C.1.3.1 Results from the licensee assessments of protection against loss of off-site power, loss of the ordinary back-up ac power sources, and loss of permanently installed diverse back-up ac power sources

If the mobile unit is assumed to be connected:

a) For Ringhals 1, no detail analysis is found for this scenario. However, the scenario is assumed to initially be equivalent to scenario 2 and it is likely that the mobile unit would improve the time before damage to fuel becomes unavoidable.
b) For Ringhals 2, Ringhals 3 and Ringhals 4, it will take approximately 70 hours before fuel damage may occur.

If the mobile unit is assumed to be unavailable:

a) For Ringhals 1, fuel damaged becomes unavoidable after approximately 16 hours
b) For Ringhals 2, Ringhals 3 and Ringhals 4, fuel damaged becomes unavoidable after approximately 9 hours.

The spent fuel pool

If fire water is available, fuel damage becomes unavoidable at Ringhals 1 after 2 weeks during refueling and after 4 weeks during normal operation, and at Ringhals 2, Ringhals 3 and Ringhals 4, after 1 week during refueling and after 5 during normal operation

If the fire-water systems are not available the time until top of fuel (if only one reactor is affected) is about 3-5 days for refueling and about 13-26 days for normal operation.

For the case when all reactors are affected time until top of fuel is uncovered is about 5 days.

5.C.1.3.2 Measures which can be envisaged to increase robustness of the plants in case of loss of off-site power, loss of the ordinary back-up ac power sources, and loss of permanently installed diverse back-up ac power sources

The battery backed-up power supplies capacity, is according to the safety analysis report, very limited. Extending the capacity of the battery backed-up power supplies to certain important loads during long term operation, e g. by disconnections of less important loads, would increase the availability time for the battery backed-up power supplies. Estimations indicate that for the standard battery back-up power supplies could availability time be extended to 6 hours.

Additionally, some batteries are used for maintenance purposes (and will be fully charged most of the time and will not discharged at an event). According to procedures it should be possible to connect these batteries and increase the battery capacity for the steam driven auxiliary feedwater pump.

It is possible to interconnect Ringhals 1 with Ringhals 2, and Ringhals 3 with Ringhals 4. By doing so, the consequences for an event that only affects one unit should be limited. Time necessary to have systems in operations is estimated to less than 12 hours.

It might be possible for Ringhals 3 to supply all other units at the site if the main generator(s) in Ringhals 3 are still in house load operation. Time necessary to have systems in operations is estimated to less than 12 hours.

Fire trucks should be likely to arrive from the neighbouring town and the surrounding area, as soon as the site will be available from the road. In addition, it might be possible for the fire brigade to request external mobile pump units.

The GTs, dedicated to Ringhals, are not the only GTs in the area. It is possible that other GTs could be used (with more or less manual efforts) to supply power to the Ringhals site. Time necessary to have systems in operations is estimated to be less than 24 hours.

Located on Ringhals is an additional mobile diesel generator which may be a possible power source when all normal alternate power sources are unavailable. The mobile unit is however normally used as a mobile unit to supply the emergency response centre with electrical power. Time necessary to have systems in operations is estimated to be less than 12 hours.

As only two mobile units of water and power supply exist at the site. By introducing additional mobile units it might be possible to increase the reliability of the mobile units.

A number of operating procedures need to be updated, preferably those relating to energizing the power system with discharged batteries. Operating procedures are available for scenarios when power returns. In this case, when an extended loss of off-site power scenario is considered, there are no operating procedures present.

To increase the possibilities for the emergency response organization to make decisions, in order to protect the steam-driven auxiliary feedwater pump at Ringhals 2, Ringhals 3 and Ringhals 4, the emergency response procedures should be reviewed.

Improving operating procedure for re-establishment of power supply assuming discharged batteries could possibly increase the safety of the site.

The Spent fuel pools

The following list describes possible provisions to prevent some of the identified cliff-edge effects for the spent fuel pools:

- Look into possibilities of introducing a way to cool and feed the spent fuel pools from outside the building.
- Verify the pool integrity for boiling conditions.
- Review the level and temperature instrumentation.
- Develop strategies to priory different actions when several reactors and the spent fuel pools are affected at the same time
- Develop strategies and procedures to reach a stable safe state.
- Evaluate the risk of criticality for Ringhals 2, Ringhals 3 and Ringhals 4, during boron dilution under boiling conditions.
- Look into ways of venting steam from the building.
- Review possible ways of protecting the spent fuel pools buildings from effects of severe fuel damage in the spent fuel pools, e.g. hydrogen explosions.

5.C.2 Loss of the ultimate heat sink

The Ringhals site is situated on a peninsula in the sea area Kattegatt. The primary ultimate heat sink for all units at Ringhals is sea water. With regard to the safety properties of the primary ultimate heat sink, there are no significant differences between the units at Ringhals.

The Ringhals 2, Ringhals 3 and Ringhals 4 (PWR), also have the option to release residual heat to the atmosphere through the steam generators. However, this procedure is dependent on the water sources available for the auxiliary feedwater system, and is thus, limited.

The Ringhals 1 (BWR) has no alternate ultimate heat sink.

5.C.2.1 Design provisions to prevent the loss of the primary ultimate heat sink

The sea-water cooling systems

Ringhals 1and Ringhals 2 respectively Ringhals 3 and Ringhals 4, have one common cooling channel respectively and one common outlet for the cooling water respectively. Each channel is connected to two intake buildings, one for main cooling water and one for auxiliary cooling water. Both intakes are equipped with screening equipment for removal of debris from the sea water. The inlet channels are also equipped with an outer inlet barrier at the entrance of the channel, to prevent floating objects, such as timber, oil and ice, to enter the channel. The outlet tunnels are connected to surge basins to prevent water hammer when main cooling pumps are started and stopped.

The cooling water system is designed with several sluice gates to allow different alignments in order to provide cooling water to the reactor buildings during all operation conditions (normal operation, maintenance, testing and postulated accidents). For example, if clogging of both intakes occurs, it is possible to provide auxiliary cooling water from the outlet water tunnel to the sea-water cooling systems. If additionally the outlet is clogged, it will be possible to re-circulate the cooling water in order to obtain cooling water to the sea-water cooling systems.

The main sea-water cooling system is separated into two trains. One of these trains can be used as a back-up for the auxiliary cooling water.

At normal operation, the sea-water cooling flow is approximately 80 m^3/s. In case of loss of off-site power for all units the auxiliary cooling water flow is approximately 7 m^3/s.

The intake building and its structures are designed so that the lowest recorded water level would not disturb the system functions. There is also a central alarm on the level in the sea-water cooling system sumps and there are instructions for handling blockage and low level in the inlet channel.

Clogging of the intake could be caused by chemical pollution in the sea or by natural occurrences (e.g. ice and biological material). The intake systems are protected against clogging in the following way:

- Recirculation of cooling water to melt ice and keep intake cooling water temperature above freezing is possible.
- Bigger materials e.g. floating objects, ice, are stopped by an outer inlet barrier that is located at the entrance of the water channel and intended to hold back most of the material from entering the inlet. The outer inlet barrier is built as a bridge, with a wall reaching down to a few meters below the water surface.
- Biological material like jellyfish, is separated from the water by screening equipment. It is also possible to inject chlorine in the cooling water to minimize the growth of biological material (e.g. clams and algae) on the screening equipment and in the channels and tunnels.
- Oil barriers are available and can be used if a warning of oil spill has been issued.

If the inlet channel should be totally blocked by biological material there is a possibility to recirculate the cooling water through gates in order to provide cooling water to the sea-water cooling system. Transmitters indicate the water level difference over the screening equipment. If the water level difference rises above the limit value, gate valves open automatically to recirculate the discharge water back to the suction side. If the blockage is located in the cooling water intake, the outlet tunnel can be used as an intake of cooling water in combination with the redundant outlet tunnel from the sea-water cooling system. The re-circulation is automatically activated and the redundant outlet tunnel is manually activated. However, the outlet tunnel has no equipment to separate the debris from the sea water. This means that there will be an increased possibility for clogging of the sea-water cooling system when water is taken from the outlet tunnel during re-circulation mode.

If the outlet is blocked the cooling water can be led back for recirculation, either directly to the intake or through the inlet channel.

With consideration to possible collapse of the intake tunnel systems, the auxiliary cooling water has been lined with reinforced concrete.

The fresh-water supply

The fresh-water storage is common for Ringhals 1and Ringhals 2 consists of a number of dedicated storage volumes for different applications. These are:

- a volume for compensation of 250 m3,
- back-up storage of 10750 m3,
- fire-water storage of 2000 m3,
- storage for EDG cooling water of 2000 m3, and

- a volume of 5000 m3 dedicated for either flooding of the reactor containment at Ringhals 1, and/or for the auxiliary feedwater supply.

The total minimum volume at is 9250 m^3 and the total maximum volume is 20000 m^3.

The fresh-water storage is common for Ringhals 3 and Ringhals 4 consists of a number of dedicated storage volumes for different applications. These are:

- industrial water storage of 3000 m3,
- fire-water storage of 1500 m3,
- storage for EDG cooling water of 4000 m3, and
- auxiliary feedwater storage for both Ringhals 3 and Ringhals 4, of 3000 m3

The total minimum volume is always assumed to be 11500 m^3.

Both fresh-water supplies receive water from the closes town. However, in the analysis refilling will not be credited.

The liquid waste processing storage

The liquid waste processing system at Ringhals 1 has two storage reservoirs. One storage tank of 2500 m^3 is dedicated for the Safety injection system and one storage tank of 430 m^3 (minimum allowable volume of 300 m^3) is dedicated for the auxiliary feedwater system.

The demineralised water storage

The system for storage and distribution of demineralised water at Ringhals 1 has storage of maximum 3000 m^3 (and minimum 800 m^3) where 800 m^3 is dedicated for the auxiliary feedwater system and the emergency feedwater system. This tank can also supply Ringhals 2.

The system for storage and distribution of demineralised water at Ringhals 3 and Ringhals 4 has storage tank at Ringhals 3 of maximum 1420 m^3 (and minimum 1000 m^3). Additionally there is a buffer tank at Ringhals 4 tank of maximum 475 m^3 (and minimum 450 m^3)

The condensate storage tanks

The condensate storage tanks for Ringhals 2 have a volume of 620 m^3 each (minimum 440 m^3). The condensate storage tanks for Ringhals 3 have a volume of 1000 m^3 each (minimum 650 m^3). The condensate storage tanks for Ringhals 4 have a volume of 1000 m^3 each (minimum 650 m^3).

The EDG cooling systems

The EDGs for all units are water-cooled, except the air-cooled EDGs dedicated for the DPS in Ringhals 1. The air-cooled EDGs are further discussed in subsection 5.C.1.2.

The cooling water is transferred from the fresh-water supply to the EDGs through two different pipes by static pressure. The direct driven centrifugal pumps belonging to the EDGs are designed to provide the EDGs with sufficient cooling water flow even at a low level in the fresh-water supply.

The cooling systems for the EDGs are normally lined up to the fresh-water supply. If a shortage of water in the fresh-water supply should occur or if the EDGs need to be in operation for a long time, it will be necessary to manually line up the cooling system to the Sea-water cooling system. There are operational procedures describing these manual actions. As long as it is possible to supply the EDGs with cooling water from the

During loss of sea-water cooling systems, it will be possible to keep the EDGs in operation for several days.

5.C.2.1.1 The spent fuel pool

There are two spent fuel pools at Ringhals 1. One pool is intended for long term storage and the other one is intended for defueling and are capable of storing a fully discharged core. Each pool has a volume of about 940 m^3 and contains unborated demineralised water. The spent fuel pools are connected to the rector pool via a transfer passage, which is separated from the reactor pool by two separate gates. One of these gates automatically closes if the water level decreases. The two spent fuel pools are connected via a gate, which can be closed if needed. During normal operations, the gates to the reactor pool are closed. During refueling, the reactor pool is filled with water and the transfer passage is open. All transport openings are in level with the top of the fuel. Under this level, there are no penetrations through the walls of the pools. Furthermore, there is no drainage system connected to the pools. Draining the pools can only be achieved by pumping. The discharge of the pools to the level indication tank and on to the fuel pool cooling system is constructed as an overflow, which is located 0.5 m below the brim of the pool. This means that if the water level decreases the cooling with the fuel pool cooling system stops. To regain cooling, the water level must first be restored. The spent fuel pool cooling system are designed to, cool the water in the spent fuel pools during normal operation, maintain the spent fuel heat removal during refueling together with the residual heat removal system, monitor the level of the spent fuel pools through level indication in the level indication tank, and provide an alternate flow path for demineralised water supply (of 800 m^3) to the auxiliary feedwater system when the regular water sources is unavailable. The system consists of two trains and the system belongs to the OPS. The system is supplied with power from the EDGs, and each train has a 100 % capacity. Initially during refueling, the spent fuel pools are cooled by the spent fuel pool cooling system in combination with the residual heat removal system, since the fuel pool cooling system does not have a sufficient capacity on its own to remove the generated decay heat. The residual heat removal system is designed to maintain the spent fuel residual heat removal during refueling together with the fuel pool cooling system. The system mainly consists of a single cooling train with two redundant pumps and two redundant heat exchangers. The pumps in the system are powered from separate EDGs. The system belongs to the OPS. For cooling during refueling, the residual heat removal system takes water from the reactor main circulation system and pumps it back to the reactor tank, just like during normal operation. The cooling capacity of the residual heat removal system can be used as a complement or as a redundancy to the spent fuel pool cooling system. In this case the residual heat removal system storage takes from the level indication tank and returns it directly to the spent fuel pool cooling system via a temporary connection. The design temperature for the spent fuel pools is 60°C. The pool temperature and the level in the pools are monitored in the main control room.

There are two spent fuel pools at Ringhals 2. The spent fuel pools are connected to each other and to the fuel transfer passage by manoeuvrable gates. Each pool and the transfer passage can be isolated by the manoeuvrable gates. The volume of each pool is 600 m^3 and there shall always be at least 7 m of borated water above the spent fuel assemblies. The volume of the transfer passage is 450 m^3. The spent fuel pool cooling system is designed to remove heat from the spent fuel. The system is supplied by power from the EDGs, but in case of loss of off-site power pumps must be manually restarted. However, the temperature rise in the spent fuel pools is assumed to be slow and it should be possible to wait 5 hours before restarting the pumps. The suction lines from the spent fuel pool cooling system are located at an elevation 1.2 m below the normal spent fuel pool level. The return lines terminate at an elevation 1.8 m above the top of the fuel assemblies, which ensures that the cold water will disperse to the bottom. The return lines contain anti-siphon holes near the surface to prevent drainage of the pools and a failure in the piping will not impair the water coverage of the spent fuel. However, if the water level drops 1.2 m below normal level, cooling with the spent fuel pool cooling system cannot be restored until the water level is increased above 1.2 m below normal. The minimum allowed boron concentration in the pools is 1875 ppm. Design temperature is 93°C. Each spent fuel pool is equipped with a transmitter to provide a low and a high level alarm with annunciation in the main control room. There is also a system for detecting leakage from the spent fuel pools with annunciation in the control room.

The spent fuel pools at Ringhals 3 and Ringhals 4 are very similar to the spent fuel pools at Ringhals 2. There are two spent fuel pools and these are connected to each other and to the fuel transfer passage by manoeuvrable gates. Each pool and the transfer passage can be isolated by the manoeuvrable gates. The volume of each pool is approximately 600 m^3 and there shall always be at least 7 m of borated water above the spent fuel assemblies. The volume of the transfer passage is approximately 450 m^3. The spent fuel pool cooling system is designed to remove heat from the spent fuel located in the spent fuel pools. The system is supplied by power from the EDGs and is equipped with a portable cooling pump shared between Ringhals 3 and Ringhals 4. The minimum allowed boron concentration in the pools is 2500 ppm at Ringhals 3 and 2550 ppm at Ringhals 4. The spent fuel pool cooling system consists of two redundant trains design to maintain the water temperature below 60°C with a fully discharge core and an additional 1/3 of a core of spent fuel assemblies remains in the pools. Each spent fuel pool is equipped with a transmitter to provide a low and a high level alarm with annunciation in the main control room. There is also a system for detecting leakage and one for temperature measurements with annunciation in the control room.

5.C.2.1.2 Design provisions to mitigate loss of primary ultimate heat sink

For Ringhals 1 the following cooling water supplies systems could be used in the event of a loss primary ultimate heat sink to maintain:

- Core cooling:
  - The safety injection system can be connected and supplied from the suppression pool, the fresh-water supply through the service water system, or the liquid waste processing system. Additionally, as a last option in severe accident conditions, the sea-water system can be connected through the service water system (injection of sea water).
  - The auxiliary feedwater system can be connected to the reactor pressure vessel through the residual heat removal system, or the boron injection system, and supplied from the storage tank for demineralised water, the liquid waste processing system, the spent fuel pools via the spent fuel pool cooling system, or the fresh-water supply connected to the storage tank for demineralised water through the demineralization plant.
  - The emergency feedwater system can be connected to the reactor pressure vessel through the feedwater system or the boron injection system, and supplied from the storage tank for demineralised water and the fresh-water supply connected to the storage tank for demineralised water via the demineralization plant.
- Cooling of the suppression pool and drywell:
  - The containment spray system and suppression pool cooling system can be connected and supplied from the fresh-water supply either through the fire-water system, or through the independent containment spray system and the mobile unit.
  - The filtered pressure release system, the MVSS

For Ringhals 2 the following cooling water supplies could be used in the event of a loss primary ultimate heat sink to maintain cooling of the steam generators through the emergency feedwater system:

- the condensate storage tanks,
- the fresh-water supply through the condensate storage tanks, and
- the storage tank for demineralised water through the condensate storage tanks.

For Ringhals 3 and Ringhals 4, the following cooling water supplies could be used in the event of a loss primary ultimate heat sink to maintain cooling of the steam generators through the emergency feedwater system:

- the condensate storage tanks (there is one for each unit),
- the fresh-water supply (valves have be opened manually), and
- the storage tank for demineralised water through the condensate storage tanks

The spent fuel pools

For Ringhals 1 the scenario will be equivalent to the scenario described in subsection 5.C.1.3.

For Ringhals 2, Ringhals 3 and Ringhals 4, there the mobile unit (supplied by fresh water or sea water) could potentially be connected to the spent fuel pool cooling system. However, such connections are expected to be quite difficult to obtain.

5.C.2.2 Loss of the primary ultimate heat sink (e.g., loss of access to cooling water from the river, lake or sea, or loss of the main cooling tower)

In an event of loss of primary ultimate heat sink, blockages occur both in the intake and the outlet.

Loss of intake for the primary ultimate heat sink is in the design biases for Ringhals 1and Ringhals 2 . Several of different scenarios related to blocking of the intake have been analysed and are presented in the safety analysis reports.

For Ringhals 1 these cases are limited by temperature in the suppression pool after 15 hours. In order to obtain long term cooling of the suppression pool at Ringhals 1, an alternate cooling source has to be used. For example, it is assumed that it would be possible to connect the fire-water system to the suppression pool cooling system to provide cooling of the suppression pool.

For Ringhals 2, the analysis shows that it takes 6 hours before the water temperature in the component cooling system exceeds the maximum design limit for the system. Also, in order for the EDGs to operate for 6 hours, additional cooling water from the fresh-water supply must be added to the dedicated cooling water volume for the EDGs. The residual heat is removed from the reactor by the auxiliary feedwater system and is then released to the atmosphere through relief valves. The water volume for the auxiliary feedwater system last for approximately 2 hours and additional cooling water from the fresh-water supply must therefore be added within 2 hours.

Analysis

For Ringhals 1 two scenarios are evaluated. In the first scenario recirculation in the sea-water cooling system is assumed to function. In the second scenario recirculation fails. Initially, the events will be equivalent to the design basis event included in the safety analysis report. The difference between the two scenarios will in principal be, that for scenario 2, the system for containment spray and suppression pool cooling will not be available, the temperature in the suppression pool will increase faster, and the MVSS will be activated earlier in the event sequence in order to release the pressure. In scenario 1, system MVSS is activated after approximately 20 hours and in scenario 2 it is activated after approximately 7 hours. During the first 4 days the water level in the reactor pressure vessel is 3.5 m above the fuel. This water level is assumed to be constant as long as there is water available for system emergency feedwater system. The temperature in the reactor pressure vessel will decrease after 4 days. Hence, there is no fuel damage expected as long as there is water available for system emergency feedwater system. The manual relief of pressure through system MVSS is initiated after approximately 20 hours, due to boiling in the suppression pool. For both scenarios, damage to the fuel is estimated to occur at earliest 35 hours after the initial event if only the dedicated volume of the demineralised water system is used for the emergency feedwater system. If maximum water volumes in demineralised water system and half of the volume in the fresh-water supply are available for the emergency feedwater system, it is assumed to takes at least 25 days before any damage to the fuel in the reactor core occur.

For Ringhals 2, Ringhals 3 and Ringhals 4 it is assumed that re-circulation of the cooling water will not be used since the auxiliary feedwater system is functioning and supplied by the condensate storage tank. The relief valves of the steam generators are used to remove the residual heat from the reactor. It is assumed that there is no leakage from the sealing of the reactor coolant pumps due to the fact that the off-site power is available and water for seal injection is available via the refueling water system tank (RWST). The limiting factor to withstand fuel damage is therefore the amount of water available for the auxiliary feedwater systems. The time until fuel damage becomes unavoidable is:

- For Ringhals 2:
  About 11 hours, if only the minimum volume of the condensate storage tanks is used, and 17 days if the maximum possible water volume in the condensate storage tanks and half of the volume in fresh-water supply are used for auxiliary feedwater system.
- For Ringhals 3 and Ringhals 4:
  About 8 hours, if only the minimum volume of the condensate storage tanks is used, and 11 days if the maximum possible water volume in the condensate storage tanks is used as well as the demineralised water storage tank, the buffer tank and the fresh-water supply are used for auxiliary feedwater system.

The spent fuel pools

At loss of primary ultimate heat sink, fuel damage becomes unavoidable earliest 1 week after initiating event during refueling. If all units at the site are affected simultaneously fuel damage becomes unavoidable after 8-9 days.

Also, the temperature exceeds 60°C after approximately 3 hours, boiling occurs after approximately 9 hours, loss of adequate shielding against radiation occurs after about 7 days.

5.C.2.2.1 Results from the licensee assessments of protection against loss of ultimate heat sink

At loss of primary ultimate heat sink, fuel damage becomes unavoidable at Ringhals 1 after 35 hours, at Ringhals 2 after 11 hours, and Ringhals 3 and Ringhals 4 after 8 hours. External support is not assumed to be needed.

For Ringhals 1 the MVSS activates after 7 hours for the most limiting case.

For the spent fuel pools, if all units at the site are affected simultaneously fuel damage becomes unavoidable after 8-9 days and loss of adequate shielding radiation occurs after about 7 days.

5.C.2.2.2 Measures which can be envisaged to increase robustness of the plants in case of loss of ultimate heat sink

The task of supplying the spent fuel pools from the fire-water system could possibly performed by control room personal, maintenance staff, or others as long as the environment in the spent fuel pool area is acceptable to enter without any protective equipment. When the spent fuel pools are boiling, trained personnel, such as fire fighters, will most likely be the only personnel that could enter the area. This might be a problem if all reactors are affected since the on-site fire brigade operates both system mobile unit and the on-site fire truck. The workload will be hard to handle for the on-site fire brigade.

After 72 hours or when roads are cleared, additional fire trucks, pumps and necessary human resources can be brought to the site by external fire brigades and they will bring the necessary human resources with them.

Re-evaluate the available volumes in the condensate storage tanks for Ringhals 2, Ringhals 3 and Ringhals 4. (This has already been initiated at Ringhals 2.)

Re-evaluate the prioritising of the fresh-water supply at site and evaluate the different possibilities to restore fresh water, or acquire fresh-water from the neighbouring town.

Evaluate possibilities to implement diversified cooling systems.

5.C.2.3 Loss of the primary ultimate heat sink and the alternate heat sink

At loss of the primary ultimate heat sink and the alternate heat sink all the sea-water cooling systems are lost and at the same a total loss of the possibility to release steam to the atmosphere through the steam generators occurs.

There is no alternate heat sink available for Ringhals 1.

Analysis

If the atmosphere through the steam generators is not available, pressure and temperature in the primary system will increase. To handle this situation, the recovery technique often called "bleed-and-feed" could be used. By manually releasing steam to the containment spray systems through the power-operated relief valves, pressure control of the primary systems will be maintained ("bleed"). Water inventory in the primary systems could be maintained by manually injecting water from the refueling water storage tanks ("feed").

To avoid fuel damage quick operational response to establish "bleed-and-feed" is however required, i.e. make-up water by the charging pumps has to be injected before steam generator dryout occurs.

If make-up water is delayed, damage to fuel will be unavoidable within 1 hour.

Additionally, cooling of the charging pumps must also manually be established to avoid failure of the pumps. This could be done by connecting the fire-water system to the cooling systems.

If pressures in the containment spray systems increases, due to the realised steam, water from the refueling water storage tanks could also be used for the containment spray systems.

Calculations indicate that the refueling water storage tank volume, is enough to last for approximately 13-18 hours at Ringhals 2, and 14-20 hours at Ringhals 3 and Ringhals 4.

If no manual actions are credited, this case will lead to the design sequence for the filtered containment venting system (MVSS), which is further described in the safety analysis report. In the designs sequence, station black out is combined with the conservative assumption that the steam driven systems are not available. However, in the designs sequence it is possible to cool the reactor core by natural circulation in the primary system. Steam will be released through the safety valves as long as there is water available on the secondary side. In the design sequence, this water has boiled off after approximately 1.5 hours. The reactor core begins to be uncovered half an hour thereafter. Thus, dependent on whether the fault is located in the steam generator safety/relief valves or in the water sources for the auxiliary feedwater system, the time until fuel damage becomes unavoidable differ somewhat (approximately one hour).

5.C.2.3.1 Results from the licensee assessments of protection against loss of ultimate heat sink and loss of alternate heat sink

Calculations indicate that available water inventory will be enough to provide make-up water for approximately 13-18 hours at Ringhals 2, and approximately 14-20 hours at Ringhals 3 and Ringhals 4.

If manual actions are delayed damage to fuel will be unavoidable within 2 hour.

5.C.2.3.2 *Measures which can be envisaged to increase robustness of the plants in case of loss of ultimate heat sink and loss of alternate heat sink*

Re-evaluate and potentially update the operational procedures for bleed-and-feed. In the existing procedure it is not stated how much water should be used from the refueling water storage, for make-up water respectively containment spray.

5.C.3 Loss of the primary ultimate heat sink, combined with station black out

The plant response to loss of the primary ultimate heat sink, combined with station black out will be the same as described in section 5, Loss of electrical power, where a loss of off-site power and loss of the ordinary back-up AC power sources analysed combined with and without loss of permanently installed diverse back-up AC power sources.

5.C.3.1 Time of autonomy of the site before loss of normal cooling condition of the reactor core and spent fuel pool (e.g. start of water loss from the primary circuit)

Results are presented in sections 5.C.1.3.

5.C.3.2 External actions foreseen to prevent fuel degradation

Results are presented in sections 5.C.1.3.

5.C.3.3 Measures, which can be envisaged to increase robustness of the plants in case of loss of primary ultimate heat sink, combined with station black out

Results are presented in sections 5.C.1.3.

5.2 Assessments and conclusions

5.2.1. The licensee assessments and conclusions

Results from the licensee assessments are for this chapter, due to the extent, given separately in the subsections related to respective topic and scenario. The main findings are however discussed in the following subsection where the authority prospective is given.

5.2.2. Licensees recommendations for potential improvements

The licensees identified potential measure are for this chapter, due to the extent, given separately in the subsections related to respective topic and scenario. The main findings are however discussed in the following subsection where the authority prospective is given.

5.2.3. The Swedish Radiation Safety Authority's assessments and conclusions

The Swedish licensees have performed assessments of protection against loss of electrical power and loss of ultimate heat sink in accordance with the EU "Stress tests" specifications. The authority review shows that the overall intention of the EU "Stress tests" has been fulfilled. Identified deviations from the EU "Stress tests" specifications have been documented.

In the following sections the general assessments and the main conclusions from the licensee assessments of protection against loss of electrical power and, or in combination with, loss of ultimate heat sink, as well as identified essential measures for improvements are given. Additionally, all measures for improvements listed in the licensee reports will be considered as potential measures to increase robustness of plants.

5.2.3.1. General

All Swedish NPPs are equipped with severe accident mitigation systems. The licensee assessments of loss of electrical power, and or combined with, loss of ultimate heat sink, demonstrates the importance of the severe accident mitigation systems to limit the consequences. However, long term operation of these systems has not fully been verified. Hence, further evaluations and reassessments of the severe accident mitigation systems in a long term prospective, will be considered as potential measures to increase robustness of plants.

Additionally, for the Swedish BWR designs licensees have described that the severe accident mitigation systems provides possibilities for both residual heat removal from the reactor core to the atmosphere during these events and level control of the suppression pools. The use of these systems before severe damage to fuel has occurred where not considered in the design of these systems. Further evaluations and reassessments of these applications and relating strategies, will be considered as potential measures to increase robustness of plants.

5.2.3.2. Loss of electrical power

The ordinary AC-power supply systems for all Swedish NPPs are designed to withstand loss of off-site power for 7 days. However, lubricating oil supply has to be refilled after a few days at most units, and for Oskarshamn 1, external lubricant oil delivery might be needed within less than 72 hours. Further evaluations and reassessments of the lubricating oil supply will be considered as potential measures to increase robustness of the plants.

The Swedish NPPs have alternate means, via automatic or manual actions, to provide units with electrical power from gas turbines on-site or in the vicinity of the site. These alternate back-up AC power sources are in most cases not full qualified. Assessments indicate however, that the performance and the units' connection to the alternate back-up AC powers supply might not be robust in all situations especially when all units at one site are affected. Further evaluations and reassessments of the alternate back-up AC power supply will be considered as potential measures to increase robustness of the plants.

In case of loss of off-site power, loss of the ordinary back-up AC power sources, and loss of permanently installed alternate back-up AC power source, the licensee assessments shows that the Swedish NPPs rely on the battery power supply for instrumentation and control. The battery power supplies are only qualified for 2 hour of operation without charging. Experience and estimations according to the licensees indicates however that the battery backed-up power supply might be maintained for more than 2 hours if, for example, less important loads would be disconnected. Further evaluations and reassessments of the battery power supply will be considered as potential measures to increase robustness of the plants.

During loss of off-site power, loss of the ordinary back-up AC power sources, and loss of permanently installed alternate back-up AC power source, the licensee assessments also shows that the Swedish NPPs Swedish NPPs rely on different type of the mobile equipment located at the sits. The assessments show however, that the number of available mobile units at the site is not sufficient, in particularly in case of simultaneously events at more than one unit. Further evaluations and reassessments of mobile units will be considered as potential measures to increase robustness of the plants.

Additional, the following items have been identified by the authority review in addition to the measures identified by the licensees during the licensee assessments of protection against loss of electrical power, and will be considered as potential measures to increase robustness of the plants:

- Pipes and equipment needed for emergency diesel generator refueling will be evaluated and re-assessed considering all natural phenomena and other events that could arise outside or inside the facility.
- Evaluate further needs of additional diversified equipment within the important instrumentation and control functions (incl. power supply) needed in accident conditions.
- Evaluate further need of standardised mobile diesel generators and standardise well protected connection arrangements the units/sites.
- Evaluate the accessibility of the important areas at the site and inside the plants (incl. all areas where access is need for successful execution of manual actions) during accident scenarios, especially following natural phenomena and other events that could arise outside or inside the facility.

5.2.3.3. Loss of ultimate heat sink

All Swedish NPPs are design to withstand a blockage of the sea-water inlet following any natural phenomena and other events that could arise outside the plants. In a blockage of the cooling water intake, all Swedish plants will be able to safely shut down and maintain safe shut down conditions. For Ringhals 3 and Ringhals 4, demonstrations of design basis have not been fully verified. Additional need for verification will be further evaluated.

The licensee assessments shows that blockage in both intakes and outlets could lead to significantly more challenging situations for the plants, and will require high operational performance and advanced manual actions. For the Swedish BWR designs licensees have described that the severe accident mitigation systems could be used for maintaining a safe state in these situations. However, this has not been considered in the design for these systems and will be considered for further evaluations and reassessments as described in subsection 5.2.2.1.

Additionally, the licensee assessments of blockage in both intakes and outlets demonstrate the importance of independent cooling functions, where both permanent alternate equipment and/or mobile functions will improve the safety and the robustness of the plants. Further evaluations and re-assessments of existing and/or potential new installations of permanent (and/or mobile) alternate equipment for emergency cooling of the reactor cores, the steam generators, the suppression pools, and/or the spent fuel pools, will be considered as potential measures to increase robustness of the plants. Also, Further evaluations and reassessments of strategies for mobile units (incl. availability, installation, operation and consumable supplies like fuel, lubricating oil and water) are considered as potential measure to increase robustness of the plants, will be considered as a potential measure to increase robustness of the plants.

The licensee assessments of loss of ultimate heat sink also shows that the available time before damage to fuel will be unavoidable is highly dependent on available water volumes (in storage tanks). Further evaluations and reassessments of the minimum acceptance values for levels in storage tanks and also the priority of water volumes in storage between units will be considered as a potential measure to increase robustness of the plants.

To secure spent fuel pool cooling the licensee assessments shows that manual actions are required for all Swedish NPP. Further evaluations and reassessments of spent fuel pool cooling capabilities (incl. accessibility, availability, capacity, installation, operation, etc) will be considered as a potential measure to increase robustness of the plants.

Additional, the following items have been identified by the regulatory body in addition to the measures identified by the licensees during the licensee assessments of protection against loss of ultimate heat sink, and will be considered as potential measures to increase robustness of the plants:

- Re-evaluations of emergency procedures and implementation of possible updates regarding all alternate cooling capabilities available at site.
- Re-evaluations of emergency procedures and implementation of possible updates regarding manual hydrogen venting.
- Further evaluations of simultaneously event at all units are needed for the Forsmark site.

6 Severe accident management

6.1 Introduction

This section has four chapters. Chapers 6A-6C are short versions of the licensees deliverables. Althrough, the systems and strategies are essentially the same for all licensees. There are some differences in their severe accident managements and stress tests which make their reports to some extent complementary to each other. Chapter 6.2 is the total summary of all Licensees assessments and conclusions as well as the authority's assessement and conclusions.

The authority regulations which are put on the nuclear facilities in the context of emergency preparedness are formulated in SSM 2008:1, The Radiation Safety Authority Regulation and General Recommendations about Safety and Security in Nuclear Facilities and SSM 2008:15, The Radiation Safety Authority Regulation for Emergency Preparedness at Nuclear Facilities. Specifically

- SSM 2008:1, chapter 2, section 12; The accident facility shall provide information to responsible authorities on the technical situation of the facility.
- SSM 2008:15, section 5; An emergency preparedness plan for radiation protection activities in an emergency shall be established by the licensee's auspices. The emergency preparedness plan should describe the emergency organization which is intended to come into play in an emergency.

6.A. Forsmark

6.A.1 Organization and arrangements of the licensee to manage accidents

6.A.1.1 Organisation of the licensee to manage the accident

6.A.1.1.1 Staffing and shift management in normal operation

As a minimum, the following personnel are available for carrying out immediate actions:

- Shift on duty (including security guards)
- BC-operator on duty
- Engineer on duty (VHI)
- Fire-fighter's officer in command on duty (Bf) and four personnel (firefighters) in the rescue force

The operational supervisor is the head of the control room team and supervisor of the shift team. He has permanent responsibility for ensuring that operations comply with the established operational criteria. At alarm criteria according to the disturbance procedures for each unit, the operational supervisor contacts the operational management. The operational management/VHI makes the decision to declare the preparedness level.

The Engineer on duty is always on call at Forsmark according to a specific staffing plan, with a maximum response time to the concerned control room of 15 minutes. The VHI can, if necessary, replace the usual operational management (all three levels) and thereby fulfil its functions.

6.A.1.1.2 Measures taken to enable optimum intervention by personnel

The Emergency Preparedness Organisation at Forsmark consists partly of a common on-site emergency control centre (KC) and partly a unit associated section (unit preparedness). The organisation is put into operation following a decision by the operational management when certain safety-related criteria are met. At the same time, the concerned authorities with responsibilities for protecting the public are informed immediately. The actions are governed by how the event is classified, and can include everything from information to unit personnel, to information to and alerting the authorities. Operations Management/VHI (Engineer on duty) gives instructions to the personnel; this is conveyed by the BC (Security Centre) in the form of alarm signals and/or loudspeaker announcements. At proclaimed level of preparedness, the VHI is responsible for taking actions until Area Superviser (OL) arrives. The main tasks in an extraordinary situation are:

- Restore/maintain the facility to ensure a stable and safe state.
- Take appropriate protective measures for the personnel and facility.
- Minimize impacts to the environment as far as possible in case the incident poses a threat to environmental safety.
- Notify/alert the authorities that have the responsibility for third parties, continually report on the status of, and give a prognosis on, developments to these authorities.

When developing plans and activities ensure, as far as possible that established routines for normal operation are used.

When calling in personnel in the emergency preparedness organisation can be done in two ways, regardless of the time of day:

- via alarm signals and announcements over the loudspeaker system
- via telephone and radio The call in is performed by the staff in the security monitoring centre.

In addition, personnel can be called in in accordance with specific directives.

6.A.1.1.3 Use of off-site technical support for accident and protection management

In case of a severe accident, Forsmark emergency preparedness organisation contacts the emergency group at Vattenfall. The emergency group at Vattenfall has yearly joint exercises with the company's emergency preparedness organisations. The group holds competence regarding the unit's design and function, radiation protection/radiology and reactor safety. However, it is important to note that the licence holder, i.e. Forsmark, is responsible for all actions undertaken to mitigate the consequences of an event.

6.A.1.1.4 Procedures, training and exercises

Procedures for handling accidents are implemented as follows:

- The procedures consider all possibilities for cooling the core in the event of severely degraded functions in the operation and safety systems.
- The procedures specify management guidelines in order to be able to bring, even at a severe core accident, the plant to a stable condition with the damaged core covered with water,
  - in a cooled and depressurized state in the reactor vessel
  - in an intact, cooled and depressurized containment.

The procedures ÖSI (Emergency Operating Procedure) are intended to cover the entire sequence from reactor trip to a severe accident with a stable safe shutdown condition. In the procedures the emphasis is on checks and measures to be taken in the short term, i.e. within approximately 1 hour, but the Emergency Operating Procedures are intended to cover up to 24 hours and the operators are trained to deal with this time sequence.

For severe accidents, in the long term, when an emergency preparedness organisation has been established as support, the ÖSI are less detailed and of a more general nature. In these instances, the operational management (DL2/DL3) assists in handling the situation.

In order to handle disturbances in the unit, the following documents are available for the operating staff:

- TSD, Technical Support for Operational Management (Operational Management,
- Plant Operations Manager)
- ÖSI, Emergency Operating Procedure (Operation Supervisor, Operational
- Management)
- ASI, Plant Disturbance Procedures (Reactor Operator, Turbine Operator)
- ADI, Plant Operating Procedures (Operation Supervisor, Reactor Operator, Turbine Operator)
- DI, Operating Procedures per system (Operation Supervisor, Reactor Operator,
- Turbine Operator, Field Operators)
- THAL, Technical Handbook for Plant Operations Managers

The functions within the emergency preparedness organisation have function-specific checklists. The lists have an initial section for the collection of facts and early action, followed by a section on recurring measures. Management of these checklists is included as part of the training done in preparation for taking a role in the organisation. The checklists are included as part of the procedures. These are updated regularly and feedback from the exercises is an important basis for improving the instructions. All functional procedures are available in paper form in order that they are independent of computers, but still available. The functions are practised in accordance with the contingency operations training programme and courses are documented.

Exercises are repeated according to established annual plans. The purpose of the major exercises is to maintain the organisation's ability to handle the internal information and interact with external preparedness functions. These major exercises are simulation exercises, sometimes with minor elements of field exercises, such as taking care of injured persons. The exercises are normally held at the alert levels "Off site alert" or "General emergency alert", i.e. radiological release is occurring or may be expected within 12 hours. Usually the training assumes that a unit is subjected to such serious disturbances that a release from the facility cannot be excluded, but never that all three units at Forsmark have similar events in parallel. Experiences from these exercises are compiled in a report with proposals for improvement and for the designated responsible personnel.

6.A.1.2 Possibility to use existing equipment

At the on-site emergency control centre (KC) there are IT applications for support, PCs and several types of communications equipment. The centre is located in a shelter, which provides good protection against radiological release and geological disturbances. The power is supplied from Forsmark 1 or Forsmark 2, but can also be supplied by a diesel-driven generator in KC. A special telephone system is available in each unit, the so-called Forsmark's reactor containment protection system (FRISK)/Remote Shutdown Panel (RÖP) phone. This telephone network is independent with its own equipment and batteries. The system is tested and serviced regularly. The system works only within its own unit. In the control room there are hand and head torches for use in the event of total loss of electrical power.

At the Forsmark plant there is an internal rescue team consisting of a team leader and three fire-fighters. They have their own fire-fighting vehicle. There are also designated persons in the shift team of the affected unit who can be a useful resource, particularly concerning the connection of a water supply to the plant.

For major events, the Emergency Preparedness Organization needs outside assistance to rescue personnel, and extinguish fires in the plant. A prerequisite for them to operate is that they can get to the facility. The following statement applies, provided the access roads are trafficable: The emergency service in Östhammar can be on site within 20 minutes Ambulances from Östhammar can be on site within 20 minutes when required. Within Forsmark there is also a diesel-operated mobile pump set. This can be used to supply water to the spray system and several different water sources can be used. This depends however, on the condition that it is possible to move the pump units and their associated equipment and to work on the areas around the units.

In the external emergency preparedness organisation, the emergency supervisor at the County Administrative Board of Uppsala controls public emergency operations under the Act on Protection against Accidents. The rescue supervisor's powers to mitigate the consequences of an event are practically unlimited, and the whole of society's resources can be requisitioned. These resources should be used primarily within the area of responsibility that is outside the power plant, but it is reasonable to assume that resources can also be made available to the plant where they can be expected to mitigate the impact of the event.

The critical supplies are a) diesel fuel and lubricating oil for emergency power diesels, b) raw water as source for various cooling demands, c) nitrogen, d) boric acid for control of reactivity and e) ion exchange resin for purification of wastewater.

The Radiation Protection Supervisor in the emergency preparedness organization has to ensure that recovery personnel do not exceed specified limits/personal dose restrictions. In addition, there are instructions on the radiation doses that are acceptable for different types of work and on adjustment of the dose rate alarms. The intake of iodine tablets and the use of protective equipment is also governed by instructions.

Training in operation and expected environments in connection with severe accidents is instruction governed and the requirement includes staff groups, operating management, operations, and specific areas of maintenance, fire and radiation protection.

Dosimeters used at extraordinary radiological event are of the electronic direct reading type. The adjustment and setting of limits is made by the health physics office. Measurement is instantaneous and it is possible to read the received dose after every work period. This permits monitoring of received personnel doses and continuous verification that specifically determined dose levels are not exceeded.

In the rooms where personnel are continuously present there are either permanently installed dose rate alarms or mobile units available for deployment. Measurement of gamma radiation is via six fixed (as well as one mobile) measuring probes in the surrounding area.

Duty personnel, who in an emergency situation may need personal protective equipment immediately, must have such equipment directly available at their workplace. Iodine tablets should also be available. Examples of such personnel are the control room staff, rescue force and BC staff. The VHI support also have immediate access to protective equipment.

In addition to the monitoring of all emission points, an automatic gamma monitoring system is installed in the immediate area for the monitoring of air emissions, with 7 measurement stations that automatically send data to a server in KC. Access to data is achieved via Forsmark's IT network.

Alerting occurs via the plant's loudspeaker system. It is possible to give the "immediate danger" alarm from the control room, monitoring centre and control centre. Voice communications can also be transmitted from the on-site emergency control centre, control room, monitoring centre and from the main switchboard.

The assembly points are equipped with fax and phone, and some of them are also equipped with screens for written communications. Forsmark intends to introduce this valuable information method to all the plant's assembly points. Initially, Forsmark needs to alert its own Emergency Preparedness Organisation and external preparedness functions. The latter is currently carried out by an SOS Alarm, which by means of alert lists can propagate the alert further.

Once the emergency preparedness organisation is established, internal communications are required to create a joint status report and forward it to the external emergency preparedness functions, whose task is to limit the harmful effects in the surrounding area, such as the evacuation of nearby residents. Also needed are internal communications so that the Area Supervisor can control operations by prioritising actions and deciding on their execution. An example of this is evacuating the plant's own personnel. When the plant is evacuated, it is expected that most communications will take place between KC, Operation control centre and the security centre. In order to communicate internally and externally there are a number of systems such as a) Meridian PBX with fixed connections and DECT (Digitally Enhanced Cordless Telecommunications), b) Fixed telephones via external connections (so called third-party subscription), c) Forsmark radio, d) Loudspeaker system, e) Switchboard in the on-site emergency control centre, f) Mobile telephones via GSM, g) operations phone (via 400 kV network), h) Telephones via military lines (FTN) and Pager.

6.A.1.3 Evaluation of factors that may impede accident management and respective contingencies

The factors that may impede accident management has been identified for each area but are in many cases common for several areas. The more important ones are summarized here for all areas.

To assure that the staffing of the emergency preparedness organisation is available it is required that GSM/telephone network is intact. Forsmark's call in system RapidReach is based on that telephone network is operational. Infrastructure, roads and communications must be available and trafficable. The supply of food and beverage for personnel within the emergency preparedness organisation can be a critical issue during the event sequence.

Communication takes place between control centres (Security centre, Operational centre and on-site emergency control centre (KC)). To communicate with other personnel, loudspeaker equipment, radio equipment, digital notice boards and intranet is used. Instruments for dose rate measurement and weather data collection is used to survey the radiological affects on the surroundings and the present weather situation. If ordinary communication equipment is not operable, the On-site emergency control centre's separate telephone switchboard can be used. Further, the military telephone line and the Forsmark-radio can be used. Battery backup is available for ordinary telephone switchboard. Diesel generator back up supplies the KC switchboard.

Communication tools and problems with the lighting make control and management difficult. The numbers of head torches that are available have been increased. Essential parts are assessed to be accessible. However, when flooding occurs over the ground level, electrical equipment will stop to function which, will affect the ability to control and monitor various functions.

Forsmark1, 2 and 3's main control rooms are located in each E-building on the ground floor about 3 m above sea level. The control rooms are equipped with charcoal filtered emergency ventilation. When the main control rooms are not accessible, there are Remote shutdown panel locations, known as RÖPs.

Without electricity supply, the functions in the control rooms, RÖPs and LMPs for monitoring, communicating and controlling actions in the plant cease to function. Following from the above, the local control stations (LMPs) are designed for 24-hour loss of all external electrical power, as there are separate earthquake-proof batteries available. The batteries supply measurement equipment and indicators. Doors can be opened manually by using keys this requires no power supply. Rotating gates for security protection make passageways more difficult and they will probably have to be by passed before Entering-passage. The EXIT passage has a prepared emergency control.

The on-site emergency control centre (KC) has shielding and can be operational using the filtration of intake air or completely sealed with recirculation of air and the supply of oxygen from separate systems. Instruments are available to measure the radiation level in the intake air and the control room. If ordinary power supply to KC is lost, there is a separate diesel-driven generator in KC. Batteries for severe accident mitigating systems are available for 24 hours. Mobile generator must be brought from outside. Connection points prepared only for consequence mitigation systems.

The consequence mitigating systems have special manoeuvring principles. In the event that battery power is in operation, the consequence mitigating systems and the status of the reactor containment can be controlled from the main control room. If the core meltdown has been caused by a total loss of power (known as an SBO), which in itself may be primarily caused by earthquake, there are local control stations (LMPs) for these earthquake-resistant functions. The control of valves from the LMP requires no batteries. Isolation valves in the consequence-mitigating systems are remotely manually operated, by compressed air and long pneumatic control lines from the LMP.

In a hazardous situation, the staff is alerted by warning signals and loudspeaker announcements, i.e. calls to go urgently to the nearest Gathering point (SP). Staff included in the emergency preparedness organisation act, however, in accordance with their initial order. The SP is signposted and provided with instructions and iodine tablets. Activities at the assembly points are lead by a pre-designated gathering point supervisor (SPA), whose mission is to be at the operating management's disposal. At the gathering point, personnel await further information and instructions. During non-office hours, activities are directed by the person who arrives first. The gathering point supervisor follows established instructions and ensures in particular that those arriving are registered in the identification card reader.

During office hours, it is expected that the emergency organisation will be established within an hour of the initial event, after which the Area Supervisor (OL) is responsible for information and instructions. The nature of the event may be such that it takes some time for the message to be relayed to the staff at the SPs. The waiting period can, for example, be used to prepare for a possible evacuation. Portable dose rate alarms are deployed at all SP in conjunction with the "Off site alert" and "General emergency alert" levels of preparedness. Iodine tablets are always available. There are assembly points at different places such as lunch rooms, sports hall, information building, administration building, etc.

If the infrastructure is damaged and the roads are not trafficable, it will be a delay when requiring backup mobile power supplies and water supplies. Supply of oil and supply of nitrogen can be a problem in the longer perspective if the infrastructure is damaged. Existing oil tanks on Forsmark 1 may be damaged. Use of system for water filling of containment can be affected if mobile units are needed.

Aggravating conditions for working with mobile replacement equipment is high radiation outdoors, extreme cold, large amount of snow or flooding. Use of system for water filling of containment can be affected if mobile units are needed. It is assessed that it is not possible to perform whole body measurement in the plant.

Essential parts are assessed to be accessible. However, when flooding occurs over the ground level, electrical equipment will stop to function which will affect the ability to control and monitor various functions.

When all units simultaneously are subjected to severe disturbance it can be difficult to have sufficient number of competent personnel available. The KC operation is organised for one affected unit. There is no training or exercises for all the units being affected. Instructions/checklists have been prepared on the basis of one affected unit. At an event where several units are affected at the same time one fire truck can only support one unit. Within the facility, there is only one fire truck available.

The work load on those who handles dose will increase. More individuals receive higher doses. Overall it gives a greater workload. Necessaries such as food and beverages to the personnel in the Emergency preparedness organisation and the access to skilled personnel can become a vital issue. Lengthy event sequences burdens the personnel's capability. If a larger area around the units becomes contaminated and evacuations are made, the access to external resource enforcement can be strained. The accessibility is impaired due to radioactive releases and high dose rates from multiple units. Difficulties in moving around the facility – all personnel must wear protective masks, etc. In summary, release affect all manual operations required.

6.A.1.4 Conclusion on the adequacy of organisational issues for accident management

The personnel in the emergency preparedness organization are from Forsmark and they fulfil 25 different functions. There are 2-9 individuals per function. In total, there are about 190 people in the emergency preparedness organization. Add to this the personnel from organisations such as radiation protection, fire and maintenance, which are expected to perform essentially the same tasks as during normal operations. The Swedish nuclear power plants have established joint agreements for reinforcement resources for radiation protection. Skilled resources are available outside the own organization. As an added resource, there is an emergency group with technical competence within Vattenfall.

However, some of the functions included in the emergency preparedness organization have so few personnel that only after a few days it would be difficult to sustain shifts around the clock, especially if more than one unit is affected. It is of great importance to secure resources as soon as possible by establishing shift lists for future staffing. A special separate function for this is available in the on-site emergency control centre (the Communications Supervisor).

Food supplies are available in KC and they cover the requirement for the first few days. A restaurant with basic foodstuff is also available within the facility. In the longer run, food has to be taken in from an outside supplier. The County Administrative Board has extensive powers to allocate public funds when a severe event occurs, for both food supplies and transport for personnel and materials.

Supply of oil and supply of nitrogen can be a problem in the longer perspective if the infrastructure is damaged. Overdue/missing deliveries cause delays. Aggravating conditions for working with mobile replacement equipment is high radiation outdoors, extreme cold, large amount of snow or flooding. If a larger area around the units becomes contaminated and evacuations are made, the access to external resource enforcement can be strained.

If ordinary communication equipment is not operable the On-site emergency control centre's separate telephone switchboard can be used. Further, the military telephone line and the Forsmark-radio can be used.

Use of system 365 (System for water filling of containment) can be affected if mobile units are needed. Cold, snow, flooding, extreme wind are examples of external effects.

6.A.1.5 Measures which can be envisaged to enhance accident management capabilities

Current Emergency preparedness organisation is primarily designed for being able to handle a severe accident at one of Forsmark's three reactors. If two or three reactors are subjected to a severe accident, the dimension of the current staffing and shift-rotation can lead to that all required accident management cannot be executed as intended. At present, all simulator trainings and emergency trainings is performed with the prerequisite that only one unit is affected. A thoroughly developed plan for managing several, simultaneously affected units should be made.

For control, monitoring, communications and emergency lighting systems, a battery system is required. At Forsmark 1 and Forsmark 2, upgrading of the RÖP is taking place within the safety upgrade programme. In addition, there will be increased capacity for management, monitoring and computer systems.

The current Emergency preparedness has not considered the possibility of hydrogen leaking out and accumulating in the reactor building. This can lead to an explosive mixture because the reactor building is air filled. The issue should be analysed and possible countermeasures implemented. Another issure is to update the strategies for handling re-criticality, both for detection and countermeasures.

Decision support for handling hydrogen in a lengthy sequence should be reviewed and improved. Among the issues to be addressed are estimating gas composition in different volumes in the containment, the possible need to add nitrogen to render the containment inert after ventilating, and elimination of hydrogen gas by combustion.

At present, the call-in of personnel depends on a functioning GSM/Telenet. An improvement in this area could be the purchase of satellite phones. By December 31, 2011 at the latest, however, the RAKEL radio system will be operational, which also will be independent from ordinary GSM/Telenet. The system is the same as that used by the police, emergency service, ambulance and other agencies. The system is encrypted and the transmitters have a secure power supply using diesel generators. To ensure the call-in of personnel, new call-in methods has to be developed. Alternatives to be used is the local radio or via Ringhals' call-in system (which is identical to the one in Forsmark).

6.A.2 Accident management measures in place at the various stages of a scenario of loss of the core cooling function

This section describes accident management, strategies and the existing system utilized in the event of an accident in which the core cooling function is lost.

It should be noted that some of the strategies describe measures that can only be carried out if electrical power (AC) is available, while other measures are entirely passive or can be carried out when a power system with battery back-up (DC) or compressed air supply is available.

6.A.2.1 Before occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes

In a scenario where the core cooling function has been lost, recovering the cooling is extremely important. If power supply is available, efforts are made to restore the feed water system and start different safety systems to inject water to the reactor. All systems for pumping water into the reactor vessel at the Forsmark units are dependent on electrical power supply. In cases where both the external grid supply and the standby diesel system have been lost, an alternative electricity supply can be obtained from the gas turbine in Gunnarsbo.

At present, there is a risk that the gas turbine protection will be triggered if all three start-up transformers are energized simultaneously. If this does occur, manual intervention will be required and it is then probable that the power supply to the feed water system and the safety systems cannot be established before fuel damage has occurred.

In the event of station blackout and the gas-turbine plant, is not available there is no possibility to restore the core cooling function.

The current strategy is then geared to mitigate the consequences of an accident resulting in fuel damage and reactor vessel melt-through, by protecting the containment and minimizing the release of radioactive substances to the surroundings.

6.A.2.2 After occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes

In the case power supply is available, it is always prioritized to try to start different systems for injecting water to the reactor.

Damage to fuel cladding because of failure to cool or insufficient cooling leads to radioactivity release of inert gases and fission products from the fuel to the reactor vessel and the coolant circuit. This in turn gives rise to radioactivity release from the reactor vessel and/or primary circuit to the containment. From the containment, radioactivity can leak out to various systems and areas, mainly in the reactor building. In order to minimize the spread of radioactivity, a number of measures are described in THAL (Technical Handbook for Plant Operation Managers) such as a) how to minimize diffuse leakage, b) how to minimize spread of airborne radioactivity, c) how to minimize aerosols in the containment, d) how to minimize contamination of systems connected to the containment and e) how to make provision for dealing with liquid waste.

6.A.2.3 After failure of the reactor pressure vessel/a number of pressure tubes

According to THAL, if vessel melt-through has taken place, maintaining the integrity of the containment in order to minimize the consequences in the form of release of radioactivity to the surroundings is prioritized. The transition from prioritizing cooling of the core in the reactor vessel to maintaining the integrity of the containment is called "benbyte" (change of legs). So changing legs requires that a reactor vessel melt-through has taken place. The strategies are described in THAL for a number of vital areas for accident management in the long term.

The chemistry in the containment is important for minimizing corrosion and retaining iodine in the sump in the bottom of the containment. It is recommended to achieve a pH of approx. 10 within a week of the start of the accident.

The following parameters are deemed to be of greatest interest: a) radiation level, b) amount of radioactivity, c) hydrogen content, d) boron content and e) pH value.

During this period, before the reactor is returned to a stable shutdown condition, information about the radiation level in the containment is monitored. This is used to assess core damage that has occurred. The hydrogen and oxygen content in the containment can be monitored, with the aid of a gas chromatograph online. The boron content and pH can only be measured in water samples taken manually. In connection with this test an estimation of the radioactivity content, including determining the nuclide composition is made. If the radiological situation permits, samples from the containment water and gas phase can be obtained from the accident sampling of system.

The crucial factor in minimizing the leakage of radioactivity from the containment in the long term is to maintain the integrity of the containment. In the long-term, leakage of radioactive water is the biggest problem. High levels of radioactivity will remain for many months. There are two ways of detecting water leakage from containment to reactor building: Monitoring of drainage systems and waste tanks or rounds.

The cooling requirement for the containment is determined by the fuel's decay power. The containment can be cooled either passively or actively. In the former case, cooling is done by conducting heat through the containment roof, walls and floors. Active cooling refers to the use of existing systems for such cooling. The containment needs active cooling for five years after the start of the accident. If passive cooling alone is used, the pressure in the containment will rise until pressure relief occurs via the scrubber, which can produce releases to the surroundings for several years after the start of the accident.

During a severe accident, large quantities of hydrogen are formed during core meltdown. The hydrogen is contributing in raising the pressure in the containment. In the short term no hydrogen combustion can occur, since the containment is filled with nitrogen gas. In the long term, radiolysis of water in the containment occurs, resulting in the concurrent formation of hydrogen and oxygen. To handle hydrogen in the containment, it is important to be able to determine the hydrogen concentration. The post-accident sampling system can be used to continuously obtain the value of the hydrogen content in the containment. Two measuring points exist, one in the drywell, one in the wet well.

Following an accident with reactor vessel melt-through the containment filtered venting shall prevent slow overpressurization of the containment. The system can also be used to reduce the pressure in the containment in order to reduce gaseous leakage. The containment filtered venting system comprises a water scrubber with an associated piping system connecting the scrubber with the containment. The system is activated either automatically (if the pressure in the containment exceeds 5.5 bar in Forsmark 1/Forsmark 2 or 6 bar in Forsmark 3) or manually at lower pressure by opening isolation valves. By activating the system, some of the atmosphere in the containment passes through the scrubber. Nearly all radioactivity except the contributions from inert gases (and organic iodine, mostly methyl iodide) is captured in the scrubber. Therefore, all the disadvantages of discharging inert gases must be factored into the assessment. In order to eliminate the risk of hydrogen fire, the system is filled with nitrogen gas in standard operating mode. After using the system, it must be replenished with nitrogen gas.

Water filling of the containment with water after vessel melt-through is performed to achieve a stable shutdown condition with the core remains cooled at the bottom of the containment. By spraying the water into the containment, the pressure is reduced, radioactivity substances in the atmosphere of the containment is washed out and transferred to the liquid phase, and the containment is filled with water. Spray is started, at the latest at reactor vessel melt through.

All cooling and cleaning systems connected to the reactor are at risk of being contaminated with the water from the containment. The level of radioactivity is determined by the extent of the core damages and if the system has been connected to the containment for a shorter or longer period of time. Drainage system also risks becoming contaminated, as does sampling system.

Measures taken are aimed to route contaminated water back to the containment, and if this is not possible, attempt to restrict the contaminated water to as few tanks as possible. Conceivable measures are to:

- use only the post-accident sampling for sampling the containment's gas and water phases
- pump drainage from outside containment to the condensation pool

6.A.3 Maintaining the containment integrity after occurrence of significant fuel damage in the reactor core

This section describes accident management and plant design for protecting the integrity of the containment in the event of an accident in which the core cooling function has been lost.

6.A.3.1 Elimination of fuel damage / meltdown in high pressure

If core cooling function has been lost and there is a risk of fuel damage, it is important to ensure that the pressure in the reactor vessel can be lowered to a level that eliminates the risk of high-pressure reactor vessel melt-through, as this might threaten the containment integrity due to Direct Containment Heating (DCH). In order to prevent high-pressure melt-through of the reactor vessel, the reactor water discharge valves are used, opening when the level in the reactor vessel reaches extremely low level (L4). At level L4, the core is still covered with water. The depressurization function is reliant on a battery back-up system but not on any other electrical power supply. Opening of the water discharge 314 valves can also be done manually. The water discharge valves will not close but will be blocked in open position.

6.A.3.2 Management of hydrogen risks inside the containment

Under normal operational mode, the containment is filled with nitrogen, i.e. inert. During shut down and start up, brief periods of air-filled containment occur. The conditions may then be present for the gas mixture in the containment to become combustible. The times involved are relatively short, and hence the risk contribution is small. Ventilation with filtered pressure relief reduces the amount of hydrogen in the containment and reduces the probability for and the consequences of hydrogen deflagration or detonation. After pressure relief through the scrubber, it is ventilated with nitrogen in order to prevent hydrogen deflagration or detonation occurring there.

In the long-term sequence, hydrogen is produced through radiolysis in the lower drywell. In the containment, there are three separate volumes where hydrogen can accumulate: in the wetwell, in the reactor vessel and in the upper drywell.

6.A.3.3 Prevention of overpressure of the containment

The containment pressure can be decreased by following systems:

- Containment filtered venting system.
- Contianment spray system
- Containment over-pressurization protection system and

Filtered pressure relief prevents slow overpressurization of the containment and additionally results in a sizable reduction in releases except for inert gases and organically bound iodine. The filtered pressure relief system consists of a water-filled scrubber connected to the upper drywell of the containment. If the pressure in the containment exceeds 5.5/6.0 bar (Forsmark

1 and 2/Forsmark 3), a rupture disc will burst and part of the atmosphere in the containment will be released via the scrubber.

The containment can also be relieved of pressure manually, for instance to reduce diffuse leakage. Spraying of the containment is used to lower the pressure and wash out airborne radioactivity in the form of aerosols. A redundant water supply to the spraying system does exist, which means that the system is available even during total blackout. In order to lower the pressure in the containment, spraying is preferred before filtered pressure relief, which results in the discharge of radioactivity outside the plant and thus limits accessibility. Spraying of the containment must not continue after the desired water level has been reached which, however, is about one metre above the vessel bottom.

In addition, there is a pressure relief system intended to protect the containment from overpressurization in the event of a large LOCA and simultaneous impaired PS function due to leaking diaphragm floor between drywell and wetwell. The system consists of a penetration in the containment fitted with a rupture disc that bursts if the pressure in the containment exceeds 7.5 bar. This causes an unfiltered release from the containment to the surroundings and therefore presupposes that the radioactivity in the atmosphere of the containment is very low. The system is equipped with two isolation valves that close automatically 10 minutes after containment isolation.

6.A.3.4 Prevention of re-criticality

During core meltdown, the control rods will melt before the fuel. This means that there will be a time interval when a large part of the core is intact but without control rods. If the core is reflooded during this time, re-criticality will occur. If the power level exceeds the decay power for which the filtered pressure relief has been dimensioned and designed for, the temperature in the condensation pool will rise. If this carries on for long enough, the heat sink can be lost and the containment overpressurized. In order to handle this situation, it is recommended to inject borated water and to limit the water injection mass flow rate below 500 kg/s. However, the core must be cooled to such an extent that no additional fuel damage occurs.

6.A.3.5 Prevention of basemat melt through

The strategy for preventing melt-through of the containment base plate is to fill the space below the reactor vessel with water. The space below the reactor vessel is automatically filled with water 30 minutes after isolation of the containment, the aim being to cool the core remains that may penetrate the reactor vessel in those events where the accident is progressing. Furthermore, the aim is to protect the containment from melt-through of the concrete slab itself.

6.A.3.6 Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity

The systems which are used to preserve the containment integrity, all have battery backup if the regular electrical power supply is not available when the systems are needed. The regular batteries last for two hours at nominal output, but may probably be utilised also after this period of time. The batteries which are credited for the indication of the scrubber last for 24 hours.

Even if battery backup is lost and/or the valves' pneumatic-manoeuvring systems stop functioning or become exhausted, the valves that require manoeuvring in the systems can be controlled directly using the gas bottles. The systems for direct depressurisation to the atmosphere and also for depressurisation via the scrubber are also equipped with passive rupture discs which automatically open upon reaching the blow-down pressure.

The system for spraying and filling of the containment is equipped with two diesel-backed pumps in the event of failure of the electrical power supply. If these pumps should not be available, the system can be directly fed with water from external pumps such as fire trucks. There is an additional portable diesel generator for the scrubber facility (24V) that can be connected in order to supply instrumentation and some emergency lighting. It is also possible to connect a mobile 380 V unit for supplying, among others the pumps required to regulate the level of the scrubber and also to circulate the water in the scrubber pool. Credit for the mobile diesel generators will be used after 24 hours, the time period for which the battery power supply is designed.

6.A.3.7 Measuring and control instrumentation needed for protecting containment integrity

The instrumentation in the containment is described in THAL. Existing measuring parameters for the containment are a) pressure, b) water level, c) temperature, d) radiation level and e) TV cameras. The tables in THAL give the quantity to be measured, location, measuring range and any additional information such as functional constraints due to the external environment (pressure, temperature, dose rate, water level).

6.A.3.8 Capability for severe accident management in case of simultaneous core melt/fuel damage accidents at different units on the same site

The operations needed for the management of severe accidents are described in emergency operating procedures. The actions needed are either automated or conducted by the operators in shift. Thus, in case of simultaneous accidents at different units immediate actions needed for severe accident management could be taken at each unit.

When all units simultaneously are subjected to severe disturbance it can be difficult to have sufficient number of competent personnel available. The KC is not organised to handle events on several units simultaneously.

Usually the training assumes that a unit is subjected to such serious disturbances that a release from the facility cannot be excluded, but never that all three units at Forsmark have similar events in parallel.

At an event where several units are affected at the same time one fire truck can only support one unit. Within the facility, there is only one fire truck available. Also that at an event where several units are affected, fuel chemicals, etc. may run out faster.

Some of the functions included in the emergency preparedness organization have so few personnel that only after a few days it would be difficult to sustain shifts around the clock, especially if more than one unit is affected. It is of great importance to secure resources as soon as possible by establishing shift lists for future staffing

The accessibility is impaired due to radioactive releases and high dose rates from multiple units. Difficulties in moving around the facility – all personnel must wear protective masks, etc. In summary, release affect all manual operations required.

6.A.3.9 Conclusion on the adequacy of severe accident management systems for protection of containment integrity

Since the systems for management and mitigation of severe accidents have already been implemented at Forsmark 1-3 and the corresponding procedures are in place, no further measures for this purpose are foreseen at the moment. However, the adequacy of the accident management schemes is being contantly assessed against the latest knowledge and experience obtained from different sources.

6.A.3.10 Measures which can be envisaged to enhance capability to maintain containment integrity after occurrence of severe fuel damage

At present, there is a risk of the gas turbine protection to be activated if energizing all three starting transformers simultaneously. If this occurs, manual operations is required and it is then uncertain whether the power supply to the ordinary feedwater and/or safety systems will be established before fuel damages have occured.

All existing systems for injecting water to the reactor vessel are dependent on an external power supply or a diesel backed-up net (possibly with the power supply from the gas turbine in Gunnarbo). In the event of a total loss of power (SBO), there is no possibility of injecting water to the reactor vessel, which was the basis used for the design of the consequence mitigating systems (FRISK). These systems are designed to mitigate the consequences of a meltdown with subsequent reactor vessel melt-through via filtered pressure relief and water filling of the containment. Whatever the background to the selected existing design, the possibility to use make-up of the reactor even at SBO, would definitely be an improvement measure.

During a lengthy sequence without a functioning containment cooling and with continuous, depressurization, the water level in the scrubber may need to be adjusted in order to ensure the proper filtration of the releases. At present, there is no action plan or strategy for how to achieve this.

Existing mobile generators at the plant for supplying the FRISK equipment in a long term scenario are not certified for earthquakes or placed/anchored in an earthquake safe place. The existing mobile generators are primarily intended to test the system, and in case of an emergency, reliance on community resources in the form of pumps is necessary. An inventory of what resources there are, where they are and how to requisition them should be made.

Need for analyzing which pressures the containment can handle using the best available calculation tools. Examine especially if there are effects of aging that lead to impaired function of the containment during a severe accident. Investigate how the function of the containment is working in the long term (years) after a severe accident.

It has been noted that the U.S. NPP has more radiation meters throughout the plant than we have in Sweden. There is a proposal to introduce more dose rate monitors at Forsmark.

6.A.4 Accident management measures to restrict the radioactive releases

6.A.4.1 Radioactive releases after loss of containment integrity

In the current situation, there are no accident management measures in place in the form of procedures etc, for loss of the containment function. The situation would have to be dealt with based on expert assessments. Spraying would reduce emissions but its effectiveness depends on the degree of damage to the containment. The reactor building contributes with a certain consequence mitigating effect by separation of aerosols, especially if the filtered ventilation can utilised. In addition, chemicals should be added to the containment with the purpose to achieving a pH of 10, thereby counteracting the release of iodine. A containment whose function has not been lost, but has been bypassed as a result of, e.g. systems connecting with the containment having been overpressurized, cannot be dealt with using the present accident management documentation either. This situation also have to be dealt with based on expert assessments.

6.A.4.2 Accident management after uncovering of the top of fuel in the fuel pool

The storage facilities for Forsmark 1-3 consist of three pools filled with water: two fuel pools and a cask pool. All the pools are water filled during normal operation. The temperature of the fuel pool is monitored by temperature gauges.

A conservative sequence for a loss of cooling scenario has been analyzed with the following assumptions: a) highest amount of decay power in relation to the volume of water in the storage pool, b) closed sluice gates, c) normal level in fuel pools and d) starting temperature in the pools: 40°.

In one case it is assumed that the entire core is unloaded out of the reactor vessel. After 4 days 100% of the core is located in one of the spent fuel pools. The level in the pool will drop, which results in that the required 2 m radiation shielding will not be lost until about 23 hours. To maintain a stable level in the pool (when boiling), a make up flow of 1,3 kg/s has to be provided to the spent fuel storage pool.

For Forsmark 1 and 2, make-up of fuel pools can be done with "System for storage and distribution of desalinated water". Since the pumps in the system cannot be credited for the event of a total blackout, a fire hose can be connected to that system instead. The hose is pressurized by a fire engine or mobile pump. Since access to the pools is not required in order to establish make-up in accordance with the above, it is judged that the measures can be carried out even after the point at which the radiation shielding is lost. These strategies could be utilized in the event of loss of radiation shielding, uncovering of the top of the fuel and degradation of the fuel. Strategies for cases with uncovered fuel are not available at present but should be produced with the purpose to improve Forsmark's management with loss of spent fuel pool cooling.

For Forsmark 3, the fire protection system is credited for filling the reactor containment in the event of severe accidents. If the system is available it can also be used to make-up of the pools with use of fire hydrant and connecting a hose to the desired pool. At present, no special measures are described for make-up of the fuel pools at Forsmark 3 in the event of loss of sufficient radiation shielding (in this case the reactor hall must be considered not to be accessible), at uncovering of the top of the fuel and degradation of the fuel. Since access to the pools is required in order to initiate make-up in accordance with the above, this measure is considered possible to be carried out before the point at which the radiation shielding has been lost. Even if strategies for the cases with uncovered fuel are not available at present (should be produced with the purpose to improve Forsmark's management with loss of spent fuel pool cooling) other relevant measures exist, depending on the accident sequence

6.A.4.3 Conclusion on the adequacy of measures to restrict the radioactive releases

The assessment has shown that strategies for cases with uncovered fuel in the spent fuel pools are not available at present but should be produced with the purpose to improve the management with loss of spent fuel pool cooling.

In the event of a total loss of power (station blackout), there is currently no system that can be used for cooling the spent fuel pools. Make-up of the pools is possible in accordance with existing procedures.

6.A.4.4 Measures which can be envisaged to enhance capability to restrict the radioactive releases

Strategies for handling cases with lost containment integrity should be developed as a complement to expert judgement.

In the event of a total loss of power (station blackout), there is currently no system that can be used for cooling the fuel pools. Make-up of the pools is possible in accordance with existing procedures. For Forsmark 3 access to the reactor hall is required for the establishment of the make-up. In the event of loss of shielding due to a low water level in the spent fuel pools, it may therefore be problematic to establish this make-up for Forsmark 3.

6.B. Oskarshamn

6.B.1 Organization and arrangements of the licensee to manage accidents

6.B.1.1 Organisation of the licensee to manage the accident

6.B.1.1.1 Staffing and shift management in normal operation

The shift personnel in the control rooms led by the Shift Supervisor are responsible for the facility during normal operations as well as during abnormal operations. During incidents affecting normal operations the Shift Supervisor calls for the Engineer on duty (VHI), who initially provides support in the assessment of the extent of the incident and the possible course of events. The VHI assesses the incident using established alarm criteria and calls for the support necessary in order to handle the further course of events and to minimize their consequences.

The Emergency organization is described in an Emergency Preparedness Plan. To support the Emergency Preparedness organization, procedures are compiled into six folders, one for each function/role respectively (VHI - Engineer on Duty, OL – Area Supervisor, AL – Plant Supervisor, SL – Radiation Protection Supervisor, SEL – Service Supervisor, IL – Information Supervisor) including support functions. Some procedures are contained in all folders, such as the telephone/alarm list, the procedure for assembly points, evacuation etc.

In case of disturbances, the Shift Supervisor always alerts the VHI who in turn decides, using criteria for actions in the VHI handbook, whether any alarm criteria has been met and whether the Emergency Preparedness management/organization should be alerted. The Shift Supervisor can also call in the Emergency Preparedness organization through BC (the security organization's Central Check Point) in an urgent situation.

The VHI is responsible for taking actions until the Area Supervisor (OL) arrives. The OL decides whether the Emergency Preparedness organization should take over the management of OKG or if the regular management should remain.

Continuous assessments of alarm criteria is conducted using procedures and action lists. Contact information to all personnel in the Emergency Preparedness organization is available in a procedure for information, reporting and alerting. The current Emergency Preparedness management consists of 60 persons, including 14 Engineers on Duty (VHI).

6.B.1.1.2 Measures taken to enable optimum intervention by personnel

The Emergency Preparedness Organization initially support the affected part of the organization and afterwards is able to take over the management for the purpose of being able to immediately initiate and take measures in order to

- prevent and reduce damage to people, facilities and the environment
- restore a damaged facility to a stable and safe condition
- reduce consequences and in connection with this take the required protective measures for personnel, facility and the environment
- inform/alert authorities responsible for third parties
- continuously provide support for the authorities' decisions
- predict the development of the incident
- keep the management group, owners, personnel, media and the general public up to date about the current situation.

In case of an accident, the Engineer on Duty arrives at the control room in question within 30 minutes. He will then decide if the Emergency Preparedness organization should be established. Everybody in the Emergency Preparedness management will then be alerted via pagers and will be informed of the events that have occurred. Seven individuals in the Emergency Preparedness management are formally on duty with a maximum reporting duration of two hours. For all other individuals in the Emergency Preparedness management, it is expected that one person per role reports on duty within eight hours from the alert.

Judging from the implemented call-on-duty tests, it is probable that those who are formally on duty will be at the on-site Emergency Control Center (KC) within approximately one hour. Remaining individuals in the Emergency Preparedness management will probably, if they can, also report on duty within the same period of time.

After 4-5 hours the requirements on personal protective equipment increase most dramatically for personnel who are reporting on duty. Both the KC and the substitute KC are equipped with filtered ventilation and the possibility to use auxiliary power supply. This decreases the need for personal protective equipment for the personnel staying there.

The current Emergency Preparedness support consists of approximately 90 persons. The operating shift personnel on their day-time schedule are part of this group. The Emergency Preparedness support contains appointed functions which constitute technical or operative support for the Emergency Preparedness management and the control rooms. The location during the occurrence of an event may be in the KC, in the control room or at the normal working place, depending on what is most suitable from the point of view of the situation having arisen. During the above described scenario, it might be possible that their own working place is not suitable from a radioactivity point of view, for which reason this may lead to an increasing load on the KC.

The task of the Emergency Planning support is to deliver basic decision documents to the Emergency Planning management. It is up to each function of the Emergency Planning management to call for its own Emergency Planning support.

The emergency preparedness resources must execute the decisions taken by the Emergency Preparedness management with the help of the Emergency Preparedness support. The work tasks which these resources are to perform in connection with an accident are closely related to the individual co-workers' normal work tasks, i.e. Chemists collect samples and analyze the result, Drivers conduct transports, etc. The Emergency Planning resources come on the first hand from the departments of Maintenance, Shared services, Administration and Human

Resources. Furthermore, resources from the security organization and the Rescue Services (ROK) are included.

The task of the Emergency Preparedness support is to identify and involve the co-workers who are needed to perform the work tasks which need to be performed because of the event that occurred. When these co-workers have been appointed they are participating as Emergency Support resources. The Emergency Preparedness support must also take into consideration future possible work tasks.

At station blackout the resources will probably be involved to reinstate the alternating current (AC) power supplies to the afflicted facility in order to re-start the safety systems needed in order to cool the fuel and the containment. The scope of the Emergency Preparedness support therefore may vary considerably. A first estimation for the above described scenario is that around 50-100 Emergency Preparedness resources will be involved during the first days. This estimation includes resources for food preparation.

6.B.1.1.3 Use of off-site technical support for accident management

All actions which other participants perform are led by the County Administrative Board and lie outside OKG's jurisdiction. The co-operation is well established and regularly trained.

The external assistance which OKG needs in the short-term perspective (up to a week) in order to bring the afflicted facility to a safe situation according to the above described scenario is:

1. Evacuation of remaining personnel.
2. Inward transportation of co-workers to the Emergency Preparedness management, support and as resources.
3. Inward transportation of shift teams to the control rooms.
4. Inward transportation of food deliveries to the personnel on duty.

In case of a radioactive emission, the inner zone around the nuclear power plant will be blocked, and receiving stations will be established by County Administrative Board (Länsstyrelsen). This will influence the prerequisites for the inward transportation of personnel as well as material and goods. A contaminated environment will probably place requirements for load transfers at police road block locations for the blocked inner emergency zone. Thus, items 2-4 which are normally OKG-issues become matters which must be coordinated by the Country Administrative Board and the Police authority.

In the long-term sequence of events, OKG may need external assistance in order to restore a damaged facility to a stable and safe condition. External resources will most probably be needed in order to decontaminate after a radioactive release on the Simpevarp peninsula to allow access to the power plant area. All measures performed by external participants and led by the Country Administrative Board are outside OKG's area of responsibility.

In the long run, OKG's own personnel as well as the contracted personnel will "come to an end" for different reasons, collected dose, tiredness, illness and so on. Then it is important to get hold of competent replacements from other sources. One might be forced to ask for help from the other Swedish nuclear power plants or from E.ON's German plants. The E.ON Group may assist in coordinating the resources internationally. Furthermore there is a contract of voluntary assistance between the nuclear power plants.

6.B.1.1.4 Procedures, training and exercises

The instruction packages for operational management of OKG's facilities are divided into operating and disturbance instructions. These are in turn divided up into several sub-groups. The following groups of emergency procedures are used to deal with disturbance situations within the design basis.

- System-specific emergency procedures (SSI)
- Unit-specific emergency operating procedures (ASI)

The following procedure group is used when the operating condition is already, or is at risk to be, outside the design basis.

- Overall disturbance instructions (ÖSI)

The ÖSI also includes the RAMA procedures. RAMA is an acronym for Reactor Accident Mitigation Analyses. The RAMA procedures are used when the actions in the ÖSI have not been effective and containment integrity is jeopardized because of melting through of the reactor pressure vessel (RPV).

The system-specific emergency procedures (SSI) generally have no table of contents, but are built around the alarms that the system can generate. The alarms are sorted in alphabetical order with respect to subdivisions.

For each alarm signal (miniature circuit breakers, measurement, etc.), there is a description of the consequences of, and probable cause for, the alarm. This is followed by suggestions for appropriate actions. If any special operating events or procedures shall be carried out as actions, references to this are provided.

The unit-specific emergency procedures (ASI) are built around defined and analysed disturbances in the facility operations. The ASIs contain emergency events initiated by the reactor protection system, as well as system events that affect the plant, for example, loss of cooling systems. The ASIs are designed around a defined structure consisting of a description of the operating condition, starting position, end position and sequence list.

OKG has performed basic training in emergency issues for all its shift teams as well as for the whole Emergency Preparedness organization. Refresher training for the shift teams has been developed with a double presence of the VHI in order to secure and train functional turnover between the regular line organization activity and the Emergency Preparedness organization.

A registered, reviewed and approved training plan and educational plans are established every year. Staff training is conducted for all functions in the Management of the emergency organization (BL) and the Security contractor's Central Check Point (BS). The training aims at improving internal and external collaboration in case of an incident that leads to the calling in of the Emergency Preparedness Organization, and the handling of applicable procedures and technical equipment in the KC. The training is carried out in the form of a role play with a target group, a counterpart acting group and a reference group, and may include collaboration with other organizations, such as the rescue services. Functional training is provided as well. All training and education is evaluated and the results are documented.

The County Administrative Boards in the nuclear counties conduct, at intervals of a few years, a major training exercise within the Nuclear Emergency Preparedness in Sweden. The counties take turns in arranging the exercise, which mainly focuses on testing the preparedness for an emergency situation at the nuclear power plant in question. The training is planned in collaboration with the neighbouring counties, other counties with nuclear power plants and the relevant authorities and other parties. The whole training is collaborative training where the field units in the society are trained as needed. On a regional level the training is regulated by the needs of each county and by financial or personnel resources. The aim of the training is to develop and strengthen the collaboration between the concerned parties. The County Administrative Board which is responsible for the training is also responsible for the planning, conducting and evaluation and experience feedback to the emergency preparedness operations. OKG participates in drawing up the training exercises and conducts coordinated training together with the County Administrative Board in Kalmar.

Staff training is conducted for all functions in the BL and BS. The training aims to improve internal and external collaboration in case of an incident that leads to the calling in of the Emergency Preparedness organization, and the handling of applicable procedures and technical equipment in the KC. The training is carried out in the form of a role play with a target group, a counterpart acting group and a reference group, and may include collaboration with other organizations, such as the rescue services.

VHI functional training is conducted regularly for all VHIs on duty. It is in the form of a role play with a target group and a counterpart acting group. The training aims to improve existing alarm procedures and the handling of instructions and technical equipment. The VHI also participates in simulator training with the control room personnel.

Functional training of the Radiation Protection Supervisor, aims primarily to improve the practical handling of the scattering tool LENA. In addition to the regular educational and training plan, further needs which may arise are taken into consideration. All training and education is evaluated and the results are documented.

6.B.1.1.5 Plans for strengthening the site organisation for accident management

There is an ongoing project and an investigation regarding the possibilities to release the signal "Immediate danger" from the BC and SBC as well as from the main control rooms.

A review is in progress which involves doubling of the number of assembly points and at the same time coordinating them with the fire alert assembly points. The gain will be a greater availability, less transports and a smaller number of persons who assemble at each assembly point. This also means a greater flexibility if one or more of the assembly points becomes unavailable.

An analysis is in progress regarding the use of the Emergency Preparedness switchboard and radio system. The idea is that the Emergency Preparedness switchboard will be used in order to maintain the contact with all the assembly points which each receive a card number and will be equipped with a radio.

An analysis is in progress about how to develop the use of RAKEL radio system, increase radio use with its own radio unit and a possible connection of RAKEL to the telephone network. This would facilitate the contact with the outside world during a situation when the normal communications have failed.

An investigation is suggested to ascertain advantages and disadvantages in replacing the present substitute KC with a suitable office outside the peninsula so that both KCs are not situated within the site where they would possibly both become affected by the same bad prerequisites.

6.B.1.2 Possibility to use existing equipment

The Emergency Preparedness organization has access to OKG's complete resources, i.e. material, personnel and finance. The KC and the substitute KC are chosen such that no single event should be able to jeopardize both of them at the same time. The KC is situated in a shelter and will thus not be impacted by a possible earthquake. In the event of flooding the KC could become water filled (because it is situated in a below-ground shelter) while the substitute KC is not impacted.

Because there are many alternatives, and alarms are an initial activity which is performed before the battery power is jeopardized, alarms are considered to be able to take place regardless of the power supply failing for the control room, the Remote Shutdown Panel or the extra surveillance place (RÖP), the BC and/or the on-site emergency control center.

Connecting auxiliary power to the KC is important. In the event that diesel engines and gas turbines (available for Oskarshamn 1 and Oskarshamn 2) are not available the situation is severe. Without the battery backed network, the KC is restricted to the available battery power (eight hours for communication equipment and available batteries in individual mobile telephones and portable computers).

The Emergency Preparedness activity is well equipped and will during the initial days not need equipment or other matters from the outside. When it then will be needed it is deemed that the initial hindrances may be removed alternatively that other supply channels have been established (via air, water, tracked vehicles and so on).

At the loss of the outer grid, the KC is dependent upon the existing connection of a mobile diesel in order to be supplied with power. Should this connection fail, problems will arise and one would only be able to put trust to the battery supply available for the communication system which lasts for approximately eight hours. If no power at all is available, the organization may consider establishing themselves in the substitute KC instead or elsewhere.

Support from the E.ON Group, other power plants, suppliers, authorities (County Administration Board, the Swedish Radiation Safety Authority, SSM, the Swedish Civil Contingencies Agency, MSB) may be counted on.

When outside support will be needed, it is deemed that the initial hindrances may be removed or alternatively that other supply channels have been established (via air, water, tracked vehicles and so on). Support from the E.ON Group, other power plants, suppliers, may be counted on.

Inside each facility's office premises, there are food vending machines and a freezer with frozen food packed in individual portions. In the KC, at each unit and in BC, there are individual portions of freeze-dried food. The restaurant Simpan has food in store. The sustainability with regard to food supplies is strongly dependent on whether or not the evacuation of personnel was successful. Contracts exist with suppliers for the diesel supply to the auxiliary power units.

It might possibly come to the situation that the available personal protective equipment is not sufficient. Then it might become necessary to get help from the Swedish nuclear power plants or E.ON's German plants. In Sweden there are about 200 "reactor filters", of which OKG has 50 and the remaining ones are distributed among the nuclear power plants. These may be transported to the afflicted facility within twelve hours. Further filters may be delivered from the supplier in Germany within 24 hours.

The communication solution chosen by OKG is sufficiently robust and even received a "Good practice" at the latest OSART-review. RAKEL will according to MSB function in all situations, and when all the interfaces (OKG, the County Administration Board, SOS Alarm, the Police, the ROK, E.ON and the SSM) have access to RAKEL, the information exchange should function at all times.

OKG has an internal mobile telephone net with 100 per cent coverage indoors which automatically switches over to the global external network if the telephone is brought outdoors. OKG-personnel have a personal OKG mobile telephone. The internal mobile network is operated by Telia as an integrated part of its network, for which reason only telephones with Telia's telephone services function within OKG's facilities.

OKG also has a Tetra radio communication system with an expanded inner network with 100 per cent coverage. Handsets are available in the control room for the requirements of Operations, in the BVB for the requirements of the security contractor as well as in the radiation protection office for remaining requirements. In the KC, the substitute KC, the BC and the Secondary Central Check Point for the security contractor (SBC) it is possible to communicate via radio in the RAKEL system as well as the 80 MHz system.

OKG has an internal loud speaker system with high coverage. The transmitting units are in the control rooms, the BC, the radiation protection office, the KC, the substitute KC and the SBC. The system can both transmit alarm signals and spoken messages. There are pre-recorded messages for abandonment both in Swedish and in English. The loud speaker system is complemented with speakers mounted on the security contractor vehicles. All communication systems have eight hours of battery back-up and will receive a stationary diesel back-up.

6.B.1.3 Evaluation of factors that may impede accident management and respective contingencies

In the case where it has been possible to establish an Emergency Preparedness organization, the demolished infrastructure is no direct threat for the decision making in the long-term perspective. If it has not been possible to establish a sufficient organization or if it is impossible to arrange for a turnover within a reasonable period of time, the decision making evidently will be influenced on a level with this loss. The demolished infrastructure will restrict the decision making to the effect that the obtaining of information may become influenced. Should it be the case that the Emergency Preparedness organization could not be established due to a demolished infrastructure, the control room personnel/shift team will become severely burdened.

If the evacuation of the facility has been unsuccessful, it would not be possible to provide the staff with food. If the evacuation is successful, the food supplies in store will be sufficient for the personnel. At high radiation levels, the staff of Simpan will not store food because there is no filtered ventilation at the restaurant facility. The possibilities of Simpan to receive, store, prepare and distribute food during a SBO event are almost non-existent.

Should a loss of communication capability occur, it would be difficult to get hold of personnel for arranging turnovers. Under the given prerequisite the personnel may be able to access the facility after 24 hours, if they are only informed that they are needed (after 72 hours also certain transportation possibilities exist). Even if the infrastructure is destroyed and OKG is a bit out of the forest terrain, it would appear reasonable to make use of boats, scooters or military vehicles to transport personnel to the plant day after day. Military vehicles have been used previously for shift relief when snowfall was great and the roads impassable.

It is possible to communicate the need of turnover personnel via radio from the KC to the Country Administration Board and others who can then forward this information to the persons in question in an appropriate manner (radio transmission, courier, etc). It is reasonable to believe that personnel are forced to remain in the facility until replacement has been arranged. Regular rest and food intake is possible at the site. After 72 hours it is possible to arrange replenishment of the food supplies. A long-term loss of voltage will however create problems. When diesels and batteries have been consumed the information handling and the communication will be made considerably more difficult as one is restricted to manual calculations.

In order to be able to work in the KC for a longer period of time you are depending on a certain comfort. If the water and sewage systems do not function you are obliged to use the available dry privies. Water for cooking and drinking is judged to be sufficiently available with the site. The restricted water availability will however decrease the possibility of decontamination.

In case the facility is afflicted with radioactive releases the ventilation of the control room protects well against airborne activity. The ventilation systems for the CKR, the RÖP, the KC, the substitute KC i.a. are prepared with filtering possibilities in case of a radioactive release. There are procedures describing how to manage during special circumstances.

With high doses on the site it is not possible for anyone to go/walk either to the emergency center or to the secondary emergency center without proper personal protective equipment. In the long run, the turnover will also be influenced by the fact that personnel leaving the facility will take their personal protective equipment with them, and alternatively it will be "consumed". The used protective equipment should be left at a designated location suitable for the purpose to enable re-use, but also to ensure that possible contamination does not leave the area and be spread.

Personnel who need to access the facilities will need both personal protective equipment and dosimeters at the checking points in the inner zone. OKG keeps personal protective equipment inside each facility. The protective equipment may need to be transported to the road block locations. The judgment is that the available personal protective equipment and the dosimeters are sufficient for the personnel who may need to work at OKG.

The police and/or ROK provide their own protective equipment. For other external professionals, OKG will provide protective equipment. With regard to the ROK which may need to contribute, they have breathing aids with forced ventilation masks.

The personal thermo-luminescent dosimeter should be used by all those who are able to access their dosimeters. For personnel who are being evacuated, somebody in each group should be equipped with an electronic personal dosimeter in order to register the group's dose. The availability of personal protective equipment may become a limiting factor in the long run as it might need to be decontaminated regularly – if equipment is not obtainable from somewhere else.

Deliveries of equipment, food and ready-cooked food are made difficult however by the increased radiation levels at the facility. In such cases load transfers and personal protective equipment as well as a dosimeter are required for the driver. Another aggravating circumstance is also the increased radiation level at the site which makes transfer of the food difficult at Simpan.

OKG has both a normal KC and a substitute KC. They are both similarly equipped with regard to communication equipment and documentation. Both the KC and the substitute KC have filtered ventilation with the possibility of auxiliary power supply. Coal filter ventilation is in place. The substitute KC is however situated on the fourth floor (in the same building as the control room for Oskarshamn 1 and Oskarshamn 2) and would therefore withstand flooding. Should the facility however be affected by an earthquake, the suitability may however be the reverse. Both withstand an earthquake beyond design.

The normal KC will with great probability withstand earthquake scenarios beyond design better than the substitute KC. The substitute KC is supplied with uninterruptible (battery backed) power. The batteries are expected to last for four to five hours. The KC will get power from independent emergency power supplies (a diesel-operated back-up power supply). It may take 20- 60 minutes before auxiliary power is connected to the KC. At a SBO therefore the substitute KC will be a more sustainable alternative.

An aggravating circumstance for the substitute KC is that during the duration of the depressurization of the containment taking place at Oskarshamn 1 or Oskarshamn 2 via the scrubber system, the personnel should leave the substitute KC because its design does not provide sufficient protection against direct radiation from a possible plume. Without CKR, the possibilities to maintain overview and control over the process and the incident progress will decrease. Supervision is still possible from the KC/substitute-KC and the BC/SBC. Maneuvers can only be performed from dedicated control centers in the plants.

There are portable emergency lights available in the KC in order that the establishment of the organization may function during the duration when the auxiliary power supply (a mobile diesel engine) will be switched on. Procedures and action lists are available on paper and are kept up-to-date continuously as per special routines. The portable computers available have batteries of their own and mobile internet connections. The remaining computers do not function until the auxiliary power supply has been switched on.

The regular assembly point for the Emergency Preparedness management is the on-site emergency control center (KC) which is in the fallout shelter adjacent to Oskarshamn 3.

The potential effects from the other neighbouring are that the more units afflicted the greater the load on the emergency organization. If it is necessary due to heavy work load, one or several of the functions will be doubled-manned, otherwise it will be the same staffing. It can be conceivable in the worst case that the possibilities of efficient work and good decision making can become jeopardized. The more units that are affected, the more personnel are expected to be needed in the facility and the more protective equipment and the more dosimeters are needed. This is also the case for more food supplies and other supplies. OKG's own equipment may become out of stock and new equipment may need to be ordered from the supplier, the other Swedish nuclear power plants or E.ON's German plants.

In the long term it may become interesting to increase the co-operation between the Emergency Preparedness organization of Clab and OKG respectively in order to co-ordinate the need of action from society. Several of the issues will then probably be common.

6.B.1.4 Conclusion on the adequacy of organisational issues for accident management

The list below illustrates the procedure status which has been identified due to the stress test.

- The situation of establishing the Emergency Preparedness organization with a demolished infrastructure has not been highlighted in the available procedure package.
- There are procedures for assembly points, abandonment, establishment of the Emergency Preparedness organization, turnover, evacuation, etc. These procedures do not currently take into account high radiation levels and for demolished buildings.

There is a reminder in the VHI-procedure to consider the situation with regard to protection in exit transportations when there is a danger of a release, or a possible release, and to communicate this to the alerted personnel.

- The situation with the alternative means of communication is highlighted in the procedure package, as are calculations and measurement collection.
- The procedures concerning food supplies should be reviewed from the point of view of possible scenarios.
- The procedure package describes the internal and external contacts which must be established at an extraordinary event, but not the approaches to attract attention from the community when all communication systems are out of order.
- An agreement exists for establishing an alternative KC outside the power plant area, but the procedures do not cover this.
- The Emergency Preparedness plan describes the KC and its alternatives.
- ROK has procedures for connecting diesel power to the KC.
- There are procedures available for the handling of IT-disturbances, IT-support and manual processes.
- There are procedures available for informing/alerting Clab in case of an event at OKG and informing/alerting the general public and the E.ON Group.
- There are contracts of co-operation existing both nationally and internationally.

6.B.1.5 Measures which can be envisaged to enhance accident management capabilities

In the event of failure to establish an emergency organization, the control room staff / shift team are heavily loaded and rely on themselves to manage the necessary measures. Then it would be extremely valuable to have additional skills / personnel permanently on site (preferably at each unit). Maintenance personnel in form of electrical- instruments- and mechanical competence but also radiation skills are requested. The most likely option is to complete the shift team with an electrician.

There are needs to:

- Further develop the procedures and routines in force and also issue new ones regarding, i.e. personal protective equipment in the event of releases, either imminent or on-going.
- Clarify the form for the co-operation and division of responsibility between OKG and the Police regarding decontamination stations outside the power plant perimeter for personnel during shift turnovers and how equipment is to be replaced/taken care of. Consider the course of action during an extended need of personal protective equipment for a long-term duration.
- Prepare procedures for radiological checks and decontamination at the boundary of the inner zone.
- Identify alternative evacuation routes. It might be better not to start abandonment immediately. If there are no roads the rescue leaders must investigate the possibility of cross country, sea or air transportation. This scenario should become highlighted and preparations possibly should be made.
- Analyze the need for extra personal protection equipment. Consider purchasing a large number of readily accessible raincoats and breathing aids to be used during an extensive evacuation.
- For the turnover personnel – issue prepared suggestions of alternative transport routes to the facility.
- Develop existing instructions and procedures and develop new respect among other protection equipment when threatened or pending release.

An analysis is in progress regarding the installation of card readers for registering of personnel at the assembly points. This will in such a case facilitate the compilation of the personnel situation when evacuating the power plant area. Other ways of positioning personnel are also analyzed.

Plan for a location off-site where staff can be equipped, dosimetry can be performed, the distribution of safety equipment may occur, etc. Prepare features for radiological monitoring and decontamination of the internal evacuation zone boundary Analysis is underway to solve the problem of not enough personal protective equipment.

Check through existing extra food supplies and supplement with bottled water so that the food supplies will last for the personnel who are intended to be needed at the site.

A stationary diesel engine will be installed according to plan at the end of 2012. Access to more portable computers should be considered.

A new system for measurements (a condensation of the existing permanent measuring points in Sweden around the nuclear power plants) is being installed by the SSM.

A project is in progress to develop and implement the MARS-tool in order to receive more complete information for the LENA-calculations. MARS gets data from the DRUS (or manually) and tries to diagnose the event and also predict the continued course of events (a type of AI-interface between the MAAP and the DRUS). If the MARS-tools is introduced into the Emergency Preparedness activities, this may facilitate different "what-if" analyses beyond design.

If more than one unit is affected, the emergency preparedness organization continues under its instructions. Access to common resources is limited, however, which places greater demands on establishing priorities.

During training exercises, it can be valuable to develop and practice scenarios where most of the units are affected. This increases the readiness to act, and ultimately the plans and procedures will be improved.

6.B.2 Accident management measures in place at the various stages of a scenario of loss of the core cooling function

During events where the core cooling function has ceased, the accident strategy follows actions in accordance with the plant's symptom-based ÖSI sheets. The strategy in this case focuses on maintaining control of the reactor's reactivity and on restoring the electrical power supply in order to be able to utilize the plant's various water and cooling systems. The focus is also on mitigating the consequences of an accident by protecting the containment and minimising releases to the environment. The actions that are taken during the accident are performed with the support of the ÖSI:s and the RAMA and depend upon what caused the accident and the accident sequence.

6.B.2.1 Before occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes (including last resorts to prevent fuel damage)

The ÖSI sheets are processed according to a symptom-based control structure, i.e., the symptoms of reactivity should be prioritised before the level in the RPV and so on. In reality, considerable resources are applied to work in parallel in several ÖSI schedules.

The activities in accordance with ÖSI are designed to prevent global and/or local criticality by inserting all control rods into the core and injecting a boron solution. The activities are focused in accordance with ÖSI on restoring one or more power supply paths to thereby make it possible to use the normal systems for control of the facility.

Restoration of some form of water injection into the reactor pressure vessel through available systems is a priority. These measures according to ÖSI require electrical power supply and also the recovery of the associated auxiliary systems such as cooling of the condensation pool, etc. In an emergency, systems that are not normally used for water injection to the reactor pressure vessel can be used. However, even these systems require electrical power supply to be available. This also includes depressurization of the reactor pressure vessel to achieve injection with low pressure core cooling system, as well as measures to prevent high pressure core melt in a later stage. Both depressurizations can be performed with only battery power available. The batteries have a capacity of two hours at nominal output, but may probably be utilised after this period of time.

Activities are aimed at securing some form of heat sink for the reactor's heat content. The primary focus is on ensuring the cooling of the condensation pool in accordance with the ÖSI. With failure of the main heat sink, Oskarshamn 1-3 have a diversified residual heat removal chain with the advantage that residual heat removal occurs without mass loss from the RPV. However, this diversified residual heat removal requires a diesel-backed power supply.

As previously described above, there are no special systems that can be used for dilution feeding and/or cooling of the reactor if all electrical power supplies fail.

According to the ÖSI, steps have been taken to prepare the facility for the possible occurrence of a core melt. As a consequence mitigating measure for this scenario, the pressure is reduced inside the reactor pressure vessel by motor-operated pressure relief valves that open automatically at low level by using battery-supplied AC power. The batteries have a capacity of two hours at nominal output, but may probably be utilised after this period of time. These motor-operated pressure relief valves are dedicated to this task and will remain open after they have opened. Alternatively, the safety relief valves and automatic depressurization valves can be opened, the governing valves of which require battery supplied DC power.

Pressure reduction occurs by the activation of the pressure relief chain, which opens all eight electrical motor operated pressure relief valves, and the automatic depressurization chain, which opens all eight safety relief valves. In this situation, these safety chains will trip automatically on low level. If these safety chains do not function, manual opening is performed from the main control room. The actions are performed in accordance with the ÖSI.

6.B.2.2 After occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes

The time available before melting through the reactor pressure vessel occurs is, in the worst case, estimated to be about four hours. The work concentrates on restoring the electrical power supply and thereby also the dilution feeding and cooling of the reactor. In parallel, a continued control of the ÖSI takes place as per above with the aim of controlling the status of the reactor (reactivity, pressure and level). This means that the measures primarily follow the design case and are directed according to the following strategy:

Troubleshooting with respect to dilution feeding systems continues until a RPV melting has been observed. Filling of the lower drywell by opening of the valves from the condensation pool is part of the preparations for the melting through of the RPV. This measure, which is in accordance with the ÖSI, means that the area under the RPV is filled with water and that the molten metal ends up in the water at a later stage. This action is performed automatically when the level is less than 0,5 m above the core for 10 min and if this action is not performed automatically it will be performed manually. Spraying of the drywell has the goal of keeping the pressure and temperature at low levels, and to wash out any radioactive particles and aerosols. This contributes to minimizing any potential spreading of activity from the containment. For the case where all the electrical power is lost, the operators try to accomplish sprinkling of the containment using system 322-independent.

6.B.2.3 After failure of the reactor pressure vessel/a number of pressure tubes

When the melting through the reactor pressure vessel occurs, only one action as shown below is performed under the ÖSI prior to the transition to the RAMA accident instructions. This action means that continued measures to cool the reactor pressure vessel using normal safety systems is stopped. The accident strategy primarily focuses on the mitigation of consequences on the environment in accordance with the RAMA instruction. This leads to actions that follow the design case and are focused as follows:

Control and closure of any open isolation valves in accordance with the ÖSI has the purpose of minimizing the spreading out of radioactivity from the reactor containment. The isolation valves that are not closed by the II-signal are closed at this time. Water filling of the containment is performed up to a level corresponding to the top of the core in accordance with actions in the RAMA instruction. This is expected to start eight hours after the initiating event.

Filling of the containment is accomplished by spraying into drywell using water injection that is not dependent on the availability of electrical power (i.e., system 322-independent). During filling the containment, pressure and temperature are maintained at a low level. Airborne activity is washed out of the containment atmosphere and is bound to water in the containment during the water filling process. Manual depressurization of the containment's steam to the scrubber is activated, according to RAMA-instruction, before automatic relief from the rupture disc occurs. The purpose is to protect the containment integrity and to prevent uncontrolled radioactive releases into the environment.

Residual heat removal through the scrubber is established and the water level in the scrubber is adjusted. This is performed using mobile generator as electrical power supply. There are prepared connections at the scrubber for connecting 24V DC and also 380V AC from mobile generators. 24V is sufficient for instruments, while 380V is required in order to pump water to and from the scrubber. When the electrical power supply has been restored, the accident strategy will focus on establishing permanent residual heat removal of the reactor and containment is established through the normal residual heat removal systems.

6.B.3 Maintaining the containment integrity after occurrence of significant fuel damage (up to core meltdown) in the reactor core

This section describes selected accident management actions for the protection of the containment integrity and functions after fuel damage has occurred.

During normal operation, the plant's containment is nitrogen-filled, which means it is inert and oxygen levels are kept within the range of 0.8 to 0.5 per cent using the permanently installed recombiner. The safety-relief valve discharge lines from the reactor pressure vessel to the condensation pool are vented with nitrogen gas flow to remove possible hydrogen gas from leaking steam during operation. In addition, there are permanent recombiners mounted within the control valves to the safety-relief valves, so the risk for hydrogen deflagration or hydrogen detonation is negligible. During start-up of the plant after an outage and during the plant shutdown process before an outage, there are short, controlled periods of time with an air-filled containment. At these times, the conditions exist for a hydrogen mixture in the containment to become flammable. However, these are relatively short times, which mean that the risk contribution is small. During the facility shutdown process and prompted by a possible disturbance, the containment remains nitrogen-filled until a decision is made to change the atmosphere to air. This atmospheric change only occurs if facility personnel need access to the containment.

6.B.3.1 Elimination of fuel damage / meltdown in high pressure

According to the ÖSI, steps have been taken to prepare the facility for the possible occurrence of a core melt. As a consequence mitigating measure for this scenario, the pressure is reduced inside the reactor pressure vessel by motor-operated pressure relief valves that open automatically at low level by using battery-supplied AC power. The batteries have a capacity of two hours at nominal output, but may probably be utilised after this period of time. These motor-operated pressure relief valves are dedicated to this task and will remain open after they have opened. Alternatively, the safety relief valves and automatic depressurization valves can be opened, the governing valves of which require battery supplied DC power.

Pressure reduction occurs by the activation of the pressure relief chain, which opens all eight electrical motor operated pressure relief valves, and the automatic depressurization chain, which opens all eight safety relief valves.

6.B.3.2 Management of hydrogen risks inside the containment

As in the case during normal operation, the design of the facility is such that the containment remains nitrogen-filled with control of the oxygen content during an accident. Should a situation arise where the permanently installed recombiners are not able to keep down the oxygen level, a portable recombiner is available. It is connected to the containment atmosphere through connections outside the containment. For valves and safety-relief discharge lines where past experience has shown the danger of hydrogen accumulation, the systems are equipped with permanently installed platinum recombiners and/or flushing with nitrogen.

Hydrogen production in connection with metal-water reaction will cause the pressure in the containment to rise and results in an automatic or manual depressurization via the scrubber. The pipes that can be used during depressurization are nitrogen-filled and flushed continuously with a nitrogen gas flow in order to keep oxygen away. After a depressurization, the nitrogen gas flow continues to flush the pipes and a forced flushing will be manually initiated after blow-down, which prevents hydrogen deflagration or detonation from occurring. The strategies shown in the ÖSI sheets are as follows:

- Start of the permanent recombiner occurs 90 minutes after the containment isolation is activated in accordance with the action sheet for high pressure in the containment as a preventative measure.
- The portable recombiner is ordered and prepared for installation according to actions in the PRI-sheet (pressure in containment) and the unit-specific emergency

procedures. This recombiner is calculated to be in place and connected within 24 hours.

6.B.3.3 Prevention of overpressure of the containment

The containment pressure can be decreased by following systems:

- Containment filtered venting system.
- Containment over-pressurization protection system
- Contianment spray system

The systems to prevent over-pressurization of the containment basically consist of two systems: one where releases occur directly to the atmosphere (design sequence A) and one that depressurizes the containment through a scrubber (design sequence B).

In design sequence A, the system for direct relief is intended for use at the beginning of an accident when the containment pressure increase is due to the escaping steam-water mixture at high pressure and temperature and simultaneous malfunction of the diaphragm floor. The rupture disc to the environment bursts at 0.6 MPa for Oskarshamn 1 and at 0.65 MPa for Oskarshamn 2-3. At this stage, any fuel damage has not yet occurred so a discharge can occur to the atmosphere without subjecting the environment to any high doses. After depressurization directly to the atmosphere is completed, valves (battery backed power supplied) automatically will close (20 minutes after containment isolation is activated). Also the rupture disc to the other system has at that time broken and any manual closing of the series valve to the rupture disc is closed. Spraying with water can then start in the containment in order to wash out the containment atmosphere, which causes a simultaneous pressure reduction by condensation of steam.

In design sequence B, the entire emergency cooling of the core with regular safety systems is assumed to not function. This leads to that the pressure in the containment will increase. If manual depressurization does not occur, automatic depressurization will take place at 0.5 for Oskarshamn 1-2 and at 0.6 MPa for Oskarshamn 3 by a rupture disc. The scrubber precipitates possible radioactivity content in the containment atmosphere and collects it in a water basin. An exemption applies to radioactive noble gases, which will pass through the scrubber to the atmosphere. The radioactive material that reaches the environment will not cause death by acute radiation sickness, nor will it cause ground contamination that could prevent long term use of large land areas. The systems for pressure relief of the containment have pneumatic valves that are fed from the compressed air system for priority needs. Gas bottles serve as a backup to both these functions if this system malfunctions.

Depressurization by means of compressed air is a complement to the installed rupture disc. The system for spraying the containment also has pneumatic valves that are fed from the compressed air system. Also here there are gas bottles serving as a backup. All the equipment which is credited is available at the RAMA main manoeuvring point, from where the handling takes place. The strategies that are employed are shown in the RAMA procedure as follows:

- Manually controlled depressurization with filtering through the scrubber prior to automatic depressurization through the rupture disc.
- Spraying the containment is started in order to wash out the atmosphere and reduce the pressure. The amount of water pumped in is limited to a level corresponding to the top of the core.

6.B.3.4 Prevention of re-criticality

During the accident sequence there can be a time window where the control rods are fully or partially melted but the fuel rods remain intact. If a reflooding with water should occur within this time window, it could cause re-criticality to occur, but this is considered very unlikely. An accident strategy to combat re-criticality is to only refill the core with water so that a level is reached in the reactor pressure vessel where the core will be cooled by the rising steam from the boiling water. This means a level just above the bottom of the core. Today there are no procedures that handle this situation. The accident strategies that can be implemented are:

- Injection with a boron solution, if the electrical power supply is available, in accordance with normal system procedures for the boron injection system. This strategy can only be used for Oskarshamn 3.
- Limitation of the dilution feeding to a level corresponding to the bottom of the core. The aim is to limit the water's moderation capabilities while boiling occurs and a cooling effect is created by the rising steam. Injecting water and maintaining the level occur in accordance with normal procedures provided the electrical power supply is available.

6.B.3.5 Prevention of basemat melt through

Due to different configuration of the contianments in Oskarshamn 1-3, the strategies for prevention of basemat melt through may differ in the accident sequence.

For Oskarshamn 1, the space under the reactor pressure vessel is equipped with a penetration which melts when it is hit by the melt from the reactor pressure vessel. This entails that the mixture of metal and fuel material which continued down into the water pool beneath and split into minor parts. A heat sink is created for the reactor pressure vessel by establishing cooling of the condensation pool through the activation of an electrical power supply independent system, which sprays and cools the containment and thereby also the condensation pool. Actions occur primarily in accordance with the ÖSI.

For Oskarshamn 2, the condensation pool has a volume of 2 000 m^3 and with an area of 322 m^2 entails that the melt is cooled and can remain cooled at the bottom of the reactor containment. When melting through the reactor pressure vessel, the melt will dwell for a short time in the control rod drive pit before it finally flows downwards and is split into minor parts in wetwell. No preparations are required to secure this course of accident sequence. Only continued surveillance of the level and temperature of the heat sink is performed during the course of the sequence.

For Oskarshamn 3, in an accident situation where isolation of the containment has occurred and the level in the reactor pressure vessel is very low, an automatically initiated filling occurs in the space below the reactor pressure vessel (lower drywell). The filling occurs through the opening of valves from the condensation pool. This entails that the space beneath the RPV becomes water-filled and that the melt at a later stage will end up in water and there be cooled by thermal circulation. The measure takes place automatically or according to the ÖSI. The filling is initiated when both low levels L3 and L4 are present for greater than 10 minutes. Initiation of this function requires battery-backed power. The system's pneumatic valves are fed from the compressed nitrogen system. As a backup, there are gas cylinders for manoeuvres during manual operation.

6.B.3.6 Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity

The systems that are aimed to preserve the containment integrity, all have battery backup if the regular electrical power supply is not available when the systems are needed.

The regular batteries last for two hours at nominal output, but may probably be utilised also after this period of time. The batteries which are credited for the indication of the scrubber last for 24 hours.

Even if battery backup is lost and/or the valves' pneumatic-manoeuvring systems stop functioning or become exhausted, the valves that require manoeuvring in the systems can be controlled directly using the gas bottles. The systems for direct depressurisation to the atmosphere and also for depressurisation via the scrubber are also equipped with passive rupture discs which automatically open upon reaching the blow-down pressure.

The system for spraying and filling of the containment is equipped with two diesel-backed pumps in the event of failure of the electrical power supply. If these pumps should not be available, the system can be directly fed with water from external pumps such as fire trucks. For Oskarshamn 1-2, it is also possible to connect the diesel-secured fire water pumps for Oskarshamn 3 to the circular network for fire extinguishing water.

There is an additional portable diesel generator for the scrubber facility (24 V) that can be connected in order to supply instrumentation and some emergency lighting. It is also possible to connect a mobile 380 V unit for supplying, among others the pumps required to regulate the level of the scrubber and also to circulate the water in the scrubber pool. Credit for the mobile diesel generators will be used after 24 hours, the time period for which the battery power supply is designed.

6.B.3.7 Measuring and control instrumentation needed for protecting containment integrity

Existing measuring parameters for the containment are a) pressure, b) water level, c) temperature, d) radiation level and e) TV cameras.

6.B.3.8 Capability for severe accident management in case of simultaneous core melt/fuel damage accidents at different units on the same site

The operations needed for the management of severe accidents are described in emergency operating procedures. The actions needed are either automated or conducted by the operators in shift. Thus, in case of simultaneous accidents at different units immediate actions needed for severe accident management could be taken at each unit.

When all units simultaneously are subjected to severe disturbance it can be difficult to have sufficient number of competent personnel available. The on-site emergency control centre is not organised to handle events on several units simultaneously.

6.B.3.9 Conclusion on the adequacy of severe accident management systems for protection of containment integrity

Since the systems for management and mitigation of severe accidents have already been implemented at Oskarshamn 1-3 and the corresponding procedures are in place, no further measures for this purpose are foreseen at the moment. However, the adequacy of the accident management schemes is being contantly assessed against the latest knowledge and experience obtained from different sources.

6.B.3.10 Measures which can be envisaged to enhance capability to maintain containment integrity after occurrence of severe fuel damage

For Oskarshamn 1-3, no additional measures were identified.

6.B.4 Accident management measures to restrict the radioactive releases

6.B.4.1 Radioactive releases after loss of containment integrity

This section describes selected accident management measures to mitigate the consequences of a loss of the containment function.

In the event that the containment integrity can no longer be maintained, the reactor building is the next barrier to the environment. A continued sprinkling of the containment atmosphere by the power supply independent system holds down the amount of airborne activity that could leak out. Depending on how much leakage occurs from the reactor containment, an isolation and emergency filter ventilation of the reactor building through carbon filters will occur to prevent or limit the release of activity into the environment.

The accident strategy for suspected or confirmed reactor containment leakage is based on the ÖSI and means:

- Re-inspection and closure of any open isolation valves.
- Rechecking the isolation of the reactor building by control of closed doors, locks, blowout panels, etc.
- Rechecking that the transition to filtered ventilation for the reactor building via the charcoal filter has occurred. (This transition is automatically initiated at the signal for the isolation of the containment.)

6.B.4.2 Accident management after uncovering of the top of fuel in the fuel pool

This section describes selected accident management measures to mitigate the consequences of the loss of cooling of the fuel pools.

The management of problems with loss of cooling of the fuel pools is handled through unit-specific emergency procedures. There are predetermined strategies for dilution feeding and cooling if the ordinary systems should fail.

Accident strategies are therefore developed from these instructions and are made up of assessments and decisions from the collective expertise of the emergency preparedness organization.

Depending on the operational condition of the station when the fuel pool cooling systems are postulated inoperable the duration differs for how rapidly the water boils off in the pools.

During power operation the accumulated residual power in the pools is low, therefore the time duration until boiling occur will be long. For the time periods found below the difference is not remarkable and this is because the pool locks are closed during power operations. A simple measure is to open the locks during the initiating stage (within 24 hours) whereupon a larger volume of water will be heated which will provide more time to perform measures.

During outages with a totally unloaded core the residual power is high and thus the attendance time to perform measures is relatively short. During outages the reactor hall very often is manned which increases the likelihood of discovery and possibility of rapid measures.

A calculation of the available time in the worst scenario, i.e., shortly after the start of an outage shows that there are about 18 hours for Oskarshamn 1, 40 hours for Oskarshamn 2 and 21 hours for Oskarshamn 3 available before the pool temperature reaches 100°C where action should be taken. Measures may also be taken after 100°C, but then the working environment has deteriorated due to the boiling in the pools and breathing protection, which is kept near the main control room, may be required. If also the duration with adequate radiation protection is taken into account (2 m water above top of fuel), about 91 hours for Oskarshamn 1, 6 days for Oskarshamn 2 and 74 hours for Oskarshamn 3 are available to take measures.

In the event that fuel pool cooling has ceased and the water has boiled off such that the water's radiation shielding capability is reduced, the pools can be dilution fed through a pre-identified fire water supply.

Depending on activity levels in the reactor building, various connection points are identified for fire water make up supply via fire trucks.

The accident strategy for suspected or confirmed case of no cooling of the fuel pools is based on the established emergency procedures and means that fire hoses are connected to pre-identified hydrants and dilution feeding is performed via fire hoses directly to the fuel pools.

Mobile pump units are acquired through the fire brigade if the regular electric or diesel driven fire pumps are not available. These are connected to the existing connections. The water supply can then be accomplished by:

- Tanker truck
- Water from Söråmagasinet using pumps with gas turbinebacked power supply.
- Water from Götemaren assuming that the regular power network functions.
- Water directly from the sea
- Fire hose from mobile pump to the fuel pool.

If the accident scenario has reached the point where the upper parts of the fuel are exposed in spite of dilution feeding water via the fire water supply, no pre-discussed accident strategies have been developed. The spent fuel pool area is not likely to be accessible from an environmental perspective among others from a radiation protection perspective. The actual situation will determine the accident strategies that are selected, which will consist of assessments and decisions from the collective expertise of the emergency preparedness organization.

6.B.4.3 Conclusion on the adequacy of measures to restrict the radioactive releases

In the event of a total loss of power (station blackout), there is currently no system that can be used for cooling the fuel pools. Make-up of the pools is possible with use of fire water. If the shielding is lost due to a low water level in the spent fuel pools, it may be problematic to establish this make-up.

6.B.4.4 Measures which can be envisaged to enhance capability to restrict the radioactive releases

Management and mitigation of severe accidents have already been implemented at Oskarshamn 1-3 and the corresponding procedures are in place. Especiellay, considering measures to restrict radioactive releases in the event of a severe accident, the containment filtered venting system is a major asset. This would minimize the release into a small fraction of the initial activity released into the containment. Thus, no further measures for this purpose are foreseen at the moment.

For makeup water supply to the spent fuel pools during severe accidents, a solid pipeline from the protected location should be installed. This should be seismically qualified. In addition, some type of robust possibility to read the level in the fuel pools from a protected location should be considered.

6.C. Ringhals

6.C.1 Organisation and arrangements of the licensee to manage accidents

6.C.1.1 Organisation of the licensee to manage the accident

6.C.1.1.1 Staffing and shift management in normal operation

During normal operation, the following staff is available at a minimum:

- Shift team on duty
- Engineer On Duty (VHI)
- Shift radiation protection engineer
- Guards, as well as fire fighters (Group leader + 3 fire fighters).

During transition from normal operation to a threat against safe operation, an alarm phase commences, initiated by the shift supervisor at the affected unit. Further actions are directed by how the event is classified and can include information/requests on protective measures for power plant personnel, as well as contacts and information to the general public.

The decision on alarm level, based on alarm criteria, and thereafter the Engineer On Duty (VHI) takes the subsequent activation of the emergency response organization after consultation with the shift supervisor. The Engineer On Duty is responsible for the initial measures governing declaration of the alarm level. The Engineer On Duty always has the operational responsibility, regardless of time of day and location, until it is turned over to the Site Emergency Director (OL). The Engineer On Duty carries out the Unit Manager and Site Emergency Director's actions until called-out personnel arrive at their posts. When the emergency response organization is in place at the power plant it takes over the responsibility after a turnover from the Engineer On Duty.

The security centre's task is to alert predetermined external authorities and to call in personnel to the power plant's internal emergency response organization. Security personnel have at their disposal a computerized call up system (Rapid Reach) to help with this task. The system searches for people in the emergency response organization via office telephone, mobile telephone and home telephone.

A Unit Manager (BL) and a multidisciplinary Technical Support (TS) representative are established in the centre control room on the affected unit. There are nine people who have the task of supporting the shift team on duty in their work of returning the plant to a safe condition.

There is also an On-site emergency control centre (KC) at the plant. People are assembled there in thirteen different functions to support the affected unit from a strategic perspective, allocate resources, handle issues that affect the entire plant and communicate with the authorities.

6.C.1.1.2 Measures taken to enable optimum intervention by personnel

Until the emergency response organization has been established, the staff has to perform initial actions such as call out the emergency organization, manage the unit based upon its emergency instructions, issue internal alarm at the plant, classify the event, issue external alarm to external preparedness functions (authorities) and people living in the vicinity, supply information to external and company internal emergency functions, perform rescue missions and fire fighting, etc.

The main focus of this group is keeping the core cooled, save lives and to call for resources in order to handle an extensive event. The staff in the emergency response organization comes from Ringhals organization and work in 23 different functions. There are 2-8 persons per function. In total, there are ca 180 persons in the emergency response organization. In addition, there is staff from conducting organizations as for instance radiation protection, fire protection and maintenance who are expected to conduct in general the same work assignments as during normal conditions.

In the event of an off-site alert the Plant Manager of the unit and the multidisciplinary technical support are called in and established at the affected unit. In addition, personnel are called in to Ringhals On-site emergency control centre (KC).

For an accident during office hours, the emergency response organization is basically already manned. Otherwise, the emergency procedure will depend on the possibility for getting staff to Ringhals. The group of staff who is already present at Ringhals with the assignment to alert, rescue etc, will have a limited endurance and relieving will be necessary within a few hours in order to maintain the capacity.

6.C.1.1.3 Use of off-site technical support for accident management

In case of a severe accident, Ringhals emergency response organization will contact the emergency group at Vattenfall BU Nuclear Crisis Management Team. This is a specialist group for technical support, primarily in a longer time perspective. The emergency preparedness group at Vattenfall BU Nuclear Crisis Management Team has a yearly joint training with the group's emergency response organizations. The group entails competence when it comes to the unit's structure and function, radiation protection/radiology and reactor safety.

It should be pointed out that the licensee, i.e. Ringhals AB, is responsible for the safety and all actions taken in order to limit the consequences of an event. Westinghouse is represented at the power plant and the emergency response organization is able to take phone contact around the clock with the local representative and the European office in Brussels for technical consultations for the PWR-units and for Ringhals 1 Westinghouse BWR-office in Västerås.

6.C.1.1.4 Procedures, training and exercises

For Ringhals 1 the management of disturbances and accidents are guided by event-based Emergency Operating Procedures. In case the disturbance is complicated, with multiple or gross failures, the Shift Supervisor decide to do the transition to symptom oriented accident management, which for Ringhals 1 are called "Generalen". These procedures aim to maintain or restore the safety functions reactivity, core cooling, reactor pressure control, heat sink and containment pressure control.

The "Generalen" is used by the control room staff lead by the shift supervisor. Technical support may be obtained from the expert group in the on-site emergency control centre (KC).

During shutdown states additional guidance is found in "Korpralen", which for a variety of disturbances will guide the control room to relevant procedures.

For Ringhals 2-4 the management of disturbances and accidents are guided by Westinghouse Emergency Response Guidelines (ERG). Design Basis Accidents are handled by Optimal Recovery Guidelines (ORG), to some extent event based, the so called "E-procedures". Given that no critical safety functions are challenged these procedures will guide the optimal way of handling the disturbance. In case critical safety functions are challenged a transition will be made to the F-procedures - Function restorating - which are more symptom-oriented and focused upon maintaining and restoring the critical safety functions. In case the temperature measured by CET (Core Exit Thermocouple) exceeds a certain value a transition will be made to the SAMG (Severe Accident Management Guidance) where priorities are changed from restoring core cooling to protect the fission product boundaries.

The functions within the emergency response organization have specific instructions with checklists. The checklists have an initial part for collection of facts and early actions followed by a part for reoccurring actions. The administration of these checklists is part of the preparing training for introducing a function into the organization. The checklists are a part of the instructions. They are recurrently being updated and operational experience feedback from the training is an important basis when improving the instructions.

The functions are educated in accordance with the emergency preparedness function's training program and the courses are documented in the plant's computer system. Exercises are conducted recurrently. The aim with the more extensive exercises is to maintain the organization's ability to solve internal assignments and to cooperate with external emergency preparedness functions. These more extensive exercises are response exercises, sometimes with minor features of field trainings, for instance attention to an injured person. The exercises are normally conducted to the alarm level "off-site alert" or "general emergency", i.e. an on-going radiological release or an expected release within 12 hours. Experiences from these exercises are implemented in valid routines and are communicated at courses and updated position instructions. Also improvements regarding equipment and material are presented and prioritized via the exercises.

A major drill is carried out at the national level every two years. This means that such a drill involve Ringhals every eight-year. The most recent drill of this type was carried out at Ringhals in October 2006. A unit drill is carried out once a year. Ringhals emergency response organization carries out the drill and the Halland County Administrative Board, SSM (Swedish Radiation Safety Authority) and Vattenfall Business Unit Nuclear Crises Management Team participate as external parties. Technical problems in the plant with radiological releases, injured personnel and fires are regular components in these drill scenarios. The Engineer On Duty (VHI) rehearses initiation of the alarm phase six times a year. VHI is called to control room by the shift supervisor, VHI decide alarm level and then alert SSM, Halland County Administrative Board. The authorities confirm and after that the drill is concluded. In addition to regular drills, a new focus has recently been introduced and implemented at Ringhals - function drills - where only selected parts of the emergency organization participate. That could be for instance a function drill with the VHI, the Security Company and the Police, where the scenario could be a criminal attack against the plant.

Interruption of the power supply is a part of the education and exercise for the emergency preparedness staff and the Engineer on Duty. However, within the emergency preparedness function no education and training is conducted directed to flooding or earthquakes.

The staff is exercised in connecting the diesel generator for power supply to the on-site emergency control centre. The diesel generator's availability is at present not guaranteed towards Technical Specifications or similar.

It is usually exercised that one unit suffers an emergency event. Emergency events at two units have been exercised at a few occasions. It has never been exercised though that all four units at Ringhals suffer from parallel events.

6.C.1.1.5 Plans for strengthening the site organisation for accident management

During the years 2010 and 2011, Ringhals has conducted both a requirement analysis as well as a prestudy report as preparation for an update of the communication equipment and power feed to the on-site emergency control centre. The improvements within the project are aimed at expanding the mobile telephony and establishing a new radio communication system. The latter also gives an opportunity for direct calls with external emergency response organizations such as for instance the rescue service and ambulance. The corresponding system will also be available in the secondary control room and in the on-site emergency control centre.

The power feed to the on-site emergency control centre could be arranged by manual connection of a diesel generator. The transition to the diesel secured power feed should preferably be conducted automatically though. This is included as a suggestion to the pre-study mentioned above.

6.C.1.2 Possibility to use existing equipment

Unit managers and technical support within the emergency response organization are physically located in the control room and are surrounded by closed ventilation, power measurement etc. in the same extent as the rest of the control room. The equipment is unit computers, PC, stationary telephony and mobile telephony.

In the on-site emergency control centre (KC), there are calculation computers, PC, communication equipment of different types. The room is located in an underground shelter, which provides a good protection towards radiological releases. The power measurement (ordinary net) is conducted via Ringhalst 1 or Ringhals 2 but it can also be conducted via a mobile diesel driven power aggregate.

At Ringhals plant, there is an internal rescue force with a fire-fighting vehicle. The force is constituted by an operation manager with rescue service and radiation protection education and a group of four security officers with fireman education. The emergency response organization requires external support at larger events in order to rescue personnel and extinguish fires in the plant. A prerequisite for them to work, is that they can get to the plant. Under condition that the access roads are free, the rescue service from Varberg municipality arrives within 20 minutes and the ambulances from Varberg Hospital can arrive within 30 minutes.

Within the Ringhals site, there are also two mobile diesel driven pumps. They can be used for supplying water to the sprinkler system and different water sources can be used. The condition is also here that it should be possible to move the pump aggregate with the auxiliary equipment and work in the areas surrounding the units. Ringhals has also two mobile fire pumps at their disposal and they are located on trailers and are driven by petrol. It should be noted though that connection flanges towards containment filtered venting system haven't been prepared.

In the external emergency response organization, it is the rescue operation leader at the County Administrative Board in the county of Halland, who conducts the national rescue service according to the law about protection against accidents. The rescue operation leader's authority in order to mitigate the consequences of an event, is practically unlimited and the whole community's resources may be requisitioned. These resources should primarily be used within the area of responsibility, i.e. outside the power plant but it is reasonable to presume that resources also could be offered to the power plant's disposal in cases when they can be assumed to mitigate the consequences of the event.

If we assume that light equipment cannot be brought on site within the first 24 hours and heavy equipment after 72 hours this would mostly affect the organizations ability to fight fires and rescuing injured personnel. However, there are quite a limited number of people available to handle the mobile equipment on site if additional personnel cannot be brought in.

The work with mobile equipment in connection to the units, is rendered more difficult if the yard and the unit is contaminated. The staff will have to work with safety equipment.

Critical necessities after a severe accident are e.g. diesel to the emergency power diesels, lubrication oil, battery capacity for maintaining the signal systems at the units, raw water as a water source for various cooling requirements, boric acid for control of reactivity, ion exchange compound for cleaning the wastewater, process filter. Filter elements to Ringhals cleaning system has a safety layer and food.

Limitations of transports to Ringhals may result in a lack of certain necessities. Some of the necessities do not have a safety storage due to the material's limited shelf life. This is for instance the fact for ion exchange compound.

Several of the necessities are available in connection with Ringhals main storage and this requires that internal transports may be conducted. This is not a problem under the condition that the system CFV is functioning as required and that the infrastructure is mainly undamaged.

Assessment of releases is performed in the on-site emergency control centre (KC) with aid of the procedure, which has pre-calculated dose projections for a number of accidents. In case of severe accidents with releases bypassing the CFV it includes some support for identifying a relevant accident scenario with corresponding prepared source term files to be used for dose projections with the computer code LENA. In order to identify the accident scenario access to some plant parameters are necessary, normally through a system in the KC.

In case the KC and/or computer tools are not available the accident scenario identification and dose projections for severe accidents will be much more difficult to perform.

The Containment filtered vent, CFV, allows for a considerable reduction of release of airborne radioactivity. The emergency response organization and the shift team have an opportunity, with aid of the CFV-system, to protect the containment by making a pressure relief intentionally. This also presents an opportunity to control the exhaust time in order to limit the spread of airborne activity to the environment and the own staff. Maintaining the containment to be intact is of greatest importance in order to keep the releases from the plant at an acceptable level.

The containment filtered venting (CFV) is a passive system that doesn't require any actions or decisions to be activated – it is therefore not dependant on the staff's efficiency. The possibility for filtered releases by CFV, creates the prerequisites to work in connection to the units. The Containment filtered vent, CFV, requires no access to electrical power for the function. Controlled releases with CFV however do require electrical power to valves and monitoring. Equipment for handling of liquid waste requires access to electrical power. Also the below mentioned instruments, require power supply for functioning.

There are several instruments which can be used for calculating the size of the release and its spread such as meteorology equipment, computer support in the on-site emergency control centre, activity monitoring on the release filter, environment monitoring equipment, etc. A loss of instrumentation obstructs the opportunities for evaluation what releases could be foreseen and the required actions.

Alarm is signaled by the plant's loudspeaker system. The alarm can be given the status "immediate danger" from both the control room and the security head office. Verbal messages can also be sent from the on-site emergency control centre, each control room, the security head office and from the main telephone switchboard. The seven assembly points are equipped with loudspeakers and some of them also have displays for written messages.

Ringhals' initial requirement is to alert the own emergency response organization and the external emergency preparedness functions. The latter is at present conducted by SOS Alarm who passes the alarm on with aid of alarm lists.

When the emergency response organization has been established, internal communication is required in order to create a mutual situation image and convey this to the external emergency preparedness functions with the assignment to limit the damages in the environment as for instance evacuate people living in the vicinity. Internal communication is also required in order for the area supervisor to be able to control the operation through priorities of actions and decide on execution. An example on this is evacuation of own staff from the plant. When the plant has been evacuated most of the communication is expected to be conducted between the on-site emergency control centre, the affected control room and the security centre.

There are a number of systems for internal and external communication such as meridian switchboard with stationary extensions and DECT, stationary telephony with external extensions, Mobile telepony via GSM, satellite telephone (in the Security Centre and at the tele workshop), operation telephone (via 400kV-cable, communication within Vattenfall), telephony via military lines, minicall, mobile radio 80 (Vattenfall radio), the radio system RAKEL (on-site emergency control centre, internal rescue force and the Security office), etc.

The radio system RAKEL is the same system as the Police, the Rescue service and the Ambulance service use. The system is coded and the transmitters are guaranteed power supply with aid of diesel generators. This system together with the military lines, constitute the most reliable communication route for internal and external calls. The RAKEL base stations (outside the plant) are secured with battery and / or diesel operation.

Ringhals does not have a clear strategy for treatment of liquid waste from a severe accident. The waste plants were initially designed for handling 1% fuel damage but there has been a certain degradation of the capacity over time. Today, there are no prepared actions for handling fluid waste from a fuel damage larger than 1 %. There is no equipment for treating hot waste water at Ringhals either. Contaminated water from the CFV filters can be pumped back into the containment but there is no similar opportunity for fluid leakage from the containment or into the auxiliary system. It has not been analysed how the plant withstands airborne or fluid releases from fuel damages in the fuel pits. At fuel damages in the fuel pits, airborne releases cannot be limited by CFV.

At present, there is no strategy for waste handling that requires that equipment is transported to Ringhals. The ability for handling liquid waste from large fuel damages is poor regardless of accessibility. There are no agreements with external suppliers of mobile cleaning equipment.

The handling of fluid waste may be affected by flooding as this handling is conducted below ground level at the pressure water reactors.

6.C.1.3 Evaluation of factors that may impede accident management and respective contingencies

Disturbances in the infrastructure will affect Ringhals' ability to communicate with the world around. The most reliable way to reach the surrounding world has proven to be telephony via the military's line. This line was also available during the big storm that hit the south of

Sweden in January 2005. As alternatives are the satellite telephones that are available in the Security office and in the tele workshop (in close vicinity to the On-site emergency control centre) as well as RAKEL that is available in the Security office, on the rescue force's operation vehicles as well as in the on-site emergency control centre.

Handling of an event, will require communication both internally within the plant as well as externally. Lacks in the internal communication, may for instance result in reduced opportunity to inform the own staff concerning evacuation of the plant, difficulties in controlling actions both within the units as well as on the yard and deficient opportunity for the on-site emergency control centre to assist the affected unit.

Reduced access to equipment for internal communication may partly be compensated by messengers. Shortages in the external communication may for instance mean reduced opportunity to provide updated information to the Swedish Radiation Safety Authority and the county administrative board in the municipality of Halland, something that may lead to delayed relevant actions for the people living in the vicinity. This will also mean reduced opportunity to receive assistance from the Vattenfall BU Nuclear Crisis Management Team.

If the on-site emergency control centre can not be put in operation during the first 24 hours of an event due to damages to infrastructure, the emergency response organization will need to use the alternative on-site emergency control centre in Varberg. This will most likely have consequences on the accident management. The part of the emergency response organization that is expected to support the shift teams directly in each control room, will not arrive until after 24 hours and the shift team will not have any relieving of colleagues.

Disturbances in the communication systems may affect the possibilities for internal and external alarms. This will affect both the work on the inside of the plant as well as the opportunities for receiving help from the outside.

The staff for initial actions is available at the plant which makes it less sensitive for disturbances. The high mental and physical stress level at an event will require that rescue staff and shift teams are relieved within a few hours. Otherwise, it must be anticipated that the work ability and the ability to make correct decisions will be severely reduced.

External events may affect the staff's ability to conduct rescue operations as well as be relieved. Alarm and call out of the emergency response organization can be performed by satellite telephone. This work method will naturally be more time consuming. The computerized call out system for the emergency response organization is not necessary. Alarm and call out of the emergency response organization can be performed by satellite telephone. This work method will naturally be more time consuming.

If an event affects more than one unit, the requirement for additional staff will increase. The plant will quite immediately suffer lack of critical competences and the endurance will thereby be limited.

Minor exchange stations for telephony are present at the plant under ground level, for instance in the building 3P that provides telephony to Ringhals 3 and 4. There is no mapping of the effect of a possible flooding on the communication abilities. The Security Office and the On-site emergency control centre are also located under ground level, which makes them sensitive for flooding.

The Security office has diesel driven power supply. There are equipment for communication via satellite telephone and RAKEL. The communication equipment in the On-site emergency control centre is expected to be fed via UPS during 20 – 30 minutes and after that, feeding via mobile diesel generator is necessary.

6.C.1.4 Conclusion on the adequacy of organisational issues for accident management

In principle the Ringhals emergency response organization is designed to handle a severe accident with core melt at one unit at a time. Due to the mitigating systems with the CFV filter and independent containment sprinkling, a severe accident at one unit still would result in limited doses to the public, as long as the containment barrier is intact.

In order to manage an accident of Fukushima's extent it would require the nuclear power plant to be completely autonomous and independent of the world around it during the time it would take to establish communications, etc. Ringhals emergency response organization is presently not equipped to handle such an event.

The accident progression will be very much dependent on whether the Ringhals emergency response organization can be established or not. If the accident would occur during office hours the organization is basically already in place. The possibilities to manage an event at one unit would then be relatively good.

Without any possibility of getting staff or equipment to Ringhals during the initial 24 hours it is assessed that the organization will meet with difficulties within a limited time. There will be shortages within several functions. The most severe problems would be that the staff will be rapidly worn out as well as the fact that the operative supporting functions within the emergency response organization – such as operation staff and technical support – will not be available on site.

The design scenario for the containment filtered vent system, CFV, is loss of all alternating voltage and steam driven systems. There is equipment available at the plant for handling two units in such a way that the accident would end up in the designed scenario. Ringhals is equipped with two mobile units for independent sprinkling and filling of the containment. Experience from Fukushima suggests that steam driven systems for core cooling have been important for delaying the accident. These systems may possibly provide the emergency response organization with valuable time, which could improve the outcome compared with the design case.

If off-site power is lost, a mobile diesel generator can be connected to the on-site emergency control centre (KC). If this is not possible there is battery power capacity available for 30 minutes operation of the main equipment in the on-site emergency control centre. In the event that KC is unavailable, the alternative emergency control centre in Varberg will be used. The emergency preparedness function has not been designed based on flooding and earthquakes. However, the estimation is that an earthquake within design doesn't constitute any further problem than for the plant at large. At a flooding over ground level (+103), the on-site emergency control centre will be flooded since it is located in the basement. There are no analyses of how the communication equipment etc withstands a flooding or an earthquake.

In the long-term the effect on the environment will be very dependent on the plant's ability to handle large amounts of radioactive contaminated water. Ringhals' liquid waste handling is designed for 1% fuel damages but for larger damages, there is no clear strategy for the liquid waste handling. Ringhals has at present no prepared actions or plans for handling waste from more extensive fuel damages. The emergency response organization's management of loss of cooling in the spent fuel pit should be reviewed as there are insufficiently prepared actions and a lack of analyses of the consequences of core damage in the fuel storage.

6.C.1.5 Measures which can be envisaged to enhance accident management capabilities

The evaluation of potential for additional measures for Ringhals 1 gives that increased battery capacity would allow longer operating times for most accident management actions like injection, pressure relief, etc. Alternative ways to feed DC-sources would allow longer operating times for most accident management actions. Pressure relief with relief valves "Fail-as-is" would allow pressure relief even after loss of DC (planned improvement).

The evaluation of potential for additional measures for Ringhals 2-4 gives that the injection measures powered from independent/diversified sources would enhance the injection capability. Increased battery capacity would allow longer operating times for most accident management actions like injection, pressure relief, etc. Alternative ways (in addition to the system with mobile pump for containment spray) to feed DC-sources would allow longer operating times for most accident management actions.

6.C.2 Accident management measures in place at the various stages of a scenario of loss of the core cooling function

For Ringhals 1, the first priority in this scenario is to prevent fuel damage. If fuel damage has occurred the strategies are shifted towards mitigating the consequences to the public which means protecting containment integrity.

For Ringhals 2- 4, the initial accident management measures in a scenario with loss of the core cooling function are focused on restoring cooling function guided by event based Westinghouse ORG, the so called "E-procedures". In case critical safety functions are challenged a transition will be made to the F-procedures - Function restorating - which are more symptom-oriented. In case the temperature measured by CET (Core Exit Thermocouple) exceeds a certain value a transition will be made to the SAMG (Severe Accident Management Guidance) where priorities are changed from restoring core cooling to protect the fission product boundaries.

6.C.2.1 Before occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes (including last resorts to prevent fuel damage)

For Ringhals 1 the actions before occurrence of fuel damage are as follows: a) prevent criticality with control rods or boron injection, b) containment isolation, c) Reactor Pressure Vessel (RPV) pressure relief, d) RPV injection and e) maintain/restore heat sink (condensation pool).

Last resort to prevent fuel damage will be to inject water into the RPV using the plants existing safety systems. The possibility for fuel damage in high pressure (which might lead to RPV failure at high pressure) is avoided by pressure relief of the RPV.

For Ringhals 2-4 the actions before occurrence of fuel damage are as follows: a) maintaining level in steam-generators, b) pressure relief in the primary system and c) restoration of injection into the primary system.

6.C.2.2 After occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes

For Ringhals 1 the actions after occurrence of fuel damage are as follows: a) RPV pressure relief, b) RPV injection and c) maintain/restore heat sink (condensation pool).

For Ringhals 2-4 the actions after occurrence of fuel damage are as follows: a) maintaining level in steam-generators, b) pressure relief in the primary system, c) restoration of injection into the primary system and d) filling containment with water up to lower part of RPV.

6.C.2.3 After failure of the reactor pressure vessel/a number of pressure tubes

For Ringhals 1 the actions after failure of the reactor pressure vessel are as follows: a) ensure that the ex-vessel core is covered by water, b) prevent over-pressurization of containment and c) fill the containment up to lower part of RPV.

For Ringhals 2-4 the actions after failure of the reactor pressure vessel are as follows: a) ensure that the ex-vessel core is covered by water, b) prevent over-pressurization of containment and c) fill the containment with water up to lower part of RPV.

6.C.3 Maintaining the containment integrity after occurrence of significant fuel damage (up to core meltdown) in the reactor core

6.C.3.1 Elimination of fuel damage / meltdown in high pressure

The possibility for fuel damage in high pressure (which might lead to RPV failure at high pressure) is avoided by pressure relief of the RPV in Ringhals 1. The actions after occurrence of fuel damage are a) RPV pressure relief, b) RPV injection and c) maintain/restore heat sink (condensation pool).

The actions after occurrence of fuel damage in Ringhals 2-4 are a) maintaining level in steam-generators, b) pressure relief in the primary system, c) restoration of injection into the primary system and d) filling containment with water up to lower part of RPV.

6.C.3.2 Management of hydrogen risks inside the containment

For Ringhals 1, the SAM measures which will be performed for protecting the containment function after occurrence of fuel damage are as follows: a) H2-deflagration/detonation is prevented by keeping the containment atmosphere inerted with nitrogen, b) O_2-generation from a large LOCA may be recombined with "Mobile Recombiner" and c) the containment atmosphere may be adjusted through further injection of nitrogen from "Nitrogen storage for ventilation of containment". During outage as well as approximately 12 hours during start-up or shutdown the containment atmosphere is not inerted. There are no specific measures to handle hydrogen risks during these operations. There is no prepared system or strategy fully qualified for recombining O_2-generation from severe accidents. A recently performed evaluation has confirmed that no actions are required within at least 7 days.

The proposed actions are venting with CFV, possibly combined with purging with system (Nitrogen storage for ventilation of containment) and recombination with Mobile Recombiner (although not qualified for severe accidents and earthquakes). In case of high pressure and high hydrogen concentrations in the containment there is a possibility that containment leakage (even the allowed design-leakage) can give flammable and potentially dangerous hydrogen concentrations outside the containment. This risk can be mitigated through venting excessive containment pressures with the CFV.

For Ringhals 2-4 the SAM measures which will be performed for protecting the containment function after occurrence of fuel damage is that H2-deflagration/detonation is prevented through PAR (Passive Autocatalytic hydrogen Recombiners) which even for very large core degradations will within some hours reduce the hydrogen concentration to safe levels through recombination of hydrogen and oxygen. Guidance is available in the SAMG:s to keep containment steam inerted until hydrogen concentration has decreased to safe levels. Hydrogen may also be vented through CFV without dangerous hydrogen deflagrations since it is kept at an elevated temperature (90 °C) which makes it steam-inerted.

6.C.3.3 Prevention of overpressure of the containment

The containment pressure can be decreased by following system:

- Containment filtered venting system (BWR and PWR)
- Containment over-pressurization protection system (only BWR)
- Contianment spray system (BWR and PWR).

Over-pressurization of the containment in Ringhals 1 is normally prevented through the containment spray system. If this is not possible, containment overpressure may be reduced through filtered venting (CFV) which also will reduce pressure from non-condensable gasses, e.g. hydrogen. In addition, pressure can be reduced by activation of independent spray. CFV is inerted and can relieve a mixture of steam, hydrogen and nitrogen without dangerous hydrogen deflagrations.

Over-pressurization of the containments in Ringhals 2-4 is normally prevented through spray. If this is not possible, containment over-pressure may be reduced through filtered vent (CFV). In addition, pressure will be reduced by activation of independent spray.

In case of venting the amount of radioactive material released to the environment may be estimated based upon a combination of precalculated accident simulations, monitoring of radiation and flow, and known decontamination factor (DF) of the filtered vent system. An early activation of CFV may give significant but short-term doses from noble gasses, but will give relatively limited releases of e.g. cesium and iodine.

6.C.3.4 Prevention of re-criticality

After occurrence of fuel damage there is a possibility that the control rods might melt before the fuel rods. In case injection is recovered at this time there is a risk of re-criticality.

For Ringhals 1, the re-criticality is prevented through boron injection. Some guidance is available in the handbook for the unit management (HSB-B) regarding possible reduction of injection flow to RPV in order to reduce RPV water level and reactor power in a situation with re-criticality.

For Ringhals 2-4, the re-criticality is prevented through use of borated water injection sources. In case of severe core degradation recriticality is considered unlikely even if non-borated water injection sources are used.

6.C.3.5 Prevention of basemat melt through

For Ringhals 1, the basemat melt through is prevented through maintaining a sufficiently high water level in wet-well. Since the wet-well extends over the whole area below the RPV and is initially filled with several meters of water it is unlikely that wet-well will be depleted. The combination of an initial big wet-well water volume and the possibility to maintain the wetwell inventory through independent spray is supposed to give a low probability for basemat failure.

For Ringhals 2-4, the basemat melt through is prevented through maintaining a sufficiently high water level in the containment. Guidance is given in the SAMG-procedures which will ensure a sufficiently high water level, possibly through the use of independent spray, in order to give a low probability for basemat failure.

6.C.3.6 Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity

The assessment of adequacy for Ringhals 1 reveals that without DC there are no adequate strategies for injection to the RPV. Pressure relief is dependent upon DC-power which without AC will be depleted fairly early (less than 8 hours).

The assessment of adequacy for Ringhals 2-4 reveals that without AC there are no adequate strategies for injection to the RPV. Pressure relief is dependent upon DC-power which without AC will be depleted fairly early (less than 8 hours). There are some possibilities to feed DC-sources for pressure relief with the mobile pump for containment spray.

6.C.3.7 Measuring and control instrumentation needed for protecting containment integrity

The instrumentation needed for protecting containment integrity consists mainly of pressure, water level and temperature in different compartments.

6.C.3.8 Capability for severe accident management in case of simultaneous core melt/fuel damage accidents at different units on the same site

The operations needed for the management of severe accidents are described in emergency operating procedures. The actions needed are either automated or conducted by the operators in shift. Thus, in case of simultaneous accidents at different units immediate actions needed for severe accident management could be taken at each unit.

When all units simultaneously are subjected to severe disturbance it can be difficult to have sufficient number of competent personnel available. The on-site emergency control centre is not organised to handle events on several units simultaneously.

6.C.3.9 Conclusion on the adequacy of severe accident management systems for protection of containment integrity

In the context of the Fukushima accident, the CFV system appears to be an important last line of defence capable of significantly mitigate accident scenarios with loss of several safety functions and core damage. The system is designed for earthquake, is unlikely to be affected by flooding and will retain most of its capability even without supporting actions. The system will be activated passively by a rupture-disc at excessive containment pressures, and hence no decision-making is required.

In the long term operation of the scrubber, the water level may be reduced by evaporation, but may also be increased by condensation of the process flow. In case of low water level the CFV may be refilled with water processing in separation system together with mobile pump for containment spray without AC. In case of high water level the contents of the CFV may be pumped back to the containment.

In the context of the Fukushima accident the ability to vent the containment through the CFV-system appears to be an important possibility to reduce the hydrogen risk in Ringhals 1. The system is designed for earthquake, high radiation and high hydrogen concentrations, and has only minor dependencies on supporting systems. Purging containment with nitrogen will be necessary in the long term, but might be difficult to do due to high radiation. The ability to handle hydrogen/oxygen without venting is less reliable for severe accident conditions, and may require further actions within a week.

The mobile pump for containment spray consists of 2 fire-trucks equipped with independent diesel driven pump and generator. The pump can take suction at various sources including the sea, and can feed the containment spray system as well as (for PWR) feed-and-boil of the spent-fuel pit. The generator charges the batteries in the CFV-system. At PWR it also can feed a small subset of the DC-system in the ordinary safety power system of the station. The system is designed for earthquake and is fairly independent of supporting systems. Since there are 2 fire-trucks for 4 units the availability might be limited if considering more than one accident simultaneously, and/or considering several needs (e.g. fuel pool feeding and independent containment spray) simultaneously. Destroyed infrastructure might hinder the activation of the system.

For Ringhals 1, the measures for handling hydrogen risk during power operation are considered adequate and robust. Although the recombination equipment is not qualified for severe accidents there are possibilities to purge the containment with nitrogen, and vent hydrogen in a reliable way. There are no measures in place for handling hydrogen risk during outage with non-inerted containment. The risk contribution is assessed and quantified in the PSA Level 2 analysis for shutdown conditions. The measures for handling over-pressurization are considered adequate and robust. Recriticality may be handled by the boron injection system, but has limited capacity. Recriticality may be handled by keeping the RPV-water level at a reduced level. This strategy is dependent on a clear picture of the current situation which may not be available during an accident. The measures for handling re-criticality are considered less adequate and robust, but might be deemed acceptable given the low probability of the event. The event is currently assessed but screened out in the PSA Level 2 analysis. The measures for preventing basemat failure are considered adequate and robust.

For Ringhals 2-4, the measures for handling hydrogen risk are considered adequate and robust. The measures for handling over-pressurization are considered adequate and robust. The measures for handling re-criticality are considered adequate and robust. The measures for preventing basemat failure are considered adequate and robust.

However, some important cliff edge effects have been identified where the dominating "cliff-edge-effect" is depletion time for DC-sources as follows:

- For Ringhals 1 the depletion time generally is 2-4 hours, longer for some DC-sources.
- For Ringhals 2 the depletion time is at least 2 hours, longer for some DC-sources.
- For Ringhals 3-4 the depletion time is at least 1 hour, longer for some DC-sources.
- For the CFV (all units) the depletion time is at least 8 hours. There are prepared equipment (mobile pump for containment spray) and procedures for establishing a DC-feed of these batteries within 8 hours.
- For Ringhals 2-4, there are prepared equipment and procedures for establishing a DC-feed for a small subset of the DC-system in the ordinary safety power system of the station.

6.C.3.10 Measures which can be envisaged to enhance capability to maintain containment integrity after occurrence of severe fuel damage

For Ringhals 1, the measures for purging the containment with nitrogen might be difficult to perform due to radiation. This might be improved through modifying equipment and/or adding radiation shielding. Slow oxygen-production might be recombined with PAR-units (Passive Autocatalytic Recombiners). The hydrogen risk during outage might be reduced through combination of measures like recombiners, igniters, ventilation, reduction of oxygen-content in air, etc. Such measures has to be carefully studied in order to not be contraproductive. E.g. hydrogen recombiners might act as an igniter, ventilation measures in order to reduce hydrogen risk might give increased releases. Capacity and speed of boron injection might be increased which is a planned improvement.

For Ringhals 2-4, no additional measures were identified.

6.C.4 Accident management measures to restrict the radioactive releases

6.C.4.1 Radioactive releases after loss of containment integrity

For Ringhals 1, some SAM measures may mitigate the consequences in case of loss of containment integrity. In case of a minor leakage where the containment is at elevated pressures consequences may be mitigated through using spray (normal or independent). In Ringhals 1 the reactor building is an additional barrier to the containment. Ventilation of the reactor building may be important to mitigate consequences of a failed containment. In case of a gross containment failure and an ex-vessel core melt consequences may still be mitigated through ensuring that the core melt is covered with water. This can be achieved by using normal or independent spray. The desired water level is 32 m above the bottom of containment. This level will cover the bottom of the RPV. Higher levels will have adverse effects upon pressure control, flooded equipment etc. In case of a loss of containment integrity the release of iodine may be mitigated through ensuring a high pH in the wet-well. At present there are no measures in place to achieve this. This is also the case with significant water leakage.

For Ringhals 2-4, some SAM measures may mitigate the consequences in case of loss of containment integrity. In case of a minor leakage where the containment is at elevated pressures consequences may be mitigated through using spray (normal or independent). In case of a gross containment failure and an ex-vessel core melt consequences may still be mitigated through ensuring that the core melt is covered with water. This can be achieved by using normal or independent spray. The desired water level is 7 m above the bottom of the containment. This level will cover the bottom of the RPV. Higher levels will have adverse effects like flooded equipment etc. In case of a loss of containment integrity the release of iodine may be mitigated through ensuring a high pH in the containment. This is achieved through existing passive pH-adjustment with baskets filled with TSP (TriSodiumPhosphate) at the bottom of the containment. For the case with significant water leakage there are however no mitigative measures in place.

Maintaining level in steam-generators will be important to mitigate consequences of an SAI-SGTR (Severe Accident Induced Steam Generator Tube Rupture) which could result in containment by-pass. The risk for SAI-SGTR can be further reduced through improving reliability and availability of measures for maintaining level in steam-generators and reducing RCS pressure.

6.C.4.2 Accident management after uncovering of the top of fuel in the fuel pool

For spent fuel pools (SFP), "adequate shielding" is considered to be more than 2 m water above the top of fuel. Several strategies have been identified as possible "before losing adequate shielding", whereas very few strategies have been identified as possible "after losing adequate shielding". "Uncover of the top of fuel" is considered to be a further development of "losing adequate shielding" and the applicable SAM measures for cooling are the same. "Occurrence of fuel degradation (fast cladding oxidation with hydrogen production)" is considered to be an even further development of "losing adequate shielding" and the applicable SAM measures for cooling are the same. There are no prepared SAM measures in place to cope with the specific effects of cladding oxidation and hydrogen production in the SFP. The time to boiling, loss of shielding and uncovery are very much dependent on whether the plant is in normal operation with limited amounts of fuel in the storage pool or in outage for refueling. During outage, adequate shielding would be lost within 2-3 days, while the times are extended to one week or more for a plant in normal operation.

Following measures are possible in case of loss of cooling function in the SFP:

- before losing adequate shielding against radiation there are some possibilities to restore SF cooling if AC-power is available. If AC-power is unavailable it is possible to establish feed-and-boil with fire water.
- after losing adequate shielding against radiation there might be some possibilities to establish feed-and-boil if AC-power is available. Without AC-power it is not possible to establish feed-and-boil with fire-water when the SF area is not habitable. A previously established line-up may continue to be operated.

Generally, the measures for handling a loss of SFP cooling when the SFP-building is habitable are robust and adequate. Once the SF-building is no longer habitable there are very limited possibilities to re-establish SFP cooling. Monitoring equipment is not qualified for "hot and wet" conditions. There are no mitigative actions in place in case of fuel damage.

6.C.4.3 Conclusion on the adequacy of measures to restrict the radioactive releases

Since the concept of the leak-tight containment is such a fundamental aspect of reactor safety, the efficiency generally is limited for measures for mitigating loss of containment integrity. However, given that the containment integrity is lost the possibility to keep the core covered by independent spray is considered adequate and robust.

The PAR-system (Passive Autocatalytic hydrogen Recombiners) in the Ringhals PWR:s appears to be an robust and reliable way of limiting the risk of hydrogen from severe core damage. The system is designed for earthquake and independent of supporting systems. The system will be activated automatically (passively) at certain hydrogen concentrations, and hence no decision-making is required. The system has been tested in very demanding environments and is supposed to remain functional long-term without any maintenance or other actions.

For spent fuel pools there are in general several possibilities for cooling, but most of them require that the SFP-building is habitable. For PWR, the time from loss of SFP to onset of boiling is in some cases around 2 hours, and to loss of shielding around 16-18 hours, why the time available to line up alternative measures is very short. For BWR, the time from loss of SFP to onset of boiling is in some cases around 5 hours, and to loss of shielding around 40 hours, why the time available to line up alternative measures is fairly short. Once water-levels in the SFP has decreased through boiling, or because of structural failure e.g. after an earthquake, and caused loss of shielding, there are few SAM measures available.

6.C.4.4 Measures which can be envisaged to enhance capability to restrict the radioactive releases

The evaluation of potential for additional measures reveals that measures for ensuring a high pH in the wet-well could be beneficial. This is a planned improvement for Ringhals 1. The other planned plant modifications before the Fukushima accident which will improve the capabilities for severe accident management in Ringhals 1 are:

- boron injection where the capacity will be increased and the required time for completion of the injection will be reduced.
- pressure relief where some relief valves will be FAI (Fail As Is) and hence once opened pressure reduction will continue even if DC is depleted

In order to improve the capability to handle loss of SFP cooling several measures can be suggested:

- a line could be constructed from a shielded position outside the reactor building to the SFP giving the possibility to fill water in the SFP from a fire-truck. Some instrumentation might be needed for level, temperature and radiation monitoring
- an independent cooling system powered by a dedicated air-cooled diesel could be constructed with a primary cooling circuit in the SFP, a heat exchanger and an air-cooled secondary cooling circuit.

6.2 Assessments and conclusions of severe accident management

The assessments done by the licensee and the review made by the authority in the area of severe accident management has resulted in conclusions and recommendations for further analyses which should be considered as potential measures to increase robustness of the plants.

6.2.1. Licensees assessments and conclusions

Severe accident management and emergency response organization have been described for all type of accidents, starting from design basis where the plants can be brought to safe shutdown without any significant nuclear fuel damage and up to severe accidents involving core meltdown or damage of the spent nuclear fuel in the storage pool. Severe accidents involving core melt and melt-through of the reactor pressure vessel are design basis accidents for the consequence mitigating systems where the system for containment filtered venting is the main component. The containment filtered venting systems, including relevant instrumentation, is designed for passive operation during at least 24 hours.

No deviations from the design basis have been identified. There is however one important issue in this connection which needs further evaluation, namely the common system for containment filtered venting for Oskarshamn 1 and Oskarshamn 2 units.

The cliff-edge effects have been identified for various stages of accident progression according to ENSREG specifications. All cliff-edge effects are coupled to the possibility of performing a specific accident management action.

A summary of the conclusions that the licensees have drawn are as following. All conclusions have not been identified by all licensees and are not relevant to all units. Larger deviations between units are mentioned.

a) <u>The endurance of the severe accident management system in all aspects.</u>

The existing accident managing system is not designed or analysed neither to work independent of support from off-site, nor without power supply to safety systems for a longer period of time. The stress test has identified several areas were problems will occur, both regarding equipment, staff including non-durables and operating procedures.

b) <u>Capability to handle more than one affected unit</u>

In principle the emergency response organization is designed to handle a severe accident with core melt at one unit at a time. If two or more reactors are subjected to a severe accident, the dimension of the current staffing and shift-rotation can lead to that all required accident management cannot be executed as intended. At present, all simulator trainings and emergency trainings are performed with the assumption that only one unit is affected. However the containment filtered venting (CFV) is individual for each unit, except for Oskarshamn unit one and two who share filter which could affect its endurance. A severe accident at multiple units would thus still result in limited doses to the public, as long as the containment barrier is intact.

c) Capability to cool the spent fuel pool

In the event of a total loss of power (station blackout), there is currently no system that can be used for cooling the spent fuel pools. Make-up of the pools is possible with use of fire water. If the shielding is lost due to a low water level in the spent fuel pools, it may be problematic to establish this make-up.

d) Introduce/enhance alternative power back-up sources and system to inject water to the reactor vessel to handle SBO.

All existing systems for injecting water to the reactor vessel are dependent on an external power supply or a diesel backed-up net (supported also by gas turbine). In the event of a total loss of power (SBO) there is no possibility of injecting water to the reactor vessel. For PWR, cooling the core via the steam generators is still possible with emergency feed water as long as there is battery power.

e) Enhance management of hydrogen in the containment and reactor building

The current Emergency preparedness has not considered the possibility of hydrogen leaking out and accumulating in the reactor building.

f) Managing re-criticality

Need has been identified for updating of the strategies for handling re-criticality, both for detection and countermeasures. Recent information should be used.

g) Measuring radiation levels

Swedish NPPs do not have as many radiation meters throughout the plant as e.g. U.S. NPPs have which is probably needed.

h) Communication system and call-in system

The present system to call-in of personnel may not be reliable enough in all situations.

i) Managing loss of containment integrity

Severe accident managing systems are much focusing on using the dedicated systems and procedures, all aimed to maintain containment integrity with use of independent containment spray and containment filtered venting system. Loss of containment integrity with larger release of radioactive material is not included.

j) Off-site site located emergency control centre

Evaluation of the need for an alternative on-site emergency control centre outside the site, considering advantages as well as disadvantages.

6.2.2. Licensees recommendations for potential improvements

The licensees have also identified the following recommendations for further evaluations and reassessments. All recommendations have not been identified by all licensees and are not relevant to all units.

a) The endurance of the severe accident management system in all aspects.

The question on how to enhance existing accident management system to achieve a robust system capable to handle Fukushima-like condition must be further investigated.

b) Capability to handle more than one affected unit

A thoroughly developed plan for managing several, simultaneously affected units should be made, including mobile equipment for supply of water and power, staffing and procedures.

c) Capability to cool the spent fuel pool

The following improvements should be considered: Permanent filling pipes from a protected location to the spent fuel pools in units that do not have it yet. Robust/simple level measurement in the fuel pools that can be read from a radiation protected location. Analyses of the conditions with a boiling fuel pool with respect to high temperature, radiation, pathways for water and steam, and procedures.

d) Introduce/enhance alternative power back-up sources and system to inject water to the reactor vessel to handle SBO

The need for independent core cooling system and mobile diesel back-up units should be further investigated. Also need for diesel back-up generators to charging batteries and power emergency control centre should be included.

e) Enhance management of hydrogen in the containment and reactor building

The possibility of accumulating hydrogen in the reactor building should be analysed and possible countermeasures implemented. Decision support for handling hydrogen in a lengthy sequence, both in reactor building and containment, should be improved.

f) Managing re-criticality

Need has been identified for updating of the strategies for handling re-criticality, both for detection and countermeasures. Recent information should be used.

g) Measuring radiation levels

There is a proposal to introduce more dose rate monitors in the reactor building to support accident management.

h) Communication system and call-in system

To ensure the call-in of personnel, the need for new call-in methods should be investigated. Alternatives to be used are e.g. satellite phones, RAKEL (radio).

i) Managing loss of containment integrity

Strategies for handling cases with lost containment integrity should be developed.

j) Off-site located emergency control centre

Evaluation of the proposed advantages and disadvantages of replacing the existing substitute command centre with a suitable facility outside the plant area so that both command centres won't be located within the site where they could both be affected by the same bad conditions.

6.2.3. SSM's assessment and conclusions

SSM:s overall assessment is that questions specified in the ENSREG document have essentially been answered in an acceptable way and that the assessment and presentation of NPPs ability to cope with the loads beyond design have been described satisfactory at the present stage of evaluation.

The severe accident management fulfils design requirements as specified in the SSM's regulations. For the containment filtered venting system, the design requirement is passive operation during at least 24 hours. However, the containment filtered venting system is not designed for accident scenarios with the duration and aggravated conditions at the site corresponding to the situation during the Fukushima accident. Thus, an evaluation of the system for the long-term operation, more than 24 hours, should be performed.

The cliff-edge effects have been addressed and presented in essentially satisfactory way. However, the presentation could have been more complete with regard to the time before a specific cliff-edge effect is reached and the coupling between identified cliff-edge effects and recommended potential improvements. Concerning cliff-edge effects for the emergency preparedness organization, the regulatory body has found that these are specified explicitly in the text only for OKG.

SSM:s assessment is that the above conclusions are relevant and reasonable. This is also the case concerning recommendations for further evaluations and/or specific measures. However, in addition to measures identified by the licensees during the assessment of severe accident management (se previous heading), the regulatory body has identified following items which must also be considered by the licensees as potential measures to increase the robustness of the plants:

- Guidelines for emergency response organization for handling an accident in a long-term.
- Handling of containment chemistry in a long-term (a year or more).
- The function of the containment filtered venting system in long-term (more than 24 hours).
- The performance of the common system for filtered containment venting at Oskarshamn 1 and Oskarshamn 2 units.
- The analyses of possible destruction of infrastructure as well as destruction on-site and of safety systems and barriers, must be carried out with regard to that not all accident scenarios have been clearly identified in stress tests as they are defined by ENSREG.

SSM's overall assessment of the emergency response organization is that all licensees give a good description of strategies, instructions and equipment. Weak areas and suggestions for improvements as a result of the stress test are shown. Based on the assessment of emergency response organization, the SSM is of the opinion that the following areas need further and deeper evaluation:

- Emergency planning should comprise severe emergency situations involving all units at the site.
- Accessibility and functionality of the ordinary on-site emergency control centre and the alternative emergency control centre should be secured with regard to localization, protection, robust communications systems and power supply.
- The personal safety issues have to be re-assessed. High demands should be applied due to rapidly changing of high radiation and contaminations levels during execution of accident management measures. Routines for the emergency response organization should be further developed when it comes to protection of the personnel in severe accident environment. Access to protective equipment, dosimetry and management, as well as working procedures need to be clarified
- The need for common, at the site available resources should be evaluated since the currently available resources are not sufficient if all units at the site are affected (even in the short term).
- Action plans should be set up where the need for external resources, both human and material, should be identified along with the information from where and how they can be obtained as well as the time for their transport to the site.
- Areas critical for accident management in long-term should be identified. These can include for example the need for external resources, routines for access to the site and means to manage the larger quantities of radioactive water.

7 General conclusion

7.1. Key provisions enhancing robustness (already implemented)

Measures to raise the level of safety have gradually been taken at Swedish facilities in pace with new knowledge indicating possible or necessary improvements. New knowledge has emerged from lessons learnd from incidents and accidents, from research, safety analyses of the facilities that had identified weaknesses and from new reactor designs.

To some extent, the initiative for taking these measures was made by the licensees and the Swedish Radiation Safety Authority. The consistent point of departure for the measures was nonetheless legislation and regulations imposing requirements on the respective licensee to take any measures that are necessary from the standpoint of safety.

In addition, the Authority's regulations have successively been expanded and been written in more detail. On 1 January 2005, the former supervisory authority, the Swedish Nuclear Power Inspectorate (SKI), put into force new regulations concerning the design and construction of nuclear power reactors, now designated as SSMFS 2008:17. When they entered into force, the regulations contained transitional provisions providing the basis for SKI's decision concerning reactor-specific modernisation programmes, including a timetable for implementation of these programmes, see section 1.9.

The regulations impose requirements on increased resilience against internal and external events, for example through more separation and diversification of equipment and systems in facilities. The regulations also impose requirements on the facilities' resilience against natural phenomena and other events, such as earthquakes, flooding, extreme winds, extreme temperatures and extreme icing.

The stress tests have shown that the mitigation systems, including scrubber for filtered venting of the containment, implemented after the TMI accident, see section 1.8, is of great importance for the robustness of the nuclear power plants. If a situation resembling that in Fukushima should arise, with pressure building up in the containment, these systems would help to gain control over the situation through pressure release from the containment to the atmosphere via the accident filter. The filter has a purification function that captures a large proportion of the radioactive substances that may be present in the containment atmosphere before they are released into the environment. This function is intended to be used in the event of a rapid pressure buildup in the containment, but it is also possible to use this function over an extended period of time if core cooling should fail.

7.2. Safety issues, potential safety improvements and further work forecasted

SSM has reviewed the stress tests carried out by Swedish licensees and drawn the conclusion that they were largely performed in accordance with the specification resolved within the European Union. The scope and depth of these analyses and assessments are essentially in accordance with ENSREG's definition of "a comprehensive assessment of risk and safety". The stress tests also show that Swedish facilities are robust, but the tests also identify a number of possibilities to further strengthen the facilities' robustness. In a number of cases, the stress tests indicate deficiencies in relation to, or alternatively, deviations from applicable requirements imposed on safety analysis. In these cases, SSM will require the licensees to take action so that the facilities fulfil the applicable requirements. The Authority nevertheless assesses that none of the deficiencies currently identified nor the measures needed are of such a nature that the continued operation of the facilities needs to be put into question.

Some areas of improvement have been identified by the licensees as a result of the stress tests, while others have been identified by SSM when reviewing reports from licensees. Many of the areas of improvement identified imply that analyses conducted earlier need to be supplemented, or that new analyses need to be conducted. This is a prerequisite before one can adopt a standpoint as to whether a measure needs to be taken, and in this case, its approach. One cannot rule out the possibility that the analyses will result in the present design or procedure deemed adequate, even for these more extreme events. Apart from the need to conduct additional analyses, the need for more tangible action has also been identified, for example installation of equipment and improved emergency response management by allocating more resources and/or approving new routines. However, these measures also require additional analyses in order to provide a basis for the design of the measures.

The following is a description of the limitations in the facilities' design and potential safety improvements identified in the stress tests.

Earthquakes

In Sweden, only the two newest reactors, Oskarshamn 3 and Forsmark 3, were originally designed to withstand earthquakes. The other Swedish reactors became subject to general requirements imposed on resilience against earthquakes when the Swedish Nuclear Power Inspectorate's regulations concerning the design and construction of nuclear power reactors, SKIFS 2004:2, entered into force in 2005. In order to allow licensees sufficient time to take the measures and fulfil the requirements, separate decisions were taken giving the licensees a certain period of time to plan and take the requisite measures to fully comply with the mentioned regulations, now designated as SSMFS 2008:17. The deadline for taking measures under these 'transitional decisions' is the year 2013. However, it should be noted in this context that the licensees also previously took resilience against earthquakes into consideration, primarily in terms of mechanical equipment in connection with modernisation work and plant modifications.

In their design and other analyses, the licensees apply a dimensioning earthquake within a radius of twenty kilometres of a strength corresponding to a magnitude of approximately 6.0 on the Richter scale and with a probability of once per 100,000 years (10^{-5}). According to SSM this is an acceptable DBE.

As far as concerns consequence-mitigating systems, a dimensioning earthquake has been applied of a magnitude approximately four times more powerful and having a probability of once per 10 million years (10^{-7}). SSM also assesses that the application of a severe earthquake with a probability of 10^{-7}/year is feasible for the evaluation performed of systems and buildings necessary for preventing the release of radioactive material to the environment.

In the documentation submitted, SSM currently assesses that data is somewhat lacking for demonstrating that functions needed to bring the reactors Oskarshamn 2, Forsmark 1, Forsmark 2, Ringhals 2, Ringhals 3 and Ringhals 4 to a safe state will perform as intended during and after an earthquake as stipulated by the dimensioning requirements. SSM generally assesses that the licensees have not taken the measures required under the Authority's regulations for some of the reactors. For this reason, SSM will order the licensees to produce detailed action plans on how and when these detailed analyses and investigations shall be performed. The same situation applies to the additional analyses needed for achieving a more accurate estimation of the margin for safe shutdown and implementation of the improvements identified in the updated safety evaluations. As far as concerns Forsmark and Ringhals, a more detailed analysis also needs to be conducted in terms of earthquake-induced flooding. For more details, see section 2.2.

Flooding

Swedish nuclear power plants are dimensioned for sea water levels of between two and three metres above average water level. All of the plants can nonetheless withstand a sea water level of 3,0 metres above the average water level without resulting in core damage. The licensees assess that this level has a probability of 10^{-5}/year (a probability of once per 100,000 years).

High ground water level is of particular concern for Ringhals 2. The internal water level in the plant that is assumed to cause fuel damage is 3,0 meters below the average sea water level and the time from pump failures until the situation gets critical is very uncertain. Further evaluations have to be performed due to the uncertainties and to the fact that the drainage pumps is connected to the off-site power.

Combination effects of waves and high water levels are not included in the stress tests for all facilities. Moreover, investigations are needed for these kinds of combination effects, including dynamic effects.

The estimated frequency for the three-metre level is 10^{-5}/year based on statistics from SMHI (the Swedish Meteorological and Hydrological Institute). SSM is nevertheless conducting investigations for an assessment of extreme natural phenomena, to which flooding belongs. When these investigations have been completed, SSM will adopt a standpoint in terms of both the three-metre level as well as the expected frequency. For more details, see section 3.2.

Extreme weather conditions

Extreme weather conditions are initially broken down into rapid and slow processes. 'Slow processes' refer to weather situations in which the facility can be brought to a safe state and time allows for compensatory measures to be taken before the extreme weather conditions have fully developed. This category for example includes high and low water and air temperatures.

The facilities' characteristics have been evaluated in relation to extreme weather conditions. The evaluation indicates that the facilities are robust against these kinds of conditions. However, certain areas need more investigation with the aim of further strengthening the facilities. Examples of these include analyses of procedures for the working staff in terms of requisite measures in the event of large quantities of precipitation in addition to requisite measures in the event of extreme temperatures. Other examples of areas needing further investigation include extreme weather conditions combined with consequential events. Also, the bearing capacity of certain roofs needs to be analysed in terms of snow load.

Ice storms are not covered by the nuclear power plants' safety analysis reports (SAR) and have not been analysed in detail in the stress tests. However, an ice storm would be expected to knock out the offsite power and also risk blocking ventilation systems. This is a deficiency in relation to current regulations and SSM intends to order the licensees to conduct analyses of facility robustness against ice storms and take any action that is needed.

SSM is currently conducting investigations into extreme natural phenomena in terms of precipitation, temperatures, wind speeds, etc. that the facilities are to be capable of withstanding under the regulations. While awaiting completion of these investigations, the values used by the licensees are deemed acceptable. For more details, see section 4.2.

Loss of electrical power and heat sink (failed cooling)

A total loss of electrical power or loss of ultimate heat sink (failed cooling) leads to an accident scenario with serious core damage. In order to mitigate the consequences of these kinds of accident scenarios, all Swedish nuclear power plants are fitted with consequence-mitigating systems in which accident filters and the independent function for the containment spray system are key functions. The stress tests demonstrate the strength of these systems in connection with station blackout and a loss of heat sink, or a combination of both these events. The function of consequence-mitigating systems should nevertheless be investigated further from the perspective of an extended accident sequence. As far as concerns Swedish boiling water reactors, the licensees have stated that the accident filters can be used to remove residual heat from the reactor core. However, this function on the part of the accident filters was not covered by the design assumptions and consequently needs further investigation.

Loss of electrical power

The ordinary auxiliary power systems are, as far as concerns all Swedish nuclear power plants, dimensioned to manage a seven-day loss of offsite power. However, it has become evident that some facilities would need refilling of lubricant within a few days. Access to and storage of lubricant at the facilities needs to be investigated further and the possible need for increased storage capacity should be evaluated.

Alternative auxiliary power systems in the form of gas turbines are also available within or close to the facilities. However, these auxiliary power systems have not been safety classified. The licensees' investigations indicate that these alternative auxiliary power systems could be crucial during a sequence of emergency events; also, the need for auxiliary power systems should be investigated further, particularly when considering situations where several reactors are affected simultaneously.

In the event of a loss of offsite power, failed house load operation in addition to loss of ordinary and alternative auxiliary power, what remains operational then is a battery-backed uninterruptible power supply for instrumentation and manoeuvring of components. These batteries are only dimensioned for one to two hours of operation according to the relevant SARs, although they are deemed capable of functioning for a longer period of time. An analysis of battery capacity needs to be conducted in order to further improve the level of robustness.

In the event of a loss of offsite power, failed house load operation in addition to loss of ordinary and alternative auxiliary power, various mobile units can be used, such as diesel-powered pumps and generators. The analyses nevertheless indicate that the capacity and number of mobile units are insufficient for all kinds of events, particularly if several reactors are affected simultaneously. For more details, see section 5.2.

Loss of ultimate heat sink (failed removal of heat to the sea or atmosphere)

All Swedish nuclear power plants are dimensioned to be brought to a safe state if the salt water intake is blocked and to keep the facility in this state. However, as far as concerns Ringhals 3 and 4, it has not been fully verified whether this requirement is fulfilled. An update of design basis events and verifying analyses needs to be conducted.

Simultaneous blockage of both intake and outlet would involve significantly more difficult situations than the above-mentioned design events and would require crucial action by personnel at the facilities. An analysis of requisite manual measures and available resources needs to be performed. This also needs to consider the personnel's access to the facility on the basis of assumed accident sequences and their impact on the work environment.

Analyses of beyond design basis accidents also demonstrate the major significance of independent core cooling, where both permanent and alternative systems as well as mobile units strengthen the facilities' safety and robustness. Evaluations of independent core cooling should be conducted and any need for further enhancements should be investigated.

The analyses also illustrate the importance of available water volumes for the purpose of extending the period of time before serious core damage is unavoidable in connection with severe accidents. A survey of water volumes in various storage tanks and set minimum levels in them needs to be performed. Also, a survey of available water volumes at and in connection with the various sites should be performed and the possible need for reinforcement should be evaluated.

Manual intervention is required to maintain cooling of fuel ponds during a situation where both the water intake and outlet are blocked. Further investigations are also required of the need for additional cooling, both by means of permanent installations and mobile units. A key prerequisite in connection with these investigations is that the environment surrounding the ponds allows the personnel access for manual action. For more details, see section 5.2.

Severe accident management and emergency preparedness

The stress tests show the strength of the consequence-mitigating systems, where the accident filters (filtered venting) constitute key systems. In an accident situation where the reactor core has melted through the reactor pressure vessel and residual heat removal has failed, pressure in the containment rises. In this type of situation, the accident filter enables the release of pressure from the containment to the environment. This filter captures the vast majority of the radioactive substances in the gas and ground contamination can thus be largely avoided. The stress tests have demonstrated a need for updated analyses of filter function in terms of accident situations with pressure relief needed in prolonged accident scenarios because these filters are dimensioned for events that are not as prolonged as was the case in Fukushima.

The batteries for instrumentation and manoeuvring are dimensioned so that they are capable of managing the initial accident sequences and subsequent recharging in an easily accessible way. A survey of battery capacity and charging possibilities needs to be performed in order to strengthen the functions of accident systems.

The stress tests have also demonstrated limitations in the emergency preparedness organisations. Investigations need to be conducted to ascertain what is needed so that a facility's emergency preparedness organisation is dimensioned to deal with situations in which several facilities are affected simultaneously.

All existing systems for supplying water to the reactor pressure vessel are dependent on offsite power, or ordinary or alternative auxiliary power systems. In the event of a total loss of power, there is no way to supply water to the reactor pressure vessel. As far as concerns pressurised water reactors, the reactor core can be cooled via the steam generators, using the auxiliary feed water system for as long as the batteries allow or the water from available water sources lasts. An independent core spray system and alternative mobile auxiliary power systems can substantially raise the level of robustness and should be investigated further.

In Sweden, work has long been underway to develop the facilities so that they are capable of dealing with the risk of hydrogen gas explosions. The stress tests nevertheless indicate that the risk of hydrogen gas leakage to reactor buildings has not been dealt with sufficiently by today's accident response organisations. The risk of hydrogen gas accumulation in reactor buildings needs to be investigated further, as well as the need for additional instrumentation to assist operators. Improvement is also needed in terms of handling hydrogen gas in a long-term perspective, both in reactor buildings and in the containment.

Emergency response management focuses on sequences where the consequence-mitigating systems, with the independent containment spray system and the accident filters, protect the containment's integrity. Lost containment integrity with a relatively large discharge of radioactive substances is not included. Strategies in the emergency response management need to be investigated further and improved. For more details, see section 6.2.

Continued work

The weaknesses and possibilities for improvements to the facilities' robustness as identified during stress testing will be managed in different ways depending on their importance from the perspective of safety and the urgency of implementing the measures.

SSM will order the respective licensees to present an action plan for dealing with the deficiencies identified, taking the necessary measures in addition to an analysis of safety issues and feasible safety improvements. SSM expects the licensees to take the measures' importance for safety into consideration when producing time schedules for the measures. SSM will review the action plans in terms of safety improvements, scope, level of detail and time schedules, and will request additional details or revision if this should prove necessary.

Based on these reviews, SSM then intends to order the licensees to make the necessary safety improvements. This for instance applies to earthquakes and ice storms.

In parallel with the work described above, SSM will be conducting investigations and preparing reports in accordance with the Government assignment from 8 May 2010 (M2010/2046/Mk), updated on 12 May 2011. This assignment for example states that SSM must by 31 October 2012:

- submit a comprehensive report on the stress testing,
- account for the measures taken by the industry by this point in time owing to the tests and the Authority's assessment of these measures, and
- present an evaluation of the issues identified in the stress tests requiring more in-depth analysis, other lessons learnd from the accident in Fukushima in addition to conclusions drawn and any further measures that should be taken at Swedish nuclear facilities.

As far as concerns nuclear power plants, the Government assignment also implies that SSM must by 31 October 2012 present:

- An overall evaluation of the nuclear power reactors' fulfilment of safety upgrade requirements imposed by the Authority in regulation SSMFS 2008:17 and the Authority's assessment as to how this modernisation work has had an impact on reactor safety
- An analysis of the preconditions for operating the reactors over extended periods of time (more than 50 years) and any further requirements imposed on safety improvements ensuing from these kinds of extended periods of operation and developments in technology and science
- International lessons learnd from safety improvements to reactors as a basis for decisions concerning extended periods of operation

SSM has pre-existing plans to revise and supplement the regulations concerning the design and construction of nuclear power reactors. In conjunction with this work, the results from the stress tests and other underlying documentation to be drawn up in accordance with the Government assignment will be an essential platform for the development work.

8 References

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