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Swedish Radiation Safety Authority

European stress tests
for nuclear power plants

National progress report – Sweden
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Abstract

In the light of the Fukushima accident in Japan, the European Council of March 24th and 25th declared that the safety of all EU nuclear power plants should be reviewed. According to the "Stress tests" specifications defined by the EU, all EU Member States shall, not later than September 15th, 2011, provide a national progress report for the stress tests.

The following report contains information about the progress of the Swedish reassessments.

The Swedish stress tests are being carried out according to schedule. Remaining work, including reviews, is expected to follow EU "Stress tests" specifications.

Due to the timeframe of the stress test process, some of the engineering judgment supporting the licensees' assessments will not be available for scenarios not included in the current design bases. In these cases, engineering judgments based on knowledge and experience are used.

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1 Introduction and background

Following an extraordinary meeting held between March 24th and March 25th, 2011, the Council of the European Union declared that Member States are 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 experiences gained from the situation in Japan so that the assessments can 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.

On May 12th, 2011, the Swedish Government decided on an assignment for the Swedish Radiation Safety Authority (SSM), including the national report in accordance with the criteria agreed within ENSREG.

On May 25th, 2011, SSM required all Swedish nuclear power plants (NPP) to perform reassessments 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. Furthermore, the Swedish stress tests also include the Swedish central interim storage facility for spent nuclear fuel (CLAB) and the scope of the Swedish stress tests has been extended to contain evaluations of extreme weather conditions.

2 The Swedish regulations and transitional provisions regarding design and construction

The former regulatory body, the Swedish Nuclear Power Inspectorate (SKI), issued new regulations and general advice concerning “Design and construction of nuclear power reactors” that entered into force on January 1st, 2005. These regulations were later transformed into SSM regulations after merging SKI and the Swedish Radiation Protection Authority (SSI) in 2008. As the new regulations are more extensive than previous regulations, various backfitting measures were foreseen. Therefore, the authority decided to put the regulations into effect through certain transitional provisions. The new regulations include requirements on resilience to different natural phenomena.

Since the ten operating NPPs in Sweden have different prerequisites for complying with the regulations on design and construction, an assessment of the consequences was made for each reactor. This assessment included determining whether further analyses and backfitting were needed in relation to the requirements. During 2006 and 2007, decisions were issued for each Swedish reactor.

The regulations stipulated that measures to comply with certain requirements shall be implemented no later than the time decided by SSM to allow licensees to implement necessary backfitting measures. For requirements related to natural phenomena, implementation measures were set for the different NPPs between 2012 and 2013. As a consequence of the set timetable in the transitional provisions, Swedish NPPs are not yet fully analysed for earthquakes and flooding.

3 The Swedish review process

The overall timetable for the EU "stress tests" means that licensee reports shall be delivered to the national authority on October 31st, 2011, and the national report shall be delivered to the EU on December 31st, 2011.

Due to the limited timeframe that would not allow much time for corrective measures if licensee reports did not completely fulfil EU "Stress tests" specifications, early reports describing in more detail the starting points and assumptions that will be implemented in the Swedish stress tests were submitted to SSM on June 8th, 2011. Starting points and assumptions, together with supplementary documentation, have been reviewed by SSM. Results from these reviews have been documented and communicated with licensees and should be used as guidance for submission of the final licensee reports.

A further means to assure quality in the Swedish stress tests is to convene regular meetings between the licensees and SSM. These meetings highlight progress and open items and have been held since June 2011.

Several open items have been identified during discussions between SSM and the licensees. These open items are followed up continuously by SSM and are discussed further in section 5 in this report. Open items are also monitored for traceability and will be closed before starting the review of the licensees' final reports.

4 Site characteristics

There are ten operating NPPs on three sites in Sweden: Ringhals on the Swedish west coast and Forsmark and Oskarshamn on the Swedish east coast. All Swedish NPPs are light water reactors with prestressed concrete containments with steel liners. The BWRs are of ASEA-Atom BWR designs with pressure suppression type containment. The PWRs are of Westinghouse PWR designs with large dry type containment.

4.1 Brief description of sites

The Ringhals site is situated on the Värö peninsula on the coast of Kattegat, 70 kilometres south of Gothenburg. There are four units on the site. The license holder is Ringhals AB. Ringhals Unit 1 is a BWR design and the other three units are of PWR designs.

The Forsmark site is situated 150 kilometres north of Stockholm on the coast of the Baltic Sea. There are three BWR units on the site. The license holder is Forsmarks Kraftgrupp AB.

The Oskarshamn site is located on the Simpevarp peninsula 330 kilometres south of Stockholm on the coast of the Baltic Sea. There are three BWR units on the site. The license holder is OKG Aktiebolag.

The central interim storage facility for spent nuclear fuel, CLAB, is also located at the Oskarshamn site. The storage facility is owned and operated by the Swedish Nuclear Fuel and Waste Management Company, SKB. CLAB is an underground storage facility with spent fuel pools about 30 metres down in the bedrock.

4.2 Main characteristics

Unit	Reactor type	Commercial operation	Thermal power MWt	Date of first criticality
R1	BWR	1976	2540	Aug 20th, 1973
R2	PWR	1975	2652	Jun 19th, 1974
R3	PWR	1981	3144	Jul 29th, 1980
R4	PWR	1983	2775	May 19th, 1982
O1	BWR	1972	1375	Dec 12th, 1970
O2	BWR	1975	1800	Mar 6th, 1974
O3	BWR	1985	3900	Dec 29th, 1984
F1	BWR	1980	2928	Apr 23rd, 1980
F2	BWR	1981	2928	Jun 11th, 1980
F3	BWR	1985	3300	Oct 28th, 1984

4.3 Accident management measures

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 society.

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

As a result of the accident, separate scrubber filter systems were installed for each unit in separate buildings. At the Oskarshamn site, Unit 1 and Unit 2 have a shared scrubber filter system. However, this system is designed to withstand accident conditions simultaneous for both units.

4.4 Significant differences between units

The Swedish NPPs are of different ages and reactor system designs. Ringhals Unit 1 and Unit 2, and Oskarshamn Unit 1 and Unit 2, are of the oldest reactor system designs originally with two trains of safety systems. Forsmark Unit 1 and Unit 2 and Ringhals Unit 3 and Unit 4 represent the next reactor system design with more than two trains of safety systems. Forsmark Unit 3 and Oskarshamn Unit 3 have four separate trains of safety systems and represent the newest reactor system design. Extensive backfitting has taken place in all Swedish NPPs. All Swedish NPP sites are situated on stable bedrock and located on the coastline of Sweden and are supplied by seawater as the ultimate heat sink.

4.5 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 regulations, all Swedish NPPs must be analysed using probabilistic methods to supplement the basic deterministic safety analyses and be included in the Safety Analysis Report.

All operating NPPs are expected to perform complete plant-specific Level-1 and Level-2 PSAs, including all operating modes and all relevant internal and external hazards for the sites. 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.

The main results of the PSAs will be presented in the final report.

5 Extreme situations assessed

The licensees' stress tests of extreme situations are in progress and are being performed according to EU "Stress tests" specifications.

The following sections will give an area-specific description, which includes status and open items.

For every extreme situation, assessments will be performed both for single units and for all units on the specific site.

5.1 Earthquake

The seismic classified systems and structures will be re-evaluated against the design basis earthquake (DBE) in accordance with the Safety Analysis Report. This is characterised by a set of Swedish ground response spectra corresponding to an exceedance frequency probability of 1E-5 per site and year. No further analyses will be performed for seismic classified systems and structures in addition to this level. For the containment including the filtered venting system, informed judgments will be applied. Assessments will be performed for ground response spectra corresponding to an exceedance frequency of 1E-7 per site and year, which is the strongest seismic level developed for Swedish conditions. At this level, it may be challenging to verify the integrity of the safety functions and the containment.

Aside from the stress tests, the Swedish regulations concerning “Design and Construction of Nuclear Power Reactors”, issued in 2005, state the following: “The nuclear reactor shall be dimensioned to withstand natural phenomena and other events […]”. In the general advice for this regulation, an earthquake is pointed out as an example of this kind of natural phenomena. It should be noted in this context that only the two youngest Swedish NPPs, Oskarshamn 3 and Forsmark 3, were originally analysed and designed to withstand a specified earthquake.

5.1.1 Status

Assessments of earthquakes are proceeding according to schedule. The remaining work includes renewed verification of plant design against DBE as well as the verification beyond DBE. The licensee safety review of preliminary results is expected to start in September 2011.

5.1.2 Open items

The eight oldest Swedish nuclear power plants were not initially analysed and designed to withstand a specified earthquake and are not fully verified against DBE, which will limit the possibility to perform complete analyses.

Furthermore, analyses will be limited to a seismic load level of 1 E-7 per site and year. Additionally, the uncertainties from “engineering judgments” may be difficult to evaluate.

5.2 Flooding

Design basis evaluations are primarily based upon conservative analyses and information contained in the Safety Analysis Reports. For analyses beyond design basis, informed judgments are applied.

The level of the Design Basis Flood (DBF) is stated for a probability level of 1E-5 per site and year according to data from the Swedish Meteorological and Hydrological Institute (SMHI). The DBF is defined in each unit’s Safety Analysis Report.

Sloshing as a phenomenon may occur if the main cooling water pumps suddenly stop, which will cause a raised water level in the seawater inlet chamber. This is considered for those NPPs where the phenomenon is possible.

Only external flooding is considered. Flooding caused by failure of equipment inside the plant is not considered. Tsunamis are determined to be extremely unlikely in Sweden and are therefore not analysed further.

The analyses will be performed up to the level of flooding where the plant will suffer severe damage to the fuel.

5.2.1 *Status*

Assessments of flooding are proceeding according to schedule. The licensee safety review of preliminary results is expected to start in September 2011.

5.2.2 *Open items*

The application of "engineering judgments" may introduce undefined uncertainties in the results. Hence, it is essential that the distinctions between the levels of judgment are defined and described.

Additionally, the sole "cliff-edge level" analysed is the maximum level of flooding where severe damage to the fuel is likely to occur. Further discussions between the licensees and SSM on how to report the levels where different safety functions are malfunctioning are necessary.

5.3 Weather conditions

The technical scope for the analyses of weather conditions is not defined in as much detail as other phenomena in the EU "Stress tests" specifications.

The aim of the weather condition report is to describe the current status of the plant and will therefore be based on the existing Safety Analysis Report and instructions. The description will summarize the existing documentation, but refer to the original documentation regarding details.

A specification of the design basis weather phenomena is available in each unit's Safety Analysis Report.

5.3.1 *Status*

Assessments of weather conditions are proceeding according to schedule. The licensee safety review of preliminary results is expected to start in September 2011.

5.3.2 Open items

The licensees have proposed that weather conditions will not be analysed as a separate item in the stress test reports. SSM will evaluate whether this is sufficient.

Implementations of bad weather conditions in the stress tests as an aggravating circumstance will also be determined.

5.4 Loss of electrical power

The EU "Stress tests" specifications are interpreted as a renewed safety evaluation of the NPPs' ability to withstand prolonged loss of power, regardless of the reason for this occurring. This approach is intended to cover a broader spectrum of other (except for earthquakes and flooding) initiating events that may influence the plant.

Design basis evaluations are primarily based upon conservative analyses and information contained in each unit's Safety Analysis Reports. For analyses beyond design basis, informed engineering judgments are to be applied.

The power supply system design, connections to external networks, diesel generators, gas turbines, etc. will be described. As power supply systems are different for different units and also differ between Sweden and other countries, it is necessary to describe the structure of the grid and the particular facility in order to comprehend the content of the underlying chapters.

The disposition of the tentative text on loss of electrical power reviewed so far is very similar between licensees and briefly explains how each item from EU "Stress tests" specifications shall be handled. The description tentatively covers the three cases "Loss of outside power" (LOOP), "LOOP + loss of ordinary back-up source" and "LOOP + loss of ordinary back-up source and loss of any other diverse back-up source". As far as concerns coping time for different gas turbines, diesel generators and batteries, descriptions are planned for all cases.

5.4.1 Status

Assessments of loss of electrical power are proceeding according to schedule. The licensee safety review of preliminary results is expected to start in September 2011.

5.4.2 Open items
In-depth knowledge in electrical system design is vital not only for the assessment of "loss of power" but also for most of the other stress test scenarios. Therefore the identification and allocation of resources, in order to interact also with other parts of the stress test assessments, is an area of concern for the licensees.

5.5 Loss of ultimate heat sink and Loss of ultimate heat sink combined with loss of electrical power
Design basis evaluations are primarily based upon conservative analyses and information contained in the Safety Analysis Reports. For analyses beyond design basis, informed judgments are applied.

Since capabilities, functions and systems differ for each plant, heat sinks will be described for each Swedish NPP design. It is well known that design specifics can strongly influence an NPP's capability to handle a loss of ultimate heat sink. For example, the design of the seawater intakes and the presence of steam generators or an isolation condenser will significantly influence the analyses.

5.5.1 Status
Assessments of loss of ultimate heat sink, and loss of ultimate heat sink combined with loss of electrical power, are proceeding according to schedule. The licensee safety review of preliminary results is expected to start in September 2011.

5.5.2 Open items
The application of "engineering judgments" may introduce undefined uncertainties in the results. Hence, it is essential that the distinctions between the "levels" of judgment are defined and described.

The definition of alternative ultimate heat sink has not been fully settled. It will be further discussed as to whether or not the capabilities of steam generators together with the atmosphere for PWR, and the capability of an isolation condenser for BWR, should be considered alternate ultimate heat sinks.

5.6 Severe accident management
Accident management measures currently in place for coping with the following four cases will be described:

a) Loss of core cooling

b) Preservation of containment integrity in the event of fuel damage

c) Loss of containment integrity

d) Loss of cooling in the fuel pool

This description will include the following instruction packages:

- System-oriented emergency operating procedures
- Unit-specific emergency operating procedures
- Overall symptom-based emergency operating procedures
- Severe accident management guidelines

Besides descriptions and analyses of the existing accident management measures and corresponding instructions, all relevant aspects of emergency preparedness implemented at the site will be described and analysed.

The description of accident management will consist of two parts: one containing a general description, and one providing a more detailed review of existing strategies, including the strategy for handling the aggravating factors specified in EU “Stress tests” specifications. A similar approach will be used for emergency preparedness, which will be evaluated when considering the main accident management strategies for handling the four accident scenarios mentioned above as well as considering the aggravated situation at the site as specified in the ENSREG document. This includes estimations of recourses, staffing/competence and shift organisation considering long-term accident conditions (more than 24 hours).

Performance of the MVSS will be described and analysed considering long-term accident conditions (more than 24 hours). Analyses are performed to identify cliff-edge effect, specifically related to the performance of the MVSS. This includes estimating the time after which the venting system will have reduced capability to limit the radioactive releases to the environment. The time estimated in the long-term analyses will be a parameter independent of the assumptions and results of analyses in the other areas of the stress tests.

5.6.1 Status

Assessments of severe accident management are proceeding according to schedule. The licensee safety review of preliminary results is expected to start in September 2011.

5.6.2 Open items

The function of the common MVSS filter between Oskarshamn Unit 1 and Unit 2 needs to be verified for accident conditions simultaneous at both units.

Additionally, the assessment of hydrogen accumulation and hydrogen combustion in reactor buildings will need further evaluations. Moreover, the containment chemistry (e.g. pH level) when considering long-term accident conditions may be difficult to assess.