Forsmarks Kraftgrupp

	Dokumenttyp / Document type	Klassificering / Classification	
	Rapport	Hemlig	

Författare / Author	Granskad av / Checked by	Dok nr / Doc no	Rev
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Fastställd av / Approved by	Datum / Date	Projektnr / Project no	Systemnr / System no
cFT	2011-10-31		

Forsmark 1, 2 and 3

Summary report

Renewed safety assessment of the resistance against certain events - The Stress Test

Abstract

On May 25th, 2011, the Swedish Radiation Safety Authority (SSM) required all Swedish Nuclear Power Plants (NPP) to perform reassessments [1] in accordance with the joint specifications for the stress tests as agreed between European Nuclear Safety Regulatory authorities and the European Commission within the framework of ENSREG [2].

This document is Forsmarks Kraftgrupp ABs final report.

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1 Introduction

In the light of the Fukushima accident in Japan, the European Council on March 24 and 25 declared that the safety of all EU nuclear power plants should be reviewed.

On May 25th, 2011, SSM required all Swedish Nuclear Power Plants (NPP) to perform reassessments [1] in accordance with the joint specifications for the stress tests as agreed between European Nuclear Safety Regulatory authorities and the European Commission, within the framework of ENSREG [2].

This summary report and the referenced detailed reports present the results and evaluations of the stress test performed by Forsmarks Kraftgrupp AB (FKA).

Analytical work for the addressed events is carried out by the methods prescribed in the ENSREG document, see further description of the basic assumptions and methods in 3.1.

Based on the ENSREG requirements SSM has issued an injunction where Forsmark is requested to respond to these requirements.

To comply with the injunction Forsmark has previously submitted three documents:

- “Presentation of prerequisites and starting-points of the work” [3],
- “Complementary information of prerequisites and starting-points of the work” [4]
- “Progress report 15th of august” [5].

The technical scope of the stress test has been defined considering the issues that have been highlighted by the events that occurred at Fukushima, including combination of initiating events and failures. The focus will be placed on the following issues:

a) Initiating events

- Earthquake
- Flooding

b) Consequence of loss of safety functions from any initiating event conceivable at the plant site

- Loss of electrical power, including station black out (SBO)
- Loss of ultimate heat sink (UHS)
- Combination of both

c) Severe accident management issues

- Means to protect against, and manage, loss of core cooling function
- Means to protect against, and manage, loss of cooling function in the fuel storage pool
- Means to protect against, and manage, loss of containment integrity

The purpose is to search for cliff-edge effects, i.e. effects that may cause the sequence of events to change in a significant way to the worse.

2 General data about site/plant

2.1 Brief description of the site characteristics

On the 15 of January 1973, Statens Vattenfallsverk (SV) (now Vattenfall AB) and Mellansvensk Kraftgrupp Aktiebolag (MKG) established Forsmarks Kraftgrupp Aktiebolag (FKA) 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.

Within the Forsmark plant site, there are three nuclear power plants. All plants are BWR reactors and were all 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 units. 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.

2.2 Main characteristics of the units

2.2.1 Forsmark 1 and 2

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. The maximum thermal output in each unit is 2928 MW. Since the reactors have internal circulation pumps and fine motion control rods they are considered to be advanced boiling water designs of generation III.

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

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. The steam produced by the reactor is led via the main steam lines to the turbine. After passing the turbines, the steam goes to the main condensers where, with the help of sea water, it is condensed to water. The condensate is cleaned, preheated and pumped back to the reactor vessel trough the feed water lines. 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.

2.2.2 Forsmark 3

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 the table below.

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.

In difference to Forsmark 1 and 2, Forsmark 3 has 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.

2.2.3 Spent fuel storage

All radioactive waste including spent fuel is handled in Sweden by Swedish Nuclear Fuel and Waste Management AB (SKB), jointly owned by the Swedish nuclear power utilities.

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

Spent fuel is managed in accordance with internal and external procedures and instructions. Internal procedures cover handling and loading of transport casks in the power plant. External instructions, i.e. the Transport Handbook (issued by SKB), regulate transportation from the site to the Central Storage of Spent Fuel (CLAB). The fuel from Swedish nuclear power plants is stored in CLAB until it can be taken to the planned final repository.

Sweden has decided not to reprocess spent fuel. According to the Nuclear Act, Swedish nuclear power plants are not allowed to export nuclear waste.

2.3 Basic design and safety principles

2.3.1 Safety principles

In order to fulfil the single failure criterion without jeopardising safety or operation, all three units 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 units 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 units 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 unit 3, which is of a later design. The emergency diesel generators are placed in separated buildings on each side of the reactor building.

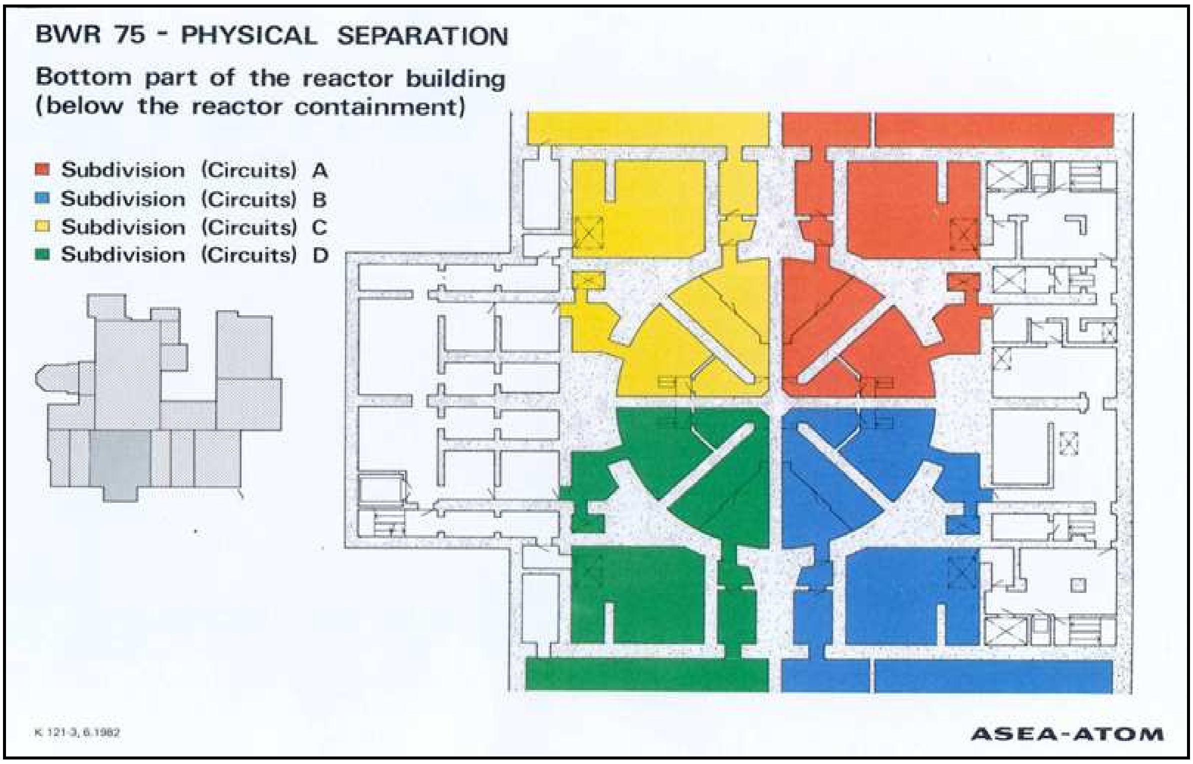

Figure 1. General plan lay-out showing the physical separation of Forsmark 3

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.

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.

2.3.2 Defence-in-Depth (DiD)

The plant is designed and operated according to the basic principles of defence-in-depth. The defence-in-dept approach is creating multiple independent and redundant layers of defence to compensate for potential failures and external hazards so that no single layer is exclusively relied on to protect the public and environment. The levels of defence in dept are described in the table below.

Level	Objective	Essential means
1	Prevention of abnormal operation and failures.	Robust design and high quality in construction and operation.
2	Control of abnormal operation and detection of failures.	Control, limiting and protection systems and other surveillance features.
3	Control of accidents within the design basis.	Engineered safety features and accident procedures
4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents.	Prepared technical measures and accident management.
5	Mitigation of radiological consequences of significant releases of radioactive materials.	Accident management and off-site emergency response mitigation.

2.3.3 Radiological barriers

The units are designed with five barriers to prevent discharge of radioactivity in the event of an accident.

- The uranium dioxide fuel pellets
- The fuel rods
- The reactor pressure vessel
- The reactor containment
- The reactor building

The first barrier is the sintered uranium dioxide fuel pellet. The ceramic composition is difficult to dissolve in water and binds the fission products well. The second barrier is the zirconium fuel rods, which hold the pellets in place and prevent the release of volatile fission products. The third barrier is the reactor pressure vessel and the reactor coolant pressure boundary (RCPB). The fourth barrier is the reactor containment. The fifth barrier is the reactor building with its emergency ventilation system.

The outer shell of the unit is the reactor building, with the majority of the safety related equipment. In the centre of the reactor building is the reactor containment. The reactor containment is built as a pressure vessel and is designed to withstand the pressure that can arise in the event of a large pipe rupture, e.g. a steam line rupture. The containment is gas-tight to prevent any leakage to the surroundings.

The reactor containment has two air locks. During operation, the reactor containment is inerted with nitrogen gas, which eliminates the risk of fire during operation. The refuelling floor houses pools for handling and temporarily storing fuel and reactor vessel components. The pools are connected to a water cooling and cleaning system.

Sweden was the first country in the world to require filtered venting to relieve pressure in the reactor containment in the event of a core melt situation. At Forsmark, this function was implemented in all three units at the end of the 1980s. The system is able to filter 99.96% of the Iodine and Cesium in case of overpressure in the containment. See description of filtered venting system below.

2.3.4 General system description

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

Main Recirculation System (MRS)

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.

Pressure Relief Function (PRF)

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 F3 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.

Emergency Core Cooling Function (ECCF)

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.

Residual Heat Removal Function

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 refuelling the system is also used for cooling the fuel pools together with the fuel in the reactor vessel.

Hydraulic Shut-down System

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).

Boron system

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.

Intermediate Cooling System

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.

Filtered Containment Venting Function

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 system is passive and activated by rupture discs. The filter is a MVSS filter (Multi Venturi Scrubber System), in which the containment pressure discharge steam through venture pipes, which are submerged in a water pool. The gas phase is kept inerted to prevent accumulation of hydrogen.

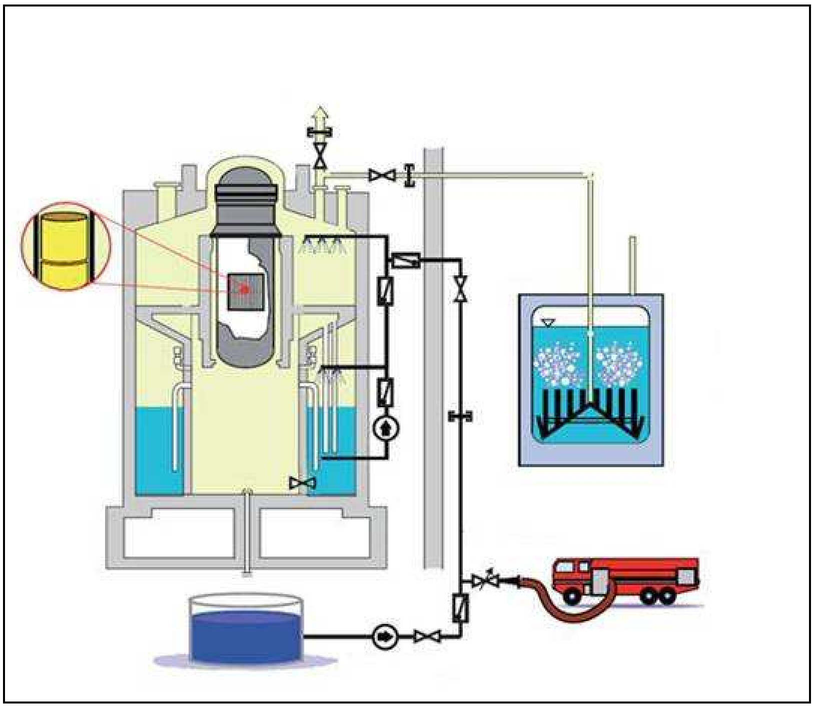

Figure 2. General view of the Filtered Containment Venting Function (FCVF).

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.

The FCVF and strategies were developed in the late 1980's to comply with a government decision, which was compiled as a result of the TMI-2 accident in 1979. The requirement is to mitigate the design scenario so that the release of ^{134}Cs and ^{137}Cs is less than 0.1% of the core inventory of an 1800 MW_{th} core.

2.4 Performed safety enhancements

The original design basis of Forsmark 1 and 2 did not include explicit consideration of Severe Accidents (DECs). Subsequently a review of the capabilities to cope with Severe Accidents was performed and several modifications were made to improve the plants capabilities to cope with DECs. These include a containment overpressure protection that provides structural protection for the containment in the event of a rapid pressure increase due to a LOCA and reduced pressure suppression function.

Forsmark has on its own initiative installed diversified pressure relief provisions. The purpose was to eliminate the risk of high pressure core melt-through, to enable diverse pressure relief functionality and to establish a state of feed and bleed.

In 2004 the Swedish safety authority issued new requirements (SKIFS 2004:2 later, SSMFS 2008:17) for reactor design. Forsmark has implemented a transition plan approved by SSM to comply with the new regulation and a time plan for it's execution.

Some of the safety enhancements in the scope of the transition plan:

- A detect and suppress system to mitigate power oscillations
- A diversified system for Residual Heat removal
- Increased separation between safety and non-safety equipment in electrical supply systems
- Automatization of the Boron Injection System.
- Modification of buildings to improve fire protection and external hazards protection (high winds).
- Increased separation between safety trains in unit 1 and 2.
- Improved passive fire protection by separation.
- Increased protection of components against pipe break.
- Improvements to ensure long term coolability in case of a core melt down.
- New alternate control room at Forsmark 1 and 2.

2.5 Scope and main results of probabilistic safety assessments

According to SSMFS 2008:1 probabilistic safety assessment (PSA) shall be a part of the Safety Analysis Reports (SAR) for the Swedish nuclear power plants. PSA should cover all operation modes (power operation, shutting down, start up and cold shutdown) and all initiating events that may have an effect on the nuclear safety.

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 unit's ability to limit or stop the release of radioactive material. PSA level 2 is yet only performed for power operation.

2.5.1 Scope

There are two PSA studies at Forsmark Nuclear Power Plant, one for the twin units 1 and 2, and one for unit 3. The PSA studies are updated with an interval of approximately two years. Figure 3 show the current status of the two PSA studies at Forsmark Nuclear Power Plant. Areas not yet in the PSA studies are planned to be implemented during the coming years.

	PSA Level 1			PSA Level 2		
	Power Operation	Shutting Down and Start-up	Cold Shutdown	Power Operation	Shutting Down and Start-up	Cold Shutdown
LOCAs	X	X	X	X	-	-
Transients	X	X	X	X	-	-
CCIs	X	X	X	X	-	-
Fire/Internal Flooding	X	-	-	-	-	-
External Events	(X)	(X)	(X)	(X)	-	-

X	=	Implemented in the PSA studies
(X)	=	Implemented simplified in the PSA studies
-	=	Not yet implemented in the PSA studies

Figure 3, current status of the Forsmark PSA studies

2.5.2 Results

It should be noted that PSA results are not possible to use for comparison between different facilities. This is due to differences in the set up of different PSA models. PSA results should primarily be used to identify strengths and weaknesses within the facility analysed, as a basis for safety enhancements.

2.5.2.1 Forsmark units 1 and 2

The total core damage frequency for power operation is $3.2 \cdot 10^{-5}$ /year at Forsmark units 1 and 2 respectively. External events make up 55 percent of the core damage frequency, fire and internal flooding 24 percent and internal events 21 percent. The five most significant initiating events are:

1. Clogging of sea cooling water systems caused by biological material
2. Loss of offsite power
3. Break in demineralized water distribution system in a certain area at the fourth floor of the reactor building together with failing manual actions (alternative 1)
4. Break in demineralized water distribution system in a certain area at the fourth floor of the reactor building together with failing manual actions (alternative 2)
5. Other scrams

The total core damage frequency for shutting down is $1.8 \cdot 10^{-7}$ /year at Forsmark units 1 and 2 respectively. The dominating initiating events are events that lead to loss of RHRF, loss of feedwater and loss of condenser. The total core damage frequency for start-up is $3.6 \cdot 10^{-6}$ /year at Forsmark units 1 and 2 respectively. The dominating initiating events are events that lead to loss of feedwater, spurious activation of one pressure relief valve and spurious activated scram.

The total core damage frequency for cold shutdown is $7.5 \cdot 10^{-6}$ /year at Forsmark units 1 and 2 respectively. The dominating initiating events are loss of RHRF early or late in the cold shutdown and manually initiated leakage from the main recirculation system below top of core.

In total this gives an overall core damage frequency of $4.3 \cdot 10^{-5}$ /year at Forsmark units 1 and 2 respectively.

Out of the $3.2 \cdot 10^{-5}$ /year core damage frequency for power operation 97 % of the sequences lead to acceptable releases (< 0.06 % of the volatile fission products). Non-acceptable releases have a frequency of $7.2 \cdot 10^{-7}$ /year. Large releases (> 10 % of the volatile fission products), which are a part of the non-acceptable releases, have a frequency of $5.0 \cdot 10^{-9}$ /year. Large early releases (which are a part of the non-acceptable releases and large releases) have a frequency of $5.0 \cdot 10^{-9}$ /year, which means that almost all large releases are early.

2.5.2.2 Forsmark unit 3

The total core damage frequency for power operation is $1.8 \cdot 10^{-5}$ /year at Forsmark unit 3. Internal events make up 64 percent of the core damage frequency, external events 34 percent and fire and internal flooding 2 percent. The five most significant initiating events are:

1. Loss of offsite power
2. Transients (scrams) with unavailable turbine by-pass function
3. Other scrams
4. Transients (scrams) with loss of feedwater
5. Loss of 10 kV transformer.

The total core damage frequency for shutting down is $8.5 \cdot 10^{-8}$ /year at Forsmark unit 3. The dominating initiating events are loss of feed water and condenser, failed manual switch to cooling to heat exchanger in the shutdown cooling (RHRF), loss of offsite power and turbine scram with dump prohibition. The total core damage frequency for start-up is $1.2 \cdot 10^{-6}$ /year at Forsmark unit 3. The dominating initiating events are; high water inventory level, other scrams, loss of feed water and condenser, low level caused by not closed low pressure coolant injection system (LPCI) (323) valves and unwarranted opening of one pressure relief valve.

The total core damage frequency for cold shutdown is $3.5 \cdot 10^{-5}$ /year at Forsmark unit 3. The most significant events are different forms of manually initiated leakages, both above and beneath the core, and loss of shutdown cooling (RHRF) early in the cold shutdown.

In total this gives an overall core damage frequency of $5.4 \cdot 10^{-5}$ /year at Forsmark unit 3.

Out of the $1.8 \cdot 10^{-5}$ /year core damage frequency for power operation 99 percent of the sequences lead to acceptable releases (< 0.05 % of the volatile fission products). Non-acceptable releases have a frequency of $2.3 \cdot 10^{-7}$ /year. Large releases (> 10 % of the volatile fission products), which are a part of the non-acceptable releases, have a frequency of $1.2 \cdot 10^{-7}$ /year. Large early releases (which are a part of the non-acceptable releases and large releases) have a frequency of $5.0 \cdot 10^{-8}$ /year.

3 Results of the stress tests

3.1 Basic assumptions and methods

All analytical work for the evaluated events is carried out by the methods prescribed in the ENSREG document. This means that the ability of the plant's existing design to withstand the addressed events is described as the first step in the analysis. The basis for this description is the licensed design according to the Safety Analysis Report (SAR).

Thereafter, an assessment of the plant's ability to handle the events during stressed conditions is performed. This means that the events are not limited to the plant design levels. The impact is increased or protective functions removed to such a degree that safety functions are lost or fuel damage occurs. This analysis results in an assessment of the existing design analysis, an increased knowledge of plant behavior beyond design, and the identification of areas requiring further in-depth analysis or reinforcement.

In the analysis of plant characteristics beyond the design, realistic assumptions are applied. Furthermore, in accordance with the ENSREG document, engineering judgments are made when calculations beyond the design basis are unavailable.The results of the stress test are reported in accordance with the ENSREG document structure.

The Stress tests are carried out per unit and for the entire site.

Light equipment is not credited for the first 24 hours and heavy equipment not for the first 72 hours.

For further information regarding assumptions and methods see the
"Progress report 15th of august" [5].

3.2 Earthquake

This chapter is a summary of Forsmarks assessment of earthquake, for more information see the detailed report [6].

I. Design basis

Forsmark 1 and 2

At the time when Forsmark 1 and 2 were designed, no seismic safety requirements were imposed. In the early 1990s Forsmark decided that unit 1 and 2 should apply the "Swedish Earthquake" with an estimated probability of 10^{-5} per year (E-5) for verification of safe shutdown after an earthquake. This corresponds to horizontal Peak Ground Acceleration (PGA) of 0.09g - 0.11g.

A step-wise approach was implemented at unit 1 and 2 to fulfill the ability to withstand a Design Base Earthquake (DBE). All plant modifications after 1992 are designed to meet the DBE for safe shutdown. In addition, a Seismic Margin Assessment (SMA) has been conducted. SMA is a methodology that combines experience based methods and calculations to assess a plant's ability to withstand earthquakes. Systems, structures and components necessary for safe shutdown are identified and evaluated. With this method the earthquake level that the plant is likely to withstand can be estimated, and weak components can be identified.

Generally, it can be concluded that buildings, pressure vessels and cable routes withstand the seismic load. Some other systems, structures and components were identified that need further analysis and possibly mitigation measures. Components which did not fulfill the SMA-requirements can be categorized into:

- Anchorage of some mechanical equipment and electrical distribution cabinets
- Spurious activation from affected relays.

New safety requirements were introduced by the regulator in 2004, in the regulation SKIFS 2004:2 (now SSMFS 2008:17). For Forsmark 1 and 2, measures to fulfill these regulations shall be in place no later than 2013, according to an agreed transition plan. Regarding earthquakes, no measures were included in the transition plan. As a conclusion from this assessment, remaining issues from the SMA, which now have been highlighted in the Stress Test, need to be added to the transition plan.

The consequences of not having the remaining actions completed are judged to be acceptable, since the time to complete the modification plan is short and the probability of the event is low. Therefore the accumulated risk contribution due to earthquake is acceptable.

Forsmark 3

For Forsmark 3, buildings and systems important for nuclear safety were designed according to the design response spectrum specified in Regulatory Guide 1.60 scaled to 0.15 g for horizontal ground acceleration and 0.10 g for the vertical ground acceleration (PGA 0.15 g).

Since Forsmark 3 is licensed to fulfill seismic requirements, no additional verifications or design changes have been identified.

Severe Accident Mitigation System Forsmark 1,2 and 3

The severe accident mitigation systems installed after the TMI accident at all three units in Forsmark are designed in accordance with R.G. 1.60 (horizontal PGA 0.15 g).

Adequacy of the design basis

Compared to the design response spectrum based on R.G. 1.60, the “Swedish Earthquake" shows lower accelerations at low frequencies but higher accelerations at high frequencies. This is partly due to ground accelerations for the ”Swedish Earthquake” being dominated by small earthquakes at a close distance, while the R.G. 1.60 is dominated by large earthquakes at greater distances, and partly because the ground in Sweden is harder than in the United States.

From experiences obtained from real earthquake events it can be concluded that the impact of high-frequency earthquakes are highly overestimated by traditional analysis methods. Using the concept of damage potential it is possible to show that the E-5 earthquake does not exceed the intensity VI on the Modified Mercalli Intensity scale (MMI) which is considered harmless by most experts.

In the DBA analysis Forsmark uses traditional methods. The concept of damage potential is not widely used for design verification but implies that the design basis for Forsmark 1 and 2, and even more so for Forsmark 3, is highly conservative regarding the potential harm that the DBE may cause in Forsmark.

II. Evaluation of the margins

Margins against loss of fundamental safety function or severe core damage

For Forsmark 3, the margin can be generally estimated by utilizing the fact that methods for earthquake design are very conservative. By calculation of the median failure rate, a capacity of 2.3 times the R.G. 1.60-earthquake (PGA of 0.35g) can be estimated.

After completion of the SMA requirements for Forsmark 1 and 2, the margin can be estimated in the same way as for Forsmark 3 leading to a 2.3 times the Swedish Earthquake (PGA of 0.21).

Margins against loss of containment integrity

For estimation of margins against loss of containment integrity, analysis of plant response to an earthquake implies that the containment integrity is maintained at a level about 4 times higher than the Swedish Earthquake level. Reactivity control and containment filtering is assessed not to be jeopardized. The ability to isolate the containment manually has only been partly assessed.

As stated above, utilizing the concept of damage potential, it is reasonable to assume that the systems verified against an R.G. 1.60-earthquake can withstand an earthquake 3 to 5 times stronger than a Swedish Earthquake. Thus, it is also reasonable to assume that the severe accident mitigation systems at Forsmark 1, 2 and 3 retain their function at this level.

Earthquake exceeding DBE and subsequent flooding exceeding DBF

A combination of earthquake and extremely high sea level is considered extremely unlikely at the Forsmark Nuclear Power Plant. Flooding due to human activities is not considered a risk since there are no nearby rivers and dams of the magnitude required to produce any damage to the plant.

3.3 Flooding

This chapter is a summary of Forsmarks assessment regarding flooding, for more information see the detailed report [1].

I. Design basis

a) Flooding against which the plant is designed

Design basis flooding (DBF) for all three units at Forsmark Nuclear Power Plant is ground level. In 2011 ground level is equivalent to 3 m above normal sea level (+103.0).

The design level is based on the maximum sea level at Forsmark. This was calculated from measured sea levels 1895-1975 at the lighthouse “Björns fyr”. To this, a margin was added. Later the frequency for external flooding to ground level has been estimated with data from Sweden's Meteorological and Hydrological Institute (SMHI). The extrapolation of data for low frequency extreme sea levels contains many uncertainties. If a supposedly conservative extrapolation is used, the frequency for external flooding to ground level is in the interval of 10^{-6} to 10^{-5} per year.

All known phenomena (storm surge, short term local sea level change, heavy rain, tsunami, etc.) which may occur in Forsmark with a frequency higher than 10^{-5} per year is considered to be covered by DBF. The events are estimated to develop during a period of several hours.

DBF is thus considered to be adequate.

b) Provisions to protect the plant against the DBF

Protection against DBF for Forsmark 1 and 2 is achieved by placing equipment necessary for safe shutdown either above ground level or in water proof areas below ground level.

Protection against DBF for Forsmark 3 is achieved by designing all critical buildings so that they are able to withstand external flooding up to ground level.

All buildings, safety systems and power supply are designed to be able to cool the core and the fuel in the spent fuel pools during an external flooding up to DBF. Existing procedures support provisions to maintain the water intake function.

Operating procedures for high sea level prescribes reduced intake flow at alarm level, +101.6 (F3, +101.8). For cases relevant to the Forsmark site there should be significant time duration before the ground level could be flooded. It would be a matter of hours from the alarm initiation until the sea level may exceed DBF. This time duration is considered to be enough to bring the reactors to a safe state.

c) Plant compliance with its current licensing basis

Forsmark units 1, 2 and 3 comply with the licensing basis. The compliance is achieved by design requirements and in the requirement that safety systems shall have high availability.

II. Evaluation of the margins

d) Margins to severe core damage

If an external flooding is stressed above ground level (a few decimeters above DBF) the water level inside the buildings of the plant will eventually rise to the same level. For all three units in Forsmark, an external flooding above ground level is expected to lead to a vast amount of safety equipment being affected simultaneously when safety classed power supply fails. There are several doors that will delay the water flow considerably.

It should be noted that while an external flooding exceeding DBF can lead to core damage, the FCVF will not be affected. Water filling of the lower dry-well and pressure relief of the containment via the scrubber system are unaffected by flooding. To fill the containment with water, fire trucks can be used if the fire water system, situated below ground level, has failed.

External flooding is likely to be caused by a weather phenomenon that can be foreseen in weather forecasts days in advance. Hence ordinary procedure controlled actions are possible in order to prevent or mitigate the consequences. Bilge pumps and sandbags may for example be transported to Forsmark if needed.

When electrical power supplies are lost at flooding above DBF, cooling of the spent fuel pools are lost. The time to fuel damage is normally several days, which means that there is time for provisions, e.g. pumping water to the spent fuel pools with hoses.

Flooding above DBF is likely to affect all three units.

3.4 Loss of electrical power and of the ultimate heat sink

3.4.1 Loss of electrical power

This chapter is a summary of Forsmarks assessment regarding loss of electrical power, for more information see the detailed report [8].

Electrical power sources are:

- Two independent off-site power sources, the 400 kV national grid and the 70 kV regional grid.
- Plant generators capable for house load operations.
- Each unit has 4 redundant and functionally separated emergency diesel generators (EDG).
- Forsmark have a gas turbine common for all units located at the 70 kV switch yard.

a) Loss of off-site power (LOOP)

The following stepwise order of failure is assumed when off-site power is lost:

1. Loss of 400 kV national grid.
   No consequence.
2. House load operation will fail.
   No consequence.
3. Loss of 70 kV regional grid.
   No consequence.

When the 400 kV national grid, the 70 kV regional grid and house load operation fail, the diesel generators will start up the safety objects. The endurance for the diesel generators without external fuel supply is for Forsmark 1 and 2 at least 4 days and for Forsmark 3 at least 7 days. Lubrication oil has to be supplied every 10 days for units 1 and 2 and every 8 days for unit 3. The endurance time can be prolonged considerably if not all diesel generators operate simultaneously or at full power.

Considering the limited volumes required, it is reasonable to assume that fuel supplies from external sources can be made available after 72 hours even in the case of extreme weather conditions.

b1) Loss of off-site power and of on-site power (SBO).

When off-site power and on-site power is lost the following is postulated:

4. Emergency diesel generators will fail.
   No consequence.

It is postulated that all steps (1-3) above will fail including the emergency diesel generators. The safety functions will be fulfilled, powered by the gas turbine.

During loss of off-site power and on-site power (SBO) the gas turbine will energize the diesel bus bars in all units. The gas turbine has fuel for 4 days during full power operation, which will be extended considerably due to the fact that full load power operation is not required. Fuel supply from external sources can be available within 72 hours.

b2) Loss of off-site power and of on-site power (SBO).

When off-site power and on-site power is lost together with the loss of any other diverse back up sources the following is postulated:

5. Loss of alternative AC source (AAC, gas turbine)
   A severe accident scenario will be entered

It is postulated that all steps above (1-4) will fail including the gas turbine (AAC). If the gas turbine fails and can not power the safety systems, only battery powered systems will operate. A severe accident scenario will then be entered. The mitigation systems will provide filtered venting of the reactor containment.

Batteries for primary needs are designed for 2 hours operation. Batteries for the severe accident monitoring are designed for 24 hours. Realistically the batteries are expected to operate much longer, from 4-12 hours and much more than 24 hours, respectively.

3.4.2 Loss of Ultimate Heat Sink (UHS)

This chapter is a summary of Forsmarks assessment regarding loss of ultimate heat sink, for more information see the detailed report [10].

The cooling water for Forsmark 1, 2 and 3 is taken from the archipelago southeast of the power station. It is led by natural ponds and excavated channels to the intake buildings. This provides significant protection against many external events. The common channel section has a bridge with a foam beam, which prevents floating objects from reaching the intake building.

There is one intake building, shared by Forsmark 1 and 2 and one for Forsmark 3. The intake buildings are equipped with rough gratings and basket filters for treatment of pollution.

In case of intake channel blockage for Forsmark 3, cooling water can be re-circulated from the outlet tunnel. This is performed automatically.

Considered scenarios for blockage of the intake channel are:

- Blockage from packed ice.
- Blockage from organic materials from the bottom of the sea.
- Blockage from oil.
- Blockage from foreign materials collected as a consequence of severe weather conditions.

All of these scenarios are considered to develop slowly relative to the rate at which countermeasures can be established. The countermeasures can be either in the form of activities to prevent the blockage or in the form of operational activities to remedy a loss of cooling water. Hence instantaneous complete blockage of the intake channel is considered to bee highly unlikely.

A cooling water flow of less than 1% of the nominal flow is sufficient to cool and remove the residual heat.

In case of total loss of primary UHS the sequence for Forsmark 1, 2 and 3 will be relatively similar if it is assumed that the ability to take water from the outlet tunnel is unavailable for Forsmark 3.

The following stepwise order of failure is assumed when the UHS is lost:

1. Blockage of intake channel.
   No consequence.
   >1% of nominal flow is still assumed to be possible to utilize for cooling purposes. Main cooling water pumps will be stopped manually according to operating procedures.

2. Complete blockage of intake channel.
   No consequence.
   All main cooling water pumps will be stopped. The unit will scram within 1 minute. If more than one unit is affected within 15 min the external grids may be lost.

3. Complete blockage of intake and outlet channels.
   No consequence.
   A large volume of water is still available.

4. As above and loss of siphon over the condenser.
   Safety functions will be lost but no core damage.
   The water volume for cooling is now limited to the intake channel. The water consumptions for the cooling systems will within soon empty the intake channel. Hence, diesel generators will be lost due to loss of cooling. At the same time the residual heat removal systems will be lost. Core cooling is provided from the demineralized water distribution system. The residual heat will be removed from the containment via the FCVF. After about one day additional water needs to be supplied to the demineralized water tank.

   In unit 3 there is an additional function where water can be recirculated from the outlet to the intake channel.

5. As above and with loss of gas turbine.
   Core damage will occur. External releases of radioactivity will be limited by the FCVF. This situation also constitutes a combination of SBO and UHS.

3.4.3 Loss of primary UHS and SBO

The sequence for the loss of primary UHS and SBO is basically the same as the loss of primary UHS. The difference is that the diesels will be unavailable immediately.

If the gas turbine is assumed to fail, the sequence is similar to loss of UHS without the gas turbine available. This scenario will lead to a severe accident.

3.4.4 Spent fuel pools

There are several systems installed for cooling of the spent fuel pools. All of these require either electrical power supply or operating fire diesel engines. In case of loss of cooling acceptable response times are long, generally in the order of 24 h before the water is boiling and several days before fuel damage occurs. During special occasions all fuel may be placed in the spent fuel pools. The response time will then be shorter. During a boiling scenario, the required amount of water to maintain the water level in the spent fuel pools is a few kg/s.

Earthquake

The spent fuel pools withstand design base earthquake with a substantial margin. Cooling of the spent fuel pools will be maintained by the use of the normal cooling systems.

Flooding

Cooling of the spent fuel pools will be maintained by the use of the normal cooling systems up to a few decimetres above DBF.

Station Black Out (SBO)

If all electrical powered systems would fail, cooling can still be maintained by compensation of boiled water using the diesel powered fire water system.

Loss of Ultimate Heat Sink (UHS)

Cooling can still be maintained by compensation of boiled water using the diesel powered fire water system.

For long term operation, the fire protection system is fed by gravity from the freshwater supply.

SBO and loss of Ultimate Heat Sink (UHS)

The same sequence as loss of UHS alone will result.

3.5 Severe accident management

This chapter is a summary of Forsmarks assessment regarding severe accident management, for more information see the detailed report [8].

According to the government decision of February 1986, all Swedish reactors were required to prevent unacceptable releases to the environment in case of a severe accident. The decision included prescriptive technical measures as well as administrative. This resulted in a set of systems, FCVF.

As far as administrative requirements were concerned, the following interpretation was made:

- Procedures shall consider all possibilities to cool the core with significantly degraded functions of operation and safety systems.
- Procedures shall give guidelines to bring the facility into a stable state, even after a severe core damage

To meet the requirements a comprehensive set of accident management procedures were developed. These cover the entire process from scram to a severe accident with a stable end-state. The emphasis in the procedures is on actions and measures to take in the short time perspective (within 1 hour), but the coverage is up to 24 hours, and operators are trained accordingly.

For severe accidents, in the longer perspective, when an emergency organization has been established, the procedures are more generalized and more based on symptomatic actions. Operational management in the emergency control centre (KC) assists in these cases to handle the situation.

Emergency response organization

<u>Staffing, resources and shift management</u>

Available resources on site (24 h per day) for an immediate response are:

- The shift teams including security guards
- Engineer on duty
- Fire brigade

The emergency preparedness organization at Forsmark consists of a common organisation for the entire plant and a specific organisation for each unit. The organization is initiated if a severe accident should occur.

The emergency preparedness organisation can be onsite within 1 to 2 hours provided that the roads are available.

<u>Use of off-site technical support for accident management</u>

Vattenfall has a team that will provide off-site engineering support when requested.

<u>Procedures, training and exercises</u>

The validity of the organizational procedures, education and basic training is regularly confirmed by large scale exercises. More in detail, for operators and other staff used in the organization education frequently recurs as described in education programmes.

<u>Possibility to use existing equipment</u>

The emergency control centre is equipped with IT-applications for support, calculation and communication which include several types of communication technology.

<u>Provisions to use mobile devices</u>

At Forsmark plant is an internal fire brigade consisting of four fire fighters. They have at their disposal a fire engine, on site in the fire station. One specifically designated person at each of the shift teams has a role to support actions when needed. In case of a major event, outside assistance might be necessary. This can be obtained from the community of Östhammar. Response-time is estimated to be about 20 minutes.

Forsmark has a fire engine to be used for sprinkling of the containment.

Management of radioactive releases

Forsmarks emergency preparedness organization has tools to calculate the source term for a current event, the degree of the resulting core damage and the release depending on weather and wind direction. These tools may also be used without computers.

Management of workers doses

Monitoring the worker doses is managed by the leader of the radiation protection team to assure that workers are not exposed to more radiation than accepted. There are procedures available for acceptable radiation dose constrains for different scenarios. Those levels are pre-calculated. Ordination of iodine tablets and the use of protective equipment are also managed by procedures.

Communication and information systems

The existing loudspeaker system is used for alerting messages. Alert and communication can be made from the control room, the security monitoring centre or the Emergency Control Centre (KC). Alerts to external emergency organisations are done via “SOS Alarm” which is equipped with alarm lists if forwarding of alerts is necessary.

External contacts can, apart from the conventional communication systems, also be maintained by use of the Swedish defence radio network.

Long term post accident activities

The staff within the emergency preparedness organization are all Forsmark staff and ISS Facility Service Staff. They belong to 25 different functions in the organization with 2-9 people per function. The total staff number is about 190. In addition there are technical support organisations in order to support the individual units.

The Swedish plants have established joint agreements for resource enhancement regarding radiation protection and engineering. Therefore resources are also available outside of the Forsmark organization.

Facility availability

If the main control room is not available the plants are equipped with emergency shut-down and monitoring panels in a physically separated location. For control, monitoring, communications and emergency lightening battery systems are needed.

The Emergency Control Centre is located at the site . The Control Centre is sheltered.

Stress evaluation of the emergency response organisation

More than one affected unit

The current accident management preparedness organisation is basically sized to cope with a severe accident at one of the three units at Forsmark. If two or three units are simultaneously affected, staffing and shift management resources may not be sufficient to cope with the situation and perform the required accident management measures. Presently all trainings and exercises are performed assuming failure at one unit.

Degradation of infrastructure

Degradation of the infrastructure around the site may prevent emergency response staff to reach the plant. The FCVF functionality makes it possible to secure all scenarios against major releases of radioactivity with a very small staffing on site. It is estimated that the around the clock staffing is adequate to meet this need.

External communications may be affected. In addition to the conventional communication systems, Forsmark is connected to the Swedish defence radio network which is considered to be very robust.

Long term post accident activities

For water filling of the containment, the ordinary fire diesel pumps are used. If they are lost, mobile pumps (light equipment) need to be made available within 8 hours.

No heavy equipment has been identified that need to be in place within 72 hours.

In the long term the local organisation will need a substantial external resource support. The pre-planning for this is limited.

4 Conclusions

Compliance with current licensing basis

Earthquake

At the time when Forsmark 1 and 2 were designed, no seismic safety requirements were imposed.

New safety requirements were introduced by the regulator in 2004, in the regulation SKIFS 2004:2 (now SSMFS 2008:17). For Forsmark 1 and 2, measures to fulfill these regulations shall be in place no later than 2013, according to an agreed transition plan. Regarding earthquakes, no measures were included in the transition plan. As a conclusion from this assessment, remaining issues from the SMA, which now have been highlighted in the Stress Test, need to be added to the transition plan.

The consequences of not having the remaining actions completed are judged to be acceptable, since the time to complete the modification plan is short and the probability of the event is low. Therefore the accumulated risk contribution due to earthquake is acceptable.

Forsmark 3 complies with the current licensing basis.

Flooding

The assessment demonstrates that external flooding scenarios complies with current design basis. The design basis is adequate.

Loss of electrical power and loss of the ultimate heat sink

In summary loss of off-site power and on-site power (SBO) is managed when the gas turbine (AAC) is available.

As reported in the WANO SOER 2011-2, some reliability issues for the gas turbine has been identified. Recent test results show adequate performance. Additional measures for increased availability has been identified e.g. modified test procedures and intervals.

The design requirements concerning UHS are met.

Cooling of spent fuel pools

The design requirements concerning the spent fuel pools are met.

Severe accident management

The requirements concerning the severe accident management are met.

Conclusions from the stress test

Earthquake

The stress test has not revealed any weaknesses associated with cliff-edge effect.

For Forsmark 3, the margin can be generally estimated by utilizing the fact that methods for earthquake design are very conservative. By calculation of the median failure rate, a capacity of 2.3 times the R.G. 1.60-earthquake (PGA of 0.35g) can be estimated.

After completion of the SMA requirements for Forsmark 1 and 2, the margin can be estimated in the same way as for Forsmark 3 leading to a 2.3 times the Swedish Earthquake (PGA of 0.21).

Safety systems and severe accident mitigation systems for Forsmark 1, 2 and 3 can be stressed to a level of 3 – 5 times stronger than the Swedish Earthquake.

Flooding

The evaluation implies that the design base is conservative. All units withstand a stress of a few decimeters above DBF (3 m above sea level).

If the DBF limit is exceeded the reactivity function will be activated since it is fail-safe, all other safety functions may be affected. The FCVF capability is judged to be unaffected.

The assessment has identified areas for procedure improvements that can increase the ability to withstand high water levels.

Loss of electrical power and loss of the ultimate heat sink

The station has a robust design with respect to loss of off-site and on-site power.

Loss of off-site power (LOOP) will be handled by the ordinary back up diesel generators.

At the loss of off-site power and on-site power (SBO), the gas turbine will energize the diesel bus bars in all units. Cooling of the core with the auxiliary feed water system can then be continued for several days.

Loss of off-site power and on-site power and gas turbine will result in core damages and the mitigating systems will handle the situation with filtered ventilation of the reactor containment.

The station has a robust design with respect to loss of UHS. There are several functions that will protect the plant from severe blockage of the intake cooling water. A cooling water flow of less than 1% of the nominal flow is sufficient to cool and remove the residual heat.

In the scenario with complete loss of UHS, core cooling can be provided using the demineralized water distribution system and power supply from the gas turbine.

When power supply from the gas turbine is lost and the ultimate heat sink is lost, a severe accident scenario will be entered.

Cooling of spent fuel pools

There are no systems available for cooling of the spent fuel pool in case of total loss of AC. Boiled off water can be compensated for using diesel powered fire water pumps or with fire trucks at Forsmark 1 and 2 using external flanges. At Forsmark 3 hoses have to be brought up to the fuel pool manually which can be difficult due to high levels of radiation.

Severe accident management

The stress test conclusion is that current severe accident management is adequate when one unit is affected. It is not designed for an event with multiple units affected. It should be noted that a severe accident scenario is within the design requirements of the plants, where the FCVF plays an important role.

A design requirement for the FCVF is that electrical power is restored within 24 hours. The stress test implies that the function will endure for several days. It is concluded that the FCVF is functional assuming external supplies, mainly shielding materials, can be made available within 72 hours.

The around the clock organisation is estimated to be able to support the FCVF.

External communications may be affected. In addition to the conventional communication systems, Forsmark is connected to the Swedish defence radio network which is considered to be robust.

For water filling of the containment, the ordinary fire diesel pumps are used. If they are lost, mobile pumps (light equipment) need to be made available within 8 hours.

No heavy equipment has been identified that need to be in place within 72 hours.

In the long term the local organisation will need a substantial external resource support. The pre-planning for this is limited.

5 Provisions envisaged to prevent cliff edge effects

Provisions envisaged in order to prevent cliff edge effects are listed below in general terms. It should be noted that specific measures need further investigation. The main areas for improvement are listed below:

Earthquake

The stress test has not indicated any areas for improvement regarding earthquake.

Flooding

The stress test has identified procedure improvements that can increase the ability to withstand high water levels.

Loss of electrical power and loss of the ultimate heat sink

The stress test has identified the gas turbine as essential to avoid core damage. Thus, to make the unit more robust and independent of the Gas Turbine in case of SBO, the following areas for improvement have been identified:

- Diversified core cooling function independent of the SBO and the loss of UHS scenarios.

- Improvements of operating procedures to better mitigate the effects of a prolonged SBO.

Furthermore, the stress test has identified a potential for improvements of the robustness of the cooling water inlet channel at Forsmark 1 and 2.

Spent fuel pools

The stress test has identified the following improvements for the spent fuel pools:

At Forsmark 3 hoses have to be brought up to the fuel pool manually which can be difficult due to high levels of radiation. There is a need to develop a strategy for this.

Severe accident management

The stress test has identified the following areas for improvements:

- The current emergency preparedness organisation is sized to cope with a severe accident at one of the three units. There is a need to modify procedures and routines to handle a simultaneous emergency at all three units.

- In a scenario with lost containment cooling function, pressure relief using the FCVF may be required in the long-term. For such case, the water level in the MVSS need to be adjusted to ensure that the filtering function is not degraded. Presently there are no instructions or strategies showing how this can be done.

6 References

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