Critical Review of the EU Stress Test

performed on Nuclear Power Plants

Study commissioned by Greenpeace

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Wien, Hannover, May 2012

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EXECUTIVE SUMMARY

The March 2011 accident at the Fukushima I nuclear power plant proved that highly unlikely incidents cannot be excluded. Contrary to accepted practice Probabilistic Safety Assessments (PSA) do not constitute a sufficient basis to declare a plant operation safe. Safety of nuclear power plants (NPPs) needs to be backed by deterministic assessments, which excludes initiating events and accident scenarios only if they are proven to be physically impossible.

Events at Fukushima compounded public mistrust towards nuclear power worldwide. In Europe, the European Commission welcomed a suggestion by the government of Austria to conduct stress tests at all nuclear power plants in the European Union. The EU nuclear safety regulators – ENSREG – took over this task. The tests were introduced to improve confidence in the safety of European NPPs. In particular, they should examine the consequences of earthquakes and floods, and the combination of events previously excluded. However, the tests would be limited in scope: safety features such as ageing or design faults would not be taken into account.

The EU stress test focused on analysing of the plant's robustness to cope with consequences of loss of power including station black-out, loss of ultimate heat sink[1], and a combination of both. Safety reserves (margins) should also be assessed.

The best outcome of the stress test could be an analysis of multiple faults and common mode failures at the NPP sites. "Cliff edge" effects, which could result in core melt, would be investigated and improvements proposed for water and power supply in emergency situations.

The lesson from Fukushima is clear: take the unthinkable into account and develop adequate emergency measures for the protection of the population in the densely populated regions of Europe. The exclusion of unlikely accidents from the analysis is not justified without a deterministic verification.

The majority of the European reactor fleet is old, and based on decades-old design. Not all operators have reassessed the site hazards in compliance with state-of-the-art methodologies.

This report offers a review of selected NPPs based on National Stress Test Reports. The review details the main weakness of each reactor identified by the national regulator and the remedial measures suggested in its report. Important shortcomings not mentioned in the National Stress Test Report are listed and discussed at the end of each chapter.

Those evaluations do not claim to be exhaustive, but it is hoped that the findings will contribute to a more complete understanding of nuclear power plant safety in Europe.

Earthquake

In many countries seismic threat was not considered in the design of early NPP, such as Magnox and other old reactors. The UK did not routinely include earthquakes risk in the design basis for nuclear facilities until the early 1980s. New methodologies in seismic hazard assessment have since been introduced, but not all operators have reassessed their design bases in line with state of the art methodologies.

If operators *have* considered seismic hazards, their assumptions for assessing the 'safe shutdown

1 A nuclear power plant needs an Ultimate Heat Sink (UHS) to remove heat from the primary cooling circuit and other vital systems necessary to mitigate a severe accident. Usually, the ultimate heat sink is a large body of water such as a river or a lake. If the required heat removal function fails, then the nuclear plant is bound by regulation to shut down. The water from the "Ultimate Heat Sink" is passed through large heat exchangers where it cools other mediums. If this function fails a core melt could occur. Therefore several diverse emergency cooling procedures have to be followed (so-called severe accident management).

earthquake' (SSE) differ significantly concerning the size of the region and the recurrence period applied. International Atomic Energy Agency (IAEA) guidelines recommend using the maximal horizontal peak ground acceleration (PGA) in 10,000 years and as a minimum the PGA value of 0.1g.

Reassessments of seismic hazards for NPPs often arrive at the conclusion that the SSE is not sufficient (e.g. in Krško, Mochovce and Ringhals). If safety margin assessments conclude that improvements are required, such as stronger fixation of equipment, vibration dampers for pipes, than strengthening of non-seismically qualified structures is necessary. Also new seismically qualified equipment needs to be installed, which guarantees the safe shutdown and cooling of the plant.

Earthquakes can cause problems with reactor shutdown if the geometry of the core is damaged during the event. Storage racks in the spent fuel pool can also be damaged due to an earthquake, disabling the cooling of fuel. In both cases partial fuel melt could occur.

Earthquakes also have significant indirect effects, causing fires or broken pipes, followed by flooding. The damage caused will depend on the location of equipment inside the plant, for example the flooding of rooms with electrical equipment could disturb the electricity supply needed to operate equipment such as pumps. Or an earthquake might initiate a dam break and could cause a flood wave that damages water intake buildings.

Flood

The need for cooling requires nuclear power plants to be sited beside the sea or a large river. IAEA guidelines recommend that regulators assess the potential of flooding due to rain, snow melt, high tide, storm surge, seiche and wind waves. The flood threat at many NPP sites has increased in recent decades, on the one hand due changed conditions (e.g. climate change, dam construction, reduction of natural flood plains) and on the other hand new methods of hazard assessment result in better understanding of the hazards and thus higher safety requirements.

Flooding of a NPP site poses a common cause failure for safety related systems, such as emergency power supply or the electric switchyard, while flood debris could block the pumps. The result could be total loss of ultimate heat sink (UHS). Most NPPs do not have a separate and independent alternative UHS (e.g. Muehleberg and Krško are planning to equip their NPP with a second independent cooling water source). Not a single French nuclear power plant has an alternate heat sink. The vulnerability of UHS was highlighted by the events of clogging and partial loss of the heat sink at Cruas and Fessenheim in 2009.

<u>Safety Issues</u>

Ageing

The Critical Stress Test Review considers 12 sites with 30 reactor units. These units are between nine and 40 years old, and have an average age of 28. Two units have been operating for about 10 years (Temelín) and two have operated for 40 years (Wylfa). Ignoring the ageing effects of safety-relevant equipment is a severe shortcoming of the EU stress tests.

The quality of the plants' safety-related systems and components, such as the material of pipes, reactor vessel, valves and pumps, control and instrumentation equipment is not investigated in the ENSREG stress test. The test takes no account of degradation effects, even though these could significantly aggravate the development of an accident caused by an external event.

The safety design of nuclear power plants is crucial for preventing as well as dealing with incidents or accidents, but is not part of the stress test. The safety designs of all operating plants are out-dated and show deficiencies compared to modern safety objectives.

The ENSREG stress test takes for granted that all the structures, systems and components in the nuclear power plants assessed are in place and without fault, but the operational experience shows this not to be the case.

Power uprate

Power uprating, often combined with a lifetime extension of the plant, is an option to increase the plant’s profitability by increasing power generation. To uprate the electrical power output, there are two possibilities:

- optimising the turbines to increase the thermal efficiency of the plant. This has little effect on the safety level of the plant.

- increasing the reactor’s thermal power, usually by increasing coolant temperature. This produces more steam and enables the reactor to produce more electricity via the turbines. Increasing thermal power implies a higher number of nuclear fissions and more fission products aggravate accident situations, firstly by accelerating the accident sequence leading to a decrease of intervention time during accidents, secondly due to the considerably higher potential release of radioactive material.

Containment stability

Air craft crash is not obligatory to be evaluated in the stress test, and indeed several plants have reactor buildings that are insufficiently robust to protect the reactor system from the crash of a light aircraft (e.g. NPP Doel 1/2, Almaraz). Some reactor buildings’ roofs would not even withstand a high snow load (NPP Dukovany and Ringhals).

Many operators do not consider the risk of an airplane crash because of the low likelihood of such an event. The scenario should not be excluded, however. An airplane crash must be considered as a relevant safety issue, because such a crash could result in major disaster.

Spent fuel storage pools (SFP)

Spent fuel pools pose a hazard, if they are outside the containment. In some types of boiling water reactors the SFP is in the reactor building, but outside and above the containment (NPP Gundremmingen and Muehleberg).

Some NPPs have spent fuel storage pools in a special fuel building (NPP Krško and Gösgen). At VVER 440/V213 reactors the SFP is located outside the containment barrier in the reactor hall.

Severe accident management

An important measure for the prevention of severe accidents is filtered venting of the containment vessel. Such a system can cope with pressure due to release of steam and water from the primary cooling system. Pressure reduction to secure containment function is achieved by a controlled release via the filters. Many NPPs have filtered venting systems, but not all are seismically qualified (e.g. French PWR fleet). Thus in case of a pipe break caused by an earthquake it is not guaranteed that the venting system will be operable. Some NPPs do not have filtered venting systems at all (e.g. NPP Doel and Temelín).

To cope with severe accident habitability of the control room has to be guaranteed. Several plants have an alternative emergency control room. Some of these are bunkered in a robust separate building (NPP Muehleberg). Some plants do not have an emergency control room available.

Prevention of hydrogen explosions and fire is also important to mitigate severe accident consequences. Hydrogen explosions can cause damage of concrete buildings, as it happened in Fukushima.

The stress test assesses accident sequences only under ideal preconditions. The Czech National Report makes this explicit by describing a sequence in the following way: The station black out scenario is examined under the assumption, that all other safety systems are working and no other event occurs. All systems in the power plant, besides those systems that caused the loss of power supply for own consumption, continue to work correctly: “No design accident or failure was registered immediately before or after the SBO, in particular the following are excluded: seismicity, fire, floods.”

Some old reactors have no effective severe accident management measures to prevent or mitigate a severe accident such a total failure of cooling the reactor core or the spent fuel pool. The last resort in such cases is the fire brigade with its mobile equipment. It is a desperate measure to have the fire brigade to fight a severe nuclear accident.

STRESS TEST PEER REVIEW

The Peer Review introduced a new additional layer to the standard practice of safety reports prepared by the permitting body on national level, the Nuclear Safety Authorities. This exercise was designed to compare the results, teams consisting of several members mostly from the National Safety Authorities assessed whether the national stress test was conducted sufficiently.

The stress tests´ goal was to assess whether individual NPP are robust enough to withstand accidents resulting from natural hazards.

The national reports presented the design bases for earthquake and flooding and the safety margins. Concerning natural hazards significant differences exist in national approaches; difficulties were encountered with beyond design margins and cliff-edge effects assessment.

Since the evaluation of seismic and flooding margins was inconsistent, no general conclusions could be derived. It was a shared view among the reviewers that seismic margins exist. But this is not based on intensive investigations, but mostly on engineering judgment.

Loss of electrical power and loss of ultimate heat sink were treated as accidents leading to cliff-edge effects due to various combinations of AC/DC power losses and/or cooling water. Some national reports explained extensively the methodology for determining the cliff-edge effects. For these accident scenarios safety margin can be expressed as the time span available before lack of safety functions causes overheating of the core and core melt begins. For the most severe total losses of cooling with no recovery actions credited the time to fuel heat up typically ranged from 1 to 10 hours. Numerous improvements related to hardware and procedures have been identified to extend the time for recovery actions beyond 72 hours. ENSREG had not required action times beyond 72 hours. But even this time could prove being insufficient, if the site and the traffic routes are damaged due to a beyond design earthquake or flood.

Preventive aspects of severe accident measures are better developed than aspects of mitigation. The Peer Review Board points out that the operators need to ensure the provisions required for maintaining containment integrity are in place.

The ENSREG Peer Review Report is a document of very general character which confirms at large the findings of the operators. However, “the peer review concludes that all countries have taken significant steps to improve the safety of their plants” (ENSREG 2012). These measures summarized:

Measures to increase the robustness of plants have been decided or are considered.

Provisions of additional mobile equipment to prevent or mitigate severe accidents, such as bunkered equipment to manage severe accidents including instrumentation and communication means, emergency response centres protected against natural hazards and contamination

Implementation of measures to protect containment integrity, such as depressurization of primary circuit, measures to prevent high pressure core melt, prevention of hydrogen explosions, and prevention of containment overpressure.

ENSREG PEER REVIEW BOARD RECOMMENDATIONS (ENSREG 2012):

- Develop guidance on natural hazards assessments, including earthquake, flooding and extreme weather conditions, as well as corresponding guidance on the assessment of margins beyond the design basis and cliff-edge effects.
- Underline the importance of periodic safety review. In particular, ENSREG should highlight the necessity to re-evaluate natural hazards and relevant plant provisions as often as appropriate but at least every 10 years.
- Urgent implementation of the recognised measures to protect containment integrity is a finding of the peer review that national regulators should consider.
- Necessary implementation of measures allowing prevention of accidents and limitation of their consequences in case of extreme natural hazards is a finding of the peer review that regulators should consider.

Action plans for further analysis and subsequent implementation of the improvement measures have already been, or will be shortly defined, in all countries. The general aim is to implement improvements as soon as possible; ENSREG will identify an approach to have this large volume of work monitored and to establish the mechanisms for reporting on the implementation and for further experience sharing. Such reporting could, for example, be performed as part of the reports which have to be produced by Member States in the frame of the European safety directive.

The public of course is worried about how catastrophes at plants in their neighbourhood would be managed and demands a European initiative on off-site emergency preparedness. This subject was not part of the peer review mandate. However, the Peer Review Board recognised the importance of off-site emergency preparedness. On May 8, 2012 at the second public meeting the Board informed, that it had commissioned a study to identify the institutions which are responsible for emergency planning in the countries of the European Union.

Concerning the “security track” no information was provided at this second stakeholder meeting, nor any report released.

Conclusions

The EU Stress test is not a safety assessment of the European nuclear power plants. It is a limited analysis of the vulnerability of the NPPs concerning natural hazards. The accident scenarios are focused on external events, the quality of the SSC and its degradation in the oldest nuclear power plants in Europe are not subject of the analysis. The peer review team did not consider all safety issues that could trigger or aggravate an accident situation (e.g. ageing, use of MOX fuel, safety culture).

The design basis of the plants concerning natural events are not consistent, therefore the safety margins can only be assessed by engineering judgment. For extreme weather events the design basis and the robustness evaluation were done only superficial. In December 2011, the IAEA has published a new guide for extreme weather hazards[2]. We recommend that all plants make an assessment of weather hazards according to the new IAEA guide.

Severe accident management, especially regarding spent fuel pools and multi-unit accidents like at Fukushima, is in all countries an issue, but the development and deployment of SAM guides,

[2] Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations - IAEA Safety Standards Series No. SSG-18; December 01, 2011.

measures, equipment as well as organization and training of personnel is in very different state in the countries. Only one country (Slovenia) has a simulator for severe accident management.

The peer review team has not assessed the current safety level of the European NPPs, but only the potential increased safety level which should have been achieved in the next decade. Currently, there are several known shortcomings regarding the protection against earthquake, flooding and extreme weather hazards as well as the known impossibility to cope with a severe accident especially in event of earthquake or flooding. The reviewers only described the weaknesses, but they do not present an overall assessment of all facts. This is necessary for politicians to be able to assess the risk.

The EU stress test has no direct effect on the European nuclear power plant fleet. Even the oldest plant which have obvious deficiencies in defence of depth (Muehleberg, Doel, Rivne and other old WWER 440 reactors), apply for life time extension, and ENSREG has no mandate to stop this process.

1 INTRODUCTION

In reaction to the devastating nuclear disaster in Japan, the European Council concluded in March 2011, that “the safety of all EU nuclear plants should be reviewed, on the basis of a comprehensive and transparent risk and safety assessment ("stress tests"). Public perception has it, that the EU stress tests would be a new and thorough safety assessment. This report examines this assumption in contrast to what the stress test will deliver.

The nuclear catastrophe at NPP Fukushima Daiichi in March 2011 showed the world that the nuclear industry cannot prevent severe accidents from happening. The initiating event of the Fukushima accident was a tsunami caused by an earthquake; complete loss of electricity supply and cooling was caused by the operator TEPCO and the nuclear safety regulatory NISA resulting from having ignored design deficiencies at the four older units of the Fukushima NPP in Japan. Emergency diesel generators installed almost as low as sea level were flooded by the tsunami about one hour after the earthquake and lost immediately. This was the "cliff edge” effect which resulted in a disaster comparable with the Chernobyl accident of 1986 in the Ukraine.

The accident in Japan proved that highly unlikely incidents cannot be excluded. Contrary to accepted practice Probabilistic Safety Assessments (PSA) do not constitute a sufficient basis to declare a plant operation safe. Safety of NPPs needs to be backed by deterministic assessments, which excludes initiating events and accident scenarios only if they are proven to be physically impossible (see chapter 3 WENRA Safety Objectives).

The Fukushima accident confirmed the mistrust towards nuclear power among the Japanese but also European citizens. The European Commission welcome the Austrian suggestion to conduct stress tests at all EU NPP. They were introduced to improve confidence in the safety of the European NPPs. In particular consequences of earthquakes and floods are to be assessed. These stress tests are only a very limited safety assessment.

The stress test assessed whether the nuclear power plant can withstand the effects of the following events:

- Natural disasters: earthquakes, flooding, extreme cold, extreme heat, snow, ice, storms, tornados, heavy rain and other extreme natural conditions. According to [ENSREG 2011] the analysis of impacts of natural disasters mainly earthquake and flooding is obligatory.
- Analysis of impacts of human induced events is not obligatory, but should be considered as indirect initiating events (e.g. damage of power line due to airplane crash)

The target of the stress test [ENSREG 2011] is the analysis of the plant robustness to cope with consequences of loss of power including station black out, loss of ultimate heat sink, and a combination of both; safety reserves (margins) should be assessed; only some progress reports give concrete indications of these reserves, but more often the formulation “sufficient” is used.

Best possible outcome of the stress test could be an analysis of multiple faults and common mode failures at the NPP sites. “Cliff edge” effects which could result in core melt are to be investigated and improvements for water and power supply in emergency situations could be proposed. ENSREG has no regulatory mandate. However, the implementation of safety improvements can be ordered by the national regulatory authorities, who are members of ENSREG.

The lesson from the Fukushima NPP accident is clear: Take the unthinkable into account and develop adequate emergency measures for the protection of the population in the densely populated regions of Europe. External impacts and containment stability are important safety issues for all NPPs. These issues are important because an accident with a containment failure (or a bypass) leads to radioactive emissions in the atmosphere – such emissions can affect not only the immediate vicinity of the reactor, but also regions some hundreds of kilometres away. Therefore the

Peer Review should focus on measures to prevent such emissions, since in Europe large population is living in regions with nuclear reactors.

Even if the likelihood for such an accident is small, it cannot be excluded. The stress tests is to evaluate the "unthinkable" and find out which improvements, prevention and mitigation measures need to be undertaken. The exclusion of unlikely accidents is not justified (for example the 95th percentile) from the analysis.

The first ENSREG public stress test meeting in January 2012 saw stakeholders not only discussing safety but also pointing to the fact, that emergency plans and measures outside the NPP sites are insufficient in case of severe accidents. This is an issue of high interest for the European population.

Greenpeace commissioned our team, Antonia Wenisch, Oda Becker and Patricia Lorenz, to prepare a critical review of the ENSREG stress test.

This report is a review of selected NPPs based on the National Stress Test Reports. Chapter 2 and 3 give an overview on general safety issues concerning the stress test. Chapter 4 is the review of selected NPPs. It lists the weakness of each individual reactor and the measures proposed by the national regulator in its report. The most important shortcomings the National Stress Test Report does not mention are listed and discussed at the end of each chapter on the individual reactor: external impacts, containment stability, ageing effects and outdated plant design.

The majority of the European reactor fleet is old, the design basis stemming from decades ago. Not all operators have reassessed the site hazards in compliance with state-of- the-art methodologies. An overview of some of the National Reports shows that the chosen indicators for the site hazards as earthquake and flood are very different: the Design Base Earthquake (safe shutdown – seismic level 2) is determined from the maximum intensity of the quake with a recurrence period between 500 years (France) to 100.000 years (Germany).

Greenpeace selected 12 NPPs. In chapter 4 the "Critical Stress Test Review" examines those NPPs and presents examples of the shortcomings we consider to be the most important issue for the individual plant or e.g. significantly not complying with the Stress Test Criteria; those evaluations do not claim to be a complete assessment but a contribution to full picture of the stress tests.

Almaraz (Spain)
Cattenom (France)
Doel (Belgium)
Tihange (Belgium)
Fessenheim (France)
Gundremmingen (Germany)
Gravelines (France)
Krško (Slovenia)
Mochovce (Slovak Republic)
Temelín (Czech Republic)
Wylfa (United Kingdom)
Ringhals (Sweden)
Muehleberg (Switzerland)

2 THE EU STRESS TEST

2.1 CONTENT OF THE EU STRESS TEST

In reaction to the devastating nuclear disaster in Fukushima, the European Council concluded in March 2011, that “the safety of all EU nuclear plants should be reviewed, on the basis of a comprehensive and transparent risk and safety assessment ("stress tests").

ENSREG published the scope and modalities for comprehensive risk and safety assessments of EU nuclear power plants on 13th May 2011. This “Declaration of ENSREG” determines the concept, methodology and time schedule of the EU stress test. The following principles are crucial:

“Stress test” is defined as a targeted reassessment of the safety margins of nuclear power plants in the light of the events which occurred at Fukushima: extreme natural hazards challenging the plant safety functions and leading to a severe accident.

“*Cliff edge”* is defined as a step change in the event sequence. Examples are the exhaustion of the capacity of the batteries in the event of a station black out or exceeding a point where significant flooding of the plant area starts after water overtopping a protection dike.

Meanwhile ENSREG moved toward a less ambitious definition of the stress test goals, talking about „“stress test” as a targeted reassessment of the safety margins of nuclear power plants.” Terrorist attacks e.g. sabotage are not part of the stress tests, but are to be dealt with by a especially established Council working group - Ad Hoc Group on Nuclear Security (AHGNS).

The first phase started in June, when the operators started to prepare the self-evaluation of their plants. This was followed by the second phase, when the national authorities reviewed the progress reports submitted by the operators. All reports were handed over to the EU Commission by December 31 and the last part started: The peer review, which conducts the review three topical reviews in parallel:

–Initiating Events

–Loss of Safety Functions

–Severe Accident Management

In March 2012 six country review teams continued the review in parallel and also country visits were part of this phase, 1 plant per country was visited by the EU nuclear experts. The ENSREG draft report will consist of the ENSREG Report on the three topics and 17 individual Country Reports. The ENSREG report is called draft report, because the report will be final only after the EU Commission will have submitted it to the EU Summit on 27/28 June 2012.

EU Stress Tests Timetable for 2012:

April 25: ENSREG delivers draft report

May 8: EU Stress Tests Stakeholder Meeting, Brussels

June 27/28: EU Council to adopt Final EU Stress Tests Report

However, it seems already clear, that this will not be the final say and some further steps will be taken in the context of EU Stress Tests, e.g. opening the third track, emergency preparedness of NPP. The results of the stress test, the safety margins assessed, might become part or be reflected in the EU nuclear safety directive *2011*/70/EURATOM, which is currently being updated in reaction to Fukushima. The report will consist of the ENSREG Report on the three topics and 17 individual Country Reports. The ENSREG report is called draft report, because the report will be final only after the EU Commission will have submitted it to the EU Summit on 27/28 June 2012.

2.1.1 Natural hazards

2.1.1.1 Earthquake

Design basis of many European NPPs was determined many decades ago. Not all operators have in the meantime reassessed the site hazards in compliance with state-of-the-art-methodologies.

For the design basis earthquake (DBE) two levels of ground motion hazard should be evaluated for each plant sited. Both hazard levels generate a number of design basis earthquakes grouped into two series, seismic level 1 (SL-1) and seismic level 2 (SL-2), according to the target probability levels defined for the plant design. [IAEA 2003]

In the plant design SL-2 is associated with the most stringent safety requirements, while SL-1 corresponds to a less severe, more probable earthquake level that normally has no severe safety implications. For low levels of seismic hazard, SL-1 is usually related to operational requirements only. Safety classified items should be designed with reference to either SL-1 or SL-2 according to their safety function (usually associated with SL-2) and to operational requirements (usually associated with SL-1). SL-2 is often denoted as the safe shutdown earthquake. [IAEA 2003]

In many countries seismic threat was not considered in the early NPP design such as Magnox and other old reactors. In UK inclusion of design against earthquake for nuclear facilities did not become commonplace before the early 1980ies. [ONR 2011]

Safe shutdown (SL 2) in France was determined from the maximum intensity of the quake with a recurrence period of 500 years. In Germany the design is also based on the maximum intensity but with a recurrence period of 100,000 years.

<u>In highly active areas,</u> where both earthquake data and geological data consistently reveal short earthquake recurrence intervals, periods in the order of tens of thousands of years may be appropriate for the assessment of capable faults. [IAEA 2010]

<u>In less active areas</u>, it is likely that much longer periods may be required. A structural relationship with a known capable fault has been demonstrated such that movement of the one may cause movement of the other at or near the surface. [IAEA 2010]

Besides the recurrence period also the region around the NPP in which the earthquake focuses are considered – differ in the dimension. The size of relevant region may vary depending on the geological and tectonic setting. The radial extent is typically 300 km [IAEA 2010].

These differences hardly allow for a comparison of design bases applied in different countries. The safety margins assessment is based on the design and thus it is not possible to compare the results.

To determine the seismic hazard of a nuclear power plants several different methodologies are used such as geological and geotechnical investigations, hydrogeological investigations, creation of a database using prehistoric and historical as well as instrumental earthquake data.

Uncertainties that cannot be reduced by means of site investigations do not permit hazard values to decrease below certain threshold values. For this reason a minimum earthquake level should be recognized as the lower limit to any seismic hazard study performed for a NPP. In that regard, generically, this level should be represented by a horizontal free field standardized response spectrum anchored to a PGA value of 0.1g. [IAEA 2010]

According to IAEA, any major effects to be expected from an earthquake would be related to the vibrations induced in the systems structure and components. Vibrations can affect the plant safety resulting in release of hazardous substances, fire or flooding. Deficiencies in physical separation or fire protection in the design of old plants increase the hazard of failures which accelerate the accident progress.

Only some National Stress Test Reports discuss indirect impacts of earthquakes such as impacts due to damage of non-seismically qualified buildings, fires or flooding of corridors due to pipe-breaks. Station Blackout (SBO) cannot be excluded even if the electricity supply has a high redundancy but the switchyard (cables, connections or the switches) are not seismically qualified.

2.1.1.2 Flooding

The cooling of nuclear power plants calls for a location at the sea or at a large river. Already in 2003 an IAEA Safety Guide was published that provides recommendations relating to the evaluation of the flood hazard for a nuclear power plant [IAEA 2003a].

According to this IAEA Safety Guide, the region shall be assessed to determine the potential for flooding due to natural causes such as precipitation, snow melt, high tide, storm surge, seiche and wind waves. The possible combinations of the effects of several causes shall be examined. The potential for instability of the coastal area or river channel due to erosion or sedimentation shall be investigated. Information relating to upstream water control structures shall be analysed to determine whether the NPP would be able to withstand the effects resulting from the failure of one or more of the upstream structures. The expected main effects of flooding are as follows [IAEA 2003a]:

- The presence of water in many areas of the plant may be a common cause failure for safety related systems, such as the emergency power supply systems or the electric switchyard, with the associated possibility of losing the external connection to the electrical power grid, the decay heat removal system and other vital systems.
- Considerable damage can also be caused to safety related structures, systems and components (SSC) by the infiltration of water into internal areas of the plant, by rise of the water table induced by high flood levels. Water pressure on walls and foundations may challenge their structural capacity. Deficiencies in the site drainage systems may also cause flooding of the plant.
- The dynamic effect of the water can be damaging to the structure and the foundations of the plant as well as the many systems and components located outside the plant.
- A flood may transport ice floes in very cold weather or debris of all types which may physically damage structures, obstruct water intakes or damage the water drainage system.
- Flooding may also affect the communication and transport networks around the plant site. The effects may jeopardize the implementation of safety related measures by operators and the emergency planning by making escape routes impassable and isolating the plant site in a possible emergency.
- Flooding can also contribute to the dispersion of radioactive material to the environment in an accident.

In the last few decades the threat posed by flooding has been increased at many nuclear power plant sites. The reason for this is both a change in the situation (e.g. Climate change, construction of dams, reduction of natural flood plains) and a change in the assessment of the threat. The observation of trends is essential to ensure an appropriate assessing of the flooding risk.

All in all, it has to expect that the hazards associated with flooding will increase for many nuclear power plants. It is highly questionable whether operators and regulatory authorities are fully aware of this problem.

Flooding events occurred in nuclear power plants have shown that water has damaged safety equipment located below the level of the site, because the water resistance of doors was miscalculated or seals of cable penetration were corroded.

2.1.1.3 Extreme weather

According to the Intergovernmental Panel on Climate Change (IPCC), the type, frequency and intensity of extreme weather events are expected to change as Earth's climate changes. These changes could occur even with relatively small mean climate changes. Changes in some types of extreme events have already been observed, for example, increases in the frequency and intensity of heat waves and heavy precipitation. Precipitation extremes are projected to increase more than the mean values. The frequency of extreme precipitation events is projected to increase almost everywhere [IPCC 2007].

Many of the design standards of NPP were based on an understanding of a climate system that is now 40 years out of date. Today, it is known that climate change is making floods, droughts, and hurricanes stronger and more frequent. This means the safety standards, even when followed perfectly, are probably not sufficient to prevent disaster. Large and destructive floods once thought likely to happen only once in 100 years on average are now expected to happen every 20 years.

Sometimes, what is being thought to be a "worst case" scenario is not really the worst case. Just because there is uncertainty about how climate and weather will affect the reactors does not mean to ignore the issue. Quite the opposite; it would be negligent to ignore this uncertainty [POOL 2011].

2.2 Shortcomings of the EU "Stress test"

One important lesson of the Fukushima accident is that nuclear accidents really occur – even in developed industrialised countries. The operation of nuclear power plants is always without any exemption connected with the residual risk of an uncontrolled nuclear accident.[3] Nuclear safety in the absolute sense does not exist. The expression "a nuclear plant is safe" only means that the residual risk is accepted. The cause of a severe accident, like the earthquake and the tsunami in Japan, could be the combination of a fire incident, human error, leaking pipes and the clogging of the cooling circuit or any other hazard combinations in any nuclear power plant around the world. Combinations of errors – technical and human – cannot be assessed and excluded in advance. Therefore it is a common understanding to believe that any test could make nuclear power plants safe. However, a sound safety assessment can help to reduce the nuclear risks [RENNEBERG 2011].

The European Council (March 25th) declared that "the safety of all EU nuclear plants should be reviewed, on the basis of a comprehensive and transparent risk assessment ('Stress test')". The "Stress test" should serve as a basis for the decision even to shut down unsafe plants [OETTINGER 2011]. But the original objective of the European Council was significant reduced by European Nuclear Safety Regulators Group (ENSREG): "For now we define a 'Stress test' as a targeted reassessment of the safety margins of nuclear power plants in the light of the events which occurred at Fukushima: extreme natural events challenging the plant safety functions and leading to a severe accident." [ENSREG 2011]

A study commissioned by the Greens in the European Parliament, analysed the ENSREG "Stress test" specifications for nuclear power plants in Europe [RENNEBERG 2011.] According to this study the idea of a comprehensive risk assessment of the European power plants was not the intention

3 The core melt probability of nuclear power plants is about 1: 100,000 per year and plant. This means a core melt probability in any plant around the world of about 1: 230 per year and a core melt probability of about 1:2500 per plant within an operation time of 40 years.

any more, the Stress test does not provide a method for comparing the safety of the different plants, nor does it answer how safe European plants actually are. The assessment of safety margins is something else. Safety margins describe those safety provisions of the plant that enable the operator to cool down the reactor and to prevent radioactive emissions even when the existing safety systems have failed.

The EU “Stress test” has the following important shortcomings and limitations [RENNEBERG 2011]:

- The scenarios that are under review are incomplete. In experience of the Fukushima accident specific configurations and failure modes typical for e.g., airplane crash or internal fire, human failure including combinations of those events that until now not have been under consideration are not covered by the “Stress test”.
- Particularly an air plane crash is not considered in the frame of the “Stress test”. ENSREG regards this scenario under the terms of security and therefore claims not to be competent to include it in the Stress test.4 This is an evidently misleading argument. The fact that air planes might crash on a nuclear power is completely independent of its cause and it might therefore happen without any terrorist background. An airplane crash has to be considered as a relevant safety issue.
- Obviously it is always better to prevent accidents from happening than to deal with the consequences of an occurred accident. But safety features of the plant that are needed to prevent an accident are not under review.
- The safety management of nuclear power plants, which is of acutely importance, is not included in the “Stress test”. Even not foreseen is a report whether a safety management corresponding to the state of the art is established and functioning.
- Another important area of special attention regarding the safety of nuclear power plants the quality of personal training, is not included in the “Stress test”.
- Ageing effects and out-dated safety qualification of structures, systems and components in old NPPs are not considered in the national stress test reports.
- The “Stress test” specifications require descriptions of the plants properties but requirements on the quality and the comprehensiveness of the descriptions are not defined. So far as the specifications rely on the licensed design and its safety case, it relies on aged criteria and methods. This is one of the reasons why the results of the country reports on the assessment of the European nuclear plants are not really being comparable.
- ENSREG requires a classification of the documents: “Documents referenced by the licensee shall be characterized either as: 1) validated in the licensing process, 2) not validated in the licensing process but gone through licensee's quality assurance program or 3) no one of the above.” [ENSREG 2011]
  It is to be expected that a lot of safety relevant documents will be classified in the second or third category. This will significantly weaken the confidence in the report of the operator. The “Stress test” specifications accept engineering judgements whenever there is no time for founded assessments. However, the judgement of an engineer depends on many factors, for example on his experience and on his subjective perception of the acceptance of risks. It is completely out of scope for a nuclear authority to check engineering judgements of the operator’s report within one month. Considering usual practice it would take several years to come to a founded judgement.

4 “Risks due to security threats are not part of the mandate of ENSREG and the prevention and response to incidents due to malevolent or terrorists acts (including aircraft crashes) involve different competent authorities, hence it is proposed that the Council establishes a specific working group composed of Member States and associating the European Commission, within their respective competences, to deal with that issues.”[ENSREG 2011]

- There is no definition in the “Stress test” specifications, what additional level of safety should be achieved otherwise the plans should be back-fitted or should be shut down. Any criteria that defined the robustness of a plant are missing. The German Reactor Safety Commission for example has defined four levels of robustness in the frame of the German stress test. The basic level is chosen as a level that must be fulfilled by all operating plants, taking into account that all plants meet the licensing conditions and have realised all back-fitting measures required by the authority. Each of the three levels of robustness defines a specific larger kind of safety-margin.
- The operators´ reports are the most important basis for the final national report and the assessment of the safety of the plant. For obvious reasons the operators cannot be considered independent: it is in their interest to demonstrate that a plant does not require costly back-fitting measures.
- Almost none of the experts involved in the “Stress test” are really “independent”. The EU Commission does not have any technical competence which is able to assess the safety of nuclear power plants. Therefore the ENSREG was created to give especially technical guidance. Beside the members of those countries without nuclear power plants the ENSREG consist mostly of the chairpersons of the nuclear authorities of the countries operating nuclear power plants. In the past those experts and their technical support organisations legitimated the operation of the power plants under their supervision and they informed the public that the plants were operating safely. Conducting “Stress test” makes them review their own practice and their own statements about safety and about acceptable risks.
- To enhance credibility of the “Stress test” process the national reports will be subjected to a peer review process. The complexity of data, of calculation methods, of assumptions about the safety parameters and their interdependence within the system of a nuclear power plant is extremely high. Regarding the short time frame (about three month), the immense workload and the limited number of experienced experts able to review the assessments, it is not possible to perform a well-founded peer review process. The peer reviews teams consist of the experts of the involved member countries; to criticise colleagues within an official process whose results shall be open to the public is difficult. Realistically it is necessary to expect that this peer review process will identify only evident deficiencies.

CONCLUSION

The EU “Stress test“ cannot assess the safety of the European nuclear power plants and therefore cannot serve as basis to decide which nuclear power plants need to be shut down. Stress tests do not provide information about the reliability of plant safety measures to prevent postulated failures of the safety systems or about all those other scenarios and serious events that could lead to sever accident. Considering the limited scope of the “Stress test”, the missing of clear assessment criteria, and taking into account the interests of the involved experts, reports might serve mainly to demonstrate to the public that the plants are operating safely.

Nevertheless the „Stress test” could serve as a first assessment of the plants´ capability to withstand several external events. One important new feature which introduced now is the evaluation of the so called “cliff edge effects”.[5] Of high importance will be to assess the time until critical situations arise when cooling is insufficient. The test may result in technical and organisational recommendations to be better prepared in the case of such accidents. The “Stress test” could be a first step towards a harmonised risk assessment of the European nuclear power plants. However, there are no indications that these opportunities will be made use of.

5 A cliff edge effect describes a qualitative degradation of the plant’s safety conditions (comparable to walking on a cliff and the next step fall down).

The 143 operating European nuclear power plants (NPPs) differ in age and design. None of the NPPs complies with requirements of an accident preventing defence-in-depth concept that corresponds to the state-of-the-art-requirements. The differences concerning the realisation of the defence-in-depth correspond to substantial differences of their residual risks. The EU “Stress test” should be complemented by a second part that assesses the capability of the nuclear power plants to prevent accidents. This means the assessment of the implemented defence-in-depth concept[6] against the current state-of-the-art. The safety objectives of WENRA for new reactors are an appropriate basis for this assessment (see chapter 3). In the second part of the EU “Stress test” more precise requirements for the underlying data should be defined.

2.2.1 Important safety issues of old NPPs

2.2.1.1 Ageing effects

Most operating NPPs in Europe are old. The ENSREG stress test does not consider that a NPP after 30 years of operation is not the same as it was originally.

30 years of operation leave marks on an NPP, the most important being the ageing phenomena. The hazards resulting from ageing components and systems set in approximately 20 years after plant operation and further increased by lifetime extension. Ageing renders NPPs more incident-prone.

In the past years a very small number of new reactors are under construction. Prolonging the lifetime of the operating NPPs to 40 or more years is very attractive for the operators from an economic point of view. However, significant effort is required to modernize these plants and to implement safety improvements to achieve current EU safety objectives. However, a large amount of reconstruction measures increases the enormous complexity of the systems of an NPP.

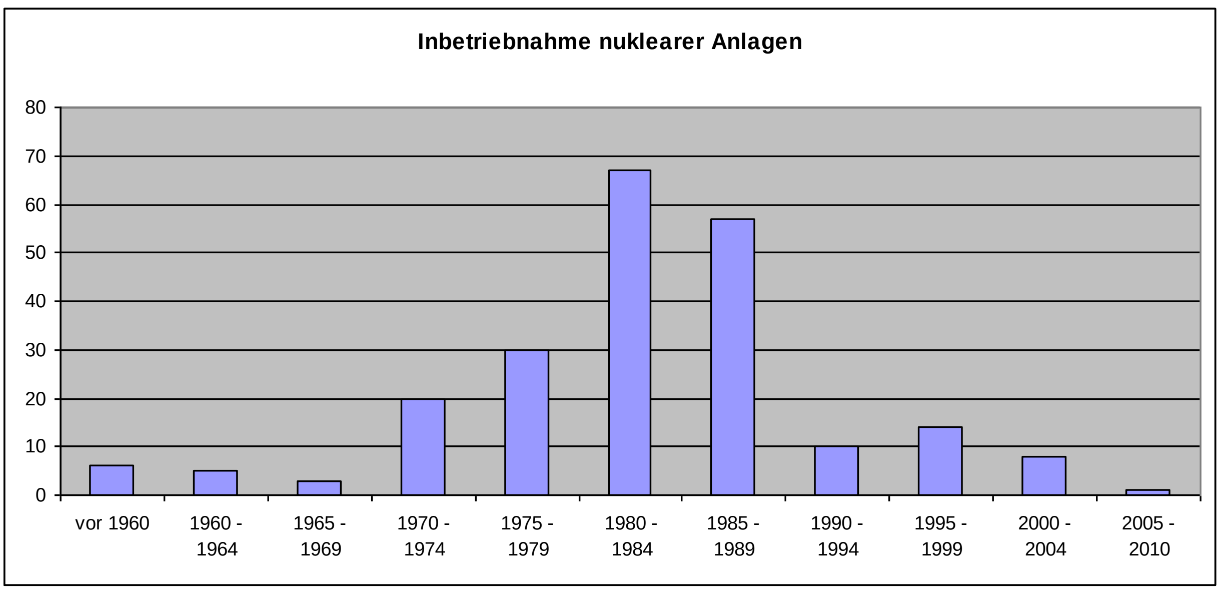

Figure 1: *Time of commissioning the NPPs*

6 The first level of defence shall provide a safe operation within the defined operational data specifications. The second level of defence serves for those cases when the operational specification data are exceeded. In those cases systems are needed to lead the reactor back into the allowed range of operational limits. If this second level fails and the reactor can get out of control there is the most important third level of defence. This third level of defence consists of safety systems that must be able to shut down the reactor and to cool the fuel.

Ageing can occur in many different forms in different components. The most significant ageing phenomena in VVER (as in all PWRs) concern the reactor pressure vessel (RPV) [HIRSCH 2005]:

Materials close to the core:

Embrittlement (reduced toughness, shift from the ductile to the brittle-transition temperature) results from neutron irradiation. This effect is particularly relevant if impurities are present. Copper and phosphorus favor embrittlement, as well as nickel at very high neutron fluences as encountered at VVER reactor vessels. Neutron embrittlement is mostly relevant for PWRs.

Welds: Crack growth occurs because of changing thermal and mechanical loads. For PWRs, this occurs mostly in embrittled welds close to the core;

Vessel head penetrations: Crack formation and growth due to corrosion mechanisms;

Inner edge of nozzles: Strong concentration of stresses because of varying wall thickness, with changing thermal and mechanical loads as well as corrosion and erosion effects. This leads to the hazard of crack formation or growth of cracks. Inspection is complicated because of geometric lay-out and high-wall thickness.

Pipelines: cracks in pipelines develop due to mechanical and thermal loads, erosion and corrosion. Thinning and material fatigue due to resonance vibrations, water hammer and so on are very difficult to keep under surveillance. For these reasons, damages become more likely with ageing materials.

Main Coolant Pumps: Crack formation and crack growth can occur due to thermal- and high-frequency fatigue processes, supported by corrosive influences. Inspections are difficult.

A further important issue is the ageing of the electrical devices and cables isolation materials that can cause simultaneous both a dangerous fire and the impossibility to activate and/or control safety systems.

A combination of a design base accident and the failure of an essential component can result in a severe accident scenario. In particular, in old plants multiple failures of structures or components cannot be excluded for example in case of an earthquake.

CONCLUSION

The quality of the safety related systems and components of the plants like the material of pipes, of the reactor vessel, of valves and pumps, of control and instrumentation equipment is not under investigation in the ENSREG stress test. Degradation effects, in particular those caused by the ageing of plants and material fatigue, are not taken into account in the stress test.

The safety design of nuclear power plants is very important to prevent as well as to deal with incidents or accidents, but is not part of the stress test. The safety designs of all operating plants are out-dated and show deficiencies. Therefore a risk assessment of a nuclear power plant has to consider the design base including the operational experience of all other comparable plants.

The “Stress test” relies on the safety cases of the licenses, which in many cases was issued several decades ago. Meanwhile many parameters of the plants have been changed, former assumptions have been revised, former calculations methods may be out dated, the knowledge about materials and systems has developed and years of experience with formerly unforeseen scenarios was made during operation.

The ENSREG stress test take if granted that all of the considered structures, systems and components of the nuclear power plant are in place and without deficiencies, but the operational experience of nuclear power plants show this is not the case.

2.2.1.2 Power Uprate

Power uprating, which is often combined with life time extension, is an option to increase the profitability of a NPP. To uprate the electrical power, there are two different possibilities:

- At constant reactor power, the thermal efficiency of the plant is increased (mostly achieved by optimising the turbines). The safety level of the plant remains nearly unaffected.
- Thermal power of the reactor is increased, usually by raising coolant temperature. Thus, more steam is produce and the reactor can produce more electricity via the turbines. An increase of thermal power implies more nuclear fissions (and so more fission products). Also, higher loads to the reactor materials are unavoidable. An increase of reactor power reduces safety margins and at the same time accelerates ageing processes.

A recent published IAEA Report highlighted that because changing the thermal power affects so many systems and analyses, there are numerous “opportunities” to overlook potential problems. Experience shows: Higher excitation/vibration of steam lines leads to accelerated wear of supporting structures and studs. Effects on electrical components may sometimes be neglected or overlooked because of lack of knowledge or incorrect assumptions (example: generator failure by local overheating in the stator or failure in insulation of transformers). Increased flow will have an impact on flow-induced vibration in the steam/feedwater path; non-linear effects might occur. Higher steam flows can also result in valves not performing as they did before the power uprate. Especially large power uprates have sometimes resulted in equipment degradation and damage in secondary piping systems and equipment. The US nuclear power industry has experienced over 60 events related to power uprates since 1997 [IAEA 2011].

CONCLUSION

It could not be completely excluded that effects related to power uprates initiate an accident. However, these effects would certainly aggravate an accident situation: first the accident sequence is accelerated, which leads to a decrease of intervention time; second the potential radioactive release is considerable higher.

3 COMPARING THE EU STRESS TEST WITH CURRENT SAFETY OBJECTIVES

The current EU stress test specifications do not require an assessment of the described existing safety systems and components against the state of the art. One reason for this limitation is the limited time frame. But it can be assumed that many, respectively all, of the nuclear power plants in the European Union do not comply with modern safety requirements. For this reason, it has been recommended by many experts that such a check will be done as a second part of a comprehensive risk assessment.

The "*Safety Objectives for New Power Reactors*" published by the reactor harmonization working group (RHWG) Western European Nuclear Regulator's Association (WENRA) can be seen as the state of the art [WENRA 2009]. These safety objectives, formulated in a qualitative manner to drive design enhancements for new plants, should be also "***used as a reference for identifying reasonably practicable safety improvements for 'deferred plants'[7] and existing plants in case of periodic safety reviews***". [WENRA 2010]

The reactor harmonization working group (RHWG) of WENRA is currently outlining more explicit positions implied by the new NPPs' safety objectives for some selected important topics. These positions are scheduled to be published by the end of 2012 [WENRA 2012].

The safety objectives (SO) 1 to 5 and 7 dealing directly with reactor safety are described and their relevance for increasing safety explained.

SO 1: Normal Operation, Abnormal Events and Prevention of Accidents

Obviously it is always better to prevent an accident than to deal with consequences of an accident. Therefore, the aim of this WENRA objective is to realize the prevention of an accident particularly by strengthening the defence-in-depth concept. **At defence-in-depth level 1** the reduction of frequencies of abnormal events has to be achieved. For example, this measure should be implemented to reduce the failure frequency of components by the use of adequate materials, by a comprehensive identification of ageing mechanisms and by design provisions for an effective in-service inspection and ageing monitoring.

Permanent operational stresses may lead to a deterioration of the specified material properties. Without appropriate countermeasures ageing and/or material fatigue can affect most plant components to a significant extent and induce failures. For instance, significant corrosion could induce a sudden rupture of a pipe which leads to an accident condition. A comprehensive ageing management system is necessary to counteract the ageing problems.

With in-service inspection in regular intervals occurring material irregularities can be detected prematurely, and the affected component or the part of the component can be repaired or changed. For an efficient in-service inspection it is necessary to prove that all safety relevant equipment is conditioned and arranged that in-service inspection for the identification of upcoming material irregularities can be done.

At defence-in-depth level 2 the reduction of the potential for escalation to accident situations by an efficient control of abnormal events has to be accomplished. Therefore, the man-machine interface (e.g. via an ergonomic design of the control room) should be improved regarding information and diagnostics provided to operators. The improvement of the man-machine interface can avoid operating errors.

7 NPPs originally based on reactor design similar to currently operating plants, the construction of which halted at some point in the past, and now being completed with more modern technology.

SO 2: Accidents without core melt

This safety objective serves three targets: Very low off-site radiological impact of accidents without core melt (no iodine prophylaxis, no sheltering or evacuation), reduce core damage frequency as far as reasonably achievable and reduce the impact of external hazards and malevolent acts. **In the defence-in-depth concept these tools belong to level 3.**

To meet this objective, WENRA calls for consideration of a more systematic analysis of critical events and situations in all operating states (operation and shutdown) and not only for the reactor, but also for the spent fuel pool and other facilities of the plant.

That means:

To check whether all events (internal and external) and particularly credible combinations of events are considered for the plant design according to the state-of-art. Some very important requirements in this context are the assumption of long standing external impacts and of the combination of multiple natural or other external impacts as well as the combination of external impacts with internal events.

The implementation of the different levels of the defence-in-depth concept was initially orientated to postulated incidents and accidents occurring under full power conditions. The search for initiating events during shutdown was less methodical. Most current PSA show that the contribution of core damage frequency (CDF) for the shutdown state is in the same order of magnitude as that for operation. Therefore, a systematic consideration of the shutdown state could improve defence-in-depth.

For existing reactors the control of accidents is mainly focused on the reactor core. However, the scope of the defence-in-depth has to cover all risks induced by the nuclear fuel, even when the fuel is stored in the spent fuel pool. The accident in Fukushima highlights this deficit of older reactor types.

Another area for improvement highlighted by WENRA to meet this safety objective is the reduction of human-induced failures particularly through more automatic or passive safety systems and longer “grace period” for operators. Human errors bear a potential for jeopardizing defence-in-depth. They have a considerable potential to trigger common cause failures (meaning they affect all redundancies of a specific safety system) as has been seen in some safety significant events, including the Chernobyl accident in 1986.

Accident conditions which are considered in the safety objectives for defence-in-depth level 3 are broader than those for existing reactors. Now they include multiple failure situations which were previously considered as “beyond design”.

The reliability of the safety systems is achieved with an adequate combination of redundancy and diversity. This means the same safety functions are available several times (redundancy) respectively the safety function is ensured by provisions with different physical or chemical mechanisms (diversity). Special attention has to be paid to minimising the possibilities of common cause failures. Therefore, physical and spatial separation shall be applied as far as possible. For example, the safety assessment of fire effects has to clearly identify common mode failure possibilities (including internal flooding risks linked to the use of fire fighting systems) which could result from incomplete separation of redundant equipment.

SO 3: Accidents with core melt

The most ambitious safety objective is to reduce potential radioactive releases to the environment from accidents with core melt. Accidents with core melt which would lead to early releases without enough time to implement off-site emergency measures or large releases which would require protective measures for the public that could not be limited in area or time have to be practically eliminated. Occurrence of certain severe accident conditions can be considered as practically eliminated “*if it is physically impossible for the conditions to occur or if the conditions can be* *considered with a high degree of confidence to be extremely unlikely to arise*”.[8]

Even though the probability of severe accidents with an early and/or large release for existing plants is estimated to be very small, the damage caused by these accidents is very large. Therefore, the risk of existing NPP for the public is relative high and has to be reduced urgently. Furthermore, the frequency of occurrence of severe accidents, calculated on the basis of the failure rates in all assessed event scenarios, is afflicted with high uncertainties. For new power plants it is envisaged to extend the design traditional beyond design basis in the area of core melt mitigation. Technical improvements which are highlighted by WENRA to meet this safety objective are mainly substantial design improvements of the containment. Another point is the use of improved materials for reactor pressure vessels to take the core melt phenomena into account.

If any accident scenario with core melt is practically eliminated by the operator, the justification should include demonstration that there is sufficient knowledge of the accident conditions and of the phenomena involved (e.g. hydrogen behaviour). Furthermore, uncertainties associated with the data and methods should be quantified.

SO 4: Independence of all Levels of Defence-in-Depth

In addition to the strengthening of each of the levels separately the aim of the fourth objective is an overall reinforcement of defence-in-depth concept by enhancing the effectiveness of independence between all levels. It is the general objective of defence-in-depth to ensure that a single failure, at one level and even a combination of failures, should not propagate to jeopardise defence-in-depth at subsequent levels. “*The independence of different levels of defence is a key element in meeting this objective*.” [IAEA 1996]

To evaluate the compliance of a plant with this safety objective all safety functions must be proved whether they have duties at two or more levels of defence-in-depth. Components of safety systems of level 3 should not fulfil an operational task.

SO 5: Safety and Security Interfaces

The aim of this safety objective is to ensure that safety measures and security measures are designed and implemented in an integrated manner. Only one area of improvement is highlighted by WENRA: Aircraft crash protection against large civil airplanes.

Although concern about malicious acts involving nuclear installations is not new, recent terrorist events – particularly the 9/11 attacks – have demonstrated that an attack on a nuclear facility might be attempted and that terrorists have formidable capabilities and dedication.

Many elements or actions serve to enhance both safety and security simultaneously. For example, the containment structure in a nuclear power plant serves to prevent a significant release of radioactive material to the environment in the event of an accident, while simultaneously providing a robust structure that protects the reactor from a terrorist attack. Nonetheless, there are also circumstances in which actions to serve one objective can be antagonistic to the achievement of the other. For example, the introduction of delay barriers for security reasons can limit rapid access to respond to a safety event [IAEA 2010a].

SO 7: Leadership and Management for Safety

This safety objective aims to ensure an effective management of safety from the design stage of a new reactor. This implies that the licensee establishes effective leadership and management of safety and has sufficient in house technical and financial resources to fulfil its prime responsibility in

8 This means that the practical elimination of a condition cannot be claimed solely based on a low probability of occurrence.

safety. Furthermore, the licensee has to ensure that all other organisations involved demonstrate awareness among the staff of the nuclear safety issues.

Practically all reactors in Europe are in operation since many years. For those reactors safety management is especially important regarding substantial back-fitting. The awareness of significance of safety management for the safe operation of NPPs has increased in recent years. This is mainly due to the occurrence of some reportable events which revealed substantial deficiencies in safety management [NITSCHKE 2004].

The safety management of the plants is of utmost importance for ensuring safety. A possible detriment of the safety shall be identified very early. A safety management system needs to prove that it state of the art is fulfilled and it is functioning. It should be based on a process-oriented approach and is to be considered as an integral part of the integrated management system.

4 REVIEW OF 13 SELECTED NPP

4.1 NPP Almaraz, Spain

Almaraz NPP is owned by the companies Iberdrola Generación S.A.U. (53%), Endesa Generación S.A. (36%) and Gas Natural SDG S.A. (11%) and operated by Centrales Nucleares Almaraz-Trillo (CNAT). NPP Almaraz comprises two Westinghouse three-loop pressurised water reactors (PWR). Almaraz 1, net capacity of 1011 MWe, started commercial operation in 1983, Almaraz 2 (1006 MWe) followed in 1984.

The site is located on the left bank of the Arrocampo brook reservoir, 180 km west-southwest of Madrid. The shortest distance to Portugal is about 100 km. A dam – Valdecañas – is situated halfway down the course of the River Tagus, upstream from the site, having a storage capacity of 1146 hm^3 at its typical maximum level.

The Spanish Nuclear Safety Council, Consejo de Seguridad Nuclear (CSN), published the Spanish stress tests report.[9]

Weaknesses the Spanish stress tests Describes [CSN 2011]

- Within the stress test programme, the scope of the seismic margin analysed has been extended. Corrective actions and further analyses are necessary.
- The analyses of the impact of external flooding caused by a Valdecañas dam break are not sufficient. CNS states that the way the licensee postulated the break is less challenging than applied in Spanish practice by the dam emergency plans. The licensee was required to review its analyses.
- Additional checks regarding the possible effects of flooding caused by the entry of water via paths located below the elevation of the plant grading level, such as a rise in the water table or ingress via the rainwater drains are required.
- To increase the safety margins against external flooding beyond the design basis, the licensee has to implement several design modification (increase the capacity of drains, water tightness of doors).
- The protection against atmospheric discharges (lightening) has to strengthen.
- Currently, Almaraz NPP has only one additional air-cooled diesel generator (5 DG) to handle a station black out (SBO) situation. If both units are affected by SBO, the licensee foresees the alignment of the fifth DG alternately to each of the two units.
- If the fifth DG fails (total SBO), the unit would initiate cooling of the primary system by using steam generator relief valves and Turbine-driven auxiliary feedwater pump (AFWTP). The operation of the turbine-driven pump is limited to 16 hours with the existing batteries.
- The water available in the auxiliary feedwater and condensate tanks would allow adequate core cooling for only 30 hours; steam generator (SG) dryout occurring after 37 hours, uncovering of the core after 41 hours and rupturing of the vessel after 49 hours.
- To cope with SBO type events for at least 24 hours with the equipment existing at the site, and 72 hours with only light equipment provided from off-site (stress tests specification), a lot of improvements will be necessary, e g. providing portable equipment.
- The ultimate heat sink (UHS) consists of two water reservoirs and the essential service water system (ESW). Each of the reservoirs guarantees the heat removal from both units. But if the

9 The Spanish stress tests report differs from other national reports, because it lacks subsections.

ESW intake fails, the ESW would be lost. This could cause a SBO situation for both units, because cooling for the two redundant emergency diesel generator of each unit is accomplished by means of the ESW.

- There are no adequate measures to manage a severe accident in the reactor. The licensee plans to implement portable equipment to inject water into the primary system, the steam generators and the containment spray system.
- There is no adequate measure to manage the large amount of hydrogen expected in the case of a severe accident in the containment. The installation of passive autocatalytic recombiners (PAR's) is planned.
- There is no filtered venting system to mitigate the radiological consequences in case of a containment overpressure.
- The current spent fuel pool cooling and water make-up alternatives would not be available in a SBO situation, except the fire protection system – but it is not resistant to seismic events. After loss of cooling boiling starts after 14.8 hours and reach the upper part of the fuel assemblies after 160.1 hours.
- During refuelling outage (all the fuel assemblies in the pool) calculated time until boiling is only 5.4 hours and 58.3 hours until fuel uncovering. The loss of the shielding would occur at a water level of 3 metres above the upper part of the fuel, after 38.2 hours.
- The licensee has to strengthen the cooling capacity in the event of SBO and earthquake through seismic qualification and availability of portable equipment for pool water inventory make-up. Furthermore the spent fuel temperature and level instrumentation has to be improved, regarding the range, the seismic qualification and availability in the control room; there should be also portable instrumentation available in SBO situation.
- To organise the emergency management, the licensee plans an alternative emergency management centre (AEMC) in the long term (2015 – 2016).

Weaknesses the Spanish stress test Ignored

The vulnerability of the units at Almaraz concerning air plane crashes is the same as for old US reactors of this type. A crash of a large or a midsize airliner is very likely to cause a major damage of the reactor building.[10] Such a crash – accidentally or deliberately – can result in a severe accident. The spent fuel pools are located in buildings adjoining the reactor buildings. These buildings are only ordinary industrial buildings. If the walls of a spent fuel pool should be damaged, large amounts of radioactive material could be released. These buildings, however, are located at lower altitudes and, therefore, are not necessarily hit by a crashing aircraft.

A US NRC report published in 2005 states that "successful terrorist attacks on spent fuel pools, though difficult, are possible." [CRS 2005] Once the pool is damaged and the water drained off water boiling start much earlier. Radiation shielding is completely lost at the moment the fuel is exposed. Intervention becomes already impossible, when the water level is 0.9 meter above the fuel, because of the high radiation dose rates. Freshly discharged fuel would then reach the point where it burns in air (900 °C) and very severe radioactive releases start within hours [ALVAREZ 2003; USNRC 2001].

A US report released in May 2011, states that at least 66 accidents with a significant loss of spent fuel water occurred over the last 30 years at US plants. The report warns that the high density

10 A generic study carried by order of the German Federal Environment Ministry (BMU) revealed, among other things, that the crash of even a small-sized commercial aircraft (e.g. an Airbus A320) against a reactor building, which has a wall thickness of 0.6 to 1 metres, would result in a major damage to the reactor building [BMU 2002].

racks in pools were not designed to handle the amount of spent fuel, as well as the burn-ups, they now hold. Corrosion of neutron-absorbing panels in the racks could endanger the preventing nuclear chain reaction [NF 2011].

In June 2010, the ten-year operating licenses[11] for Almaraz 1 and 2 were renewed [WNN 2010]. It has to be expected that ageing becomes a relevant issue for the next, the fourth, decade of operation. Despite this fact, the thermal power of both units has been increased, accelerating ageing processes further.

Unit 1 received the permission to operate at the new maximum power on 15 April 2010, unit 2 on 13 April 2011 – one month after the accident at the Fukushima Daiichi NPP. A small mini-uprate project increasing thermal power of 1.6% was carried out in 2003 by recapturing measurement uncertainty through more precise instrumentation – this means only through reducing safety margins. Work on the extent power uprating project at Almaraz began in January 2007. The power uprating added an additional 8% to bring the units up to 2947 MWth. This is an increase of nearly 10% from the initial plant capacity of 2686 MWth [NEI 2011]. Effects related to power uprating could possibly initiate an accident. However, these effects would certainly aggravate an accident situation (see 2.2.1.2).

CONCLUSION

The units Almaraz 1 and 2 have currently no effective accident management measures to prevent or mitigate a severe accident (like filtered venting). However the power supply and heat removal deficiencies at the plant make a severe accident likely. In addition no well-founded analysis regarding the impact of a potential dam break was conducted. In case a flooding would damage the water intake, a dangerous station blackout (SBO) situation occur, which could affected both units.

One of the most important "Lessons learned" by Fukushima is to realise the hazard of spent fuel pools. The hazard related to the Almaraz spent fuel pool was not realised before.

Portable equipment is presented as the future solution to compensate deficiencies of reactors and spent fuel pools. The envisaged measures however can hardly be implemented under accident conditions – and as quickly as necessary. An expert criticized additional portable equipment to cope with a severe accident as "*a highly unpredictable mixture of desperate measures*" [NW 2012a].

In the current situation, the units have to stop operation or at least operate with reduced power. However, unit 2 was given the permission to operate with an increased power last year – one month after the accident at Fukushima.

KEY RESULTS OF THE PEER REVIEW

EARTHQUAKE

The design basis earthquake (DBE) is considered to be consistent with international standards. There is a broad coverage of indirect effects of earthquakes, including effects on nearby hazardous industries that could be seen as a good practice. Within the framework of the seismic hazard update claimed by CSN, it is suggested to include geological and paleoseismological data characterizing the relevant active faults. There is evidence that a continuous improvement process by PSRs has been applied. The seismic margin assessment (SMA) shows a significant margin beyond the design bases. In addition, the seismic capacity of all the SSCs on the path to safe shutdown and other safety relevant SSCs will be upgraded up to 0.3 g.

Beyond the improvement measures already implemented by the licensees or just claimed by CSN,

11 Spanish reactors are granted ten-year operating licenses.

no additional measures are suggested by the peer review team.

FLOOD

The design basis flood (DBF) is considered to be consistent with international standards. The analyses cover a wide range of phenomena such as the effects of intense local rainfall, overflowing of rivers and ravines, tsunamis, rising sea or groundwater levels and rupture of dams. The design bases for external flooding have reviewed. Both the licensees and the CSN conclude that they remain valid. Revaluation of flooding hazard is part of the PSR process. Flooding design bases is also covered by CSN's supervision and inspection processes.

The safety margin above DBF for river flood / dam break scenario is currently estimated to be about two metres at Almaraz; but complementary studies are ongoing that could result in a reduction of the margin. A number of improvements of the robustness against external flooding are proposed by the licensee. It is suggested to consider improving the external flood volumetric protection of buildings containing safety related SSCs.

EXTREME WEATHER

A probabilistic methodology has been used as preliminary screening for events other than earthquakes and flooding that constitute the specific scope of the stress tests based on a probability of occurrence of 10^{-5} per year, in accordance with current international standards.

The national report claims that the design criteria applied and the margins estimated are reasonable, although in certain cases, such as the margins with respect to extreme temperatures, the results submitted are still being reviewed by the CSN.

There are some improvement measures ongoing and others will be required by the CSN. Further detailed information on combined external events has been required by CSN and the additional analyses are ongoing.

LOSS OF POWER AND LOSS OF UHS

The national report indicates that comprehensive complementary safety analysis has been done by the utilities for all NPPs. Measures are proposed to increase safety capabilities in case of LOOP, SBO and loss of UHS without any external support according the ENSREG specifications, and have been assessed by CSN.

As a reaction to the Fukushima event, the utilities and CSN also carried out inspections on all structures, systems and components (SSCs) relevant to LOOP, SBO and loss of UHS. As a result of these inspections, analysis and measures in respective areas are now planned and prioritized.

The possibility to use manually operated means is available for all NPP, nevertheless in some cases further analysis is under way to ensure that these dedicated manual actions can be performed under unfavourable working conditions. The manual operation of turbine-driven pumps and relief valves without DC power to remove residual heat from reactor core have been analyzed and already tested in Almaraz NPP.

The CSN evaluation has identified the need to generally analyse an additional potential improvement to address situations in which a complete loss of power might occur during the shutdown conditions.

Measures for improvement of the capabilities to cope with cooling of spent nuclear fuel pools are necessary.

The peer review team considers that the planned actions, in some cases already implemented or in the process of being incorporated, are in accordance with ENSREG specifications. The reviewers stated that CSN, and also the licensee, should complete the implementation plan of improvement measures as well as consider the conclusions and significant international recommendations regarding the Fukushima event.

Severe Accident Management

Following the US regulatory practice, severe accident management was not included in the licensing bases of the Spanish NPP. In the late nineties CSN started to consider this issue in PSR. Recently, CSN sent new directives to each NPP (including requirements for hydrogen control, containment filtered venting), with the objective of reinforcing the compliance of the Spanish plants with the WENRA AM reference levels. An important weak point is the lack of filtered containment venting system.

CSN clarified during the peer review that they currently do not require to develop the SAMG for shutdown states or AM procedures for the mitigation of spent fuel pool (SFP) accidents, but will request the licensees to do so.

Existing design features combined with existing AM measures reveal the existence of time margins for the control or mitigation of severe accidents in Spanish NPPs. However, the assumptions underlying these margins may deserve verification. The review team stated, in particular, the values quoted for Almaraz seem too long.

The review team assumes the improvements identified by the licensees and CSN will all be important in increasing the robustness of the plants. In addition the following recommendations made by the peer review team should be considered:

- Complete the establishment of a comprehensive set of requirements for AM integrated within the Spanish legal framework;
- Include AM as an explicit topic in CSN’s safety guide on the content of the PSR;
- Develop SAMGs for accidents initiated during shutdown operation and accelerate plans to include SAMGs addressing mitigating aspects for spent fuel pools;
- Fully include external events in probabilistic safety assessments including assessments of reliability of AM under such conditions.

CRITICAL COMMENT

The protection against flooding, earthquake and extreme weather events seems to be almost adequate. However, the reactor and the spent fuel pool are relatively vulnerable against other external events like an airplane crash. In case of a total station blackout (SBO) and/or loss of ultimate heat sink, measures to prevent severe accidents or to mitigate their consequences are very limited. Thus, large radioactive releases could result.

The necessary comprehensive back-fitting programme will not remedy all shortcomings. Furthermore, it will take several years to implement. To cope with the workload these activities will entail, the peer review team recommended that CSN’s technical assessment human resources should be reinforced.

4.2 NPP Doel, Belgium

The Doel nuclear power plant (NPP) is operated by Electrabel, a subsidiary of the GDF-SUEZ Group. The Federal Agency for Nuclear Control (FANC) prepared the Belgian Stress Test Report.

The site is located on the left bank of the Scheldt river, at 15 km northwest of Antwerp and at only 3 km from the border between Belgium and the Netherlands. The site houses four pressurized water reactors (PWR): the twin units Doel 1/2, commissioned in 1975, Doel 3 (1982) and Doel 4 (1985). The units Doel 1/2 are Westinghouse 2-Loop reactors with an electric power of 433 MWe each; unit Doel 3 (1006 MWe) and unit 4 (1039 MWe) are Westinghouse 3-Loop reactors.

The following chapter discusses the weaknesses of the older units, Doel 1/2.

Weaknesses Belgian stress tests Describes [FANC 2011]

- The Probabilistic Safety Analyses (PSA) for Doel NPP did not take into account the risk of fire and flooding; furthermore the PSA did not consider any event related to the spent fuel pools.
- The original design of the units Doel 1/2 did not take into account the risk of an earthquake. In 1985, the Peak ground acceleration (PGA) of the Design Basis Earthquake (DBE) was subsequently set at 0.058g. According to IAEA recommendations a value of 0.1g has to be used (see 2.1.1.1 Earthquake).[12]
- Following the accident in Fukushima, Electrabel commissioned the Royal Observatory of Belgium (ROB) to conduct a new seismic study, taking into consideration the recent developments in seismic hazard assessment. This resulted in a new value for the PGA of 0.081 g.
- But as suggested by the ROB, FANC require a more elaborated study, e.g. with due consideration of results arising from the EC-project SHARE (seismic hazard harmonization in Europe).
- The effect of a fire induced as a result of an earthquake is not considered. A fire is a hazard particularly to old nuclear power plants like Doel 1/2, because of the limited physical separation of the redundant safety systems. A fire has the potential to damage all these systems simultaneously.
- The flood level of the design basic flood (DBF: high tide + storm surge, 95th percentile for a return period of 10,000 year) remains below the minimum height of the embankment. But flooding of the site can occur in case of a combination of very high Sheldt level with an embankment breach. The initiation of an embankment failure can occur for a severe storm with a return period of 1,700 years.[13]
- To evaluate the safety margin, in case of an embankment breach near the site, the most severe storm is cumulated with a high level of the Scheldt. The maximum Scheldt level for this storm is postulated 10.2 m (85 cm higher than DBF[14]). For this scenario, the water would reach the first buildings very fast (after about one hour) and significant water depths could be found around several buildings (between 20 and 50 cm). Additionally the site, situated on a raised platform surrounded by lower-lying polders, will become an island. Electrabel claimed however, that the probability of a significant cliff edge effect is very low. A weak point is a number of buildings cannot guarantee tightness in case of tens of cm water flood the site.
- In case of an embankment breach, sand bags or mobile barriers are planned to be installed

12 In accordance with the regulatory practice, a DBE of 0.1g was used for the design of Doel 3&4.
13 The embankment fails in case of a subsequent storm event if no repairs were performed in the time after initiation.
14 This safety margin is not sufficient: in frame of the German stress test, a plant has to resist against a flood level, which is one meter above DBF only to meet the first of three levels of robustness.

at sensitive building entrances by the operator.

- A point neither FANC nor Electrabel consider is that a flood may transport debris of all types which may physically damage structures, obstruct water intakes or damage the water drainage system (see 2.1.1.2 Flooding).
- FANC requires reassessing of the capacity of the sewer system regarding both short-duration heavy rains and long-lasting rains, and a redefinition of 100-yearly rains.
- According to FANC, given the fact that tornadoes of high intensities were observed in the past years in the neighbouring countries, the robustness of the second level system of Doel 1/2 should be confirmed in case of a beyond design tornado with wind speed exceeding 250 km/h (70m/s). Electrabel has claimed that the probability of these tornadoes is very low and that the buildings probably resist such a tornado.
- In case of total station blackout (SBO), Electrabel should assess whether all containment penetrations can be closed in due time and whether the relevant containment isolation systems remain functional, in particular during outage situations.
- In case of a total SBO, only the turbo-pump of the auxiliary feedwater (AFW) system remains available in the short term to feed water into the steam generators. After one and a half hour the first cliff edge effect appears: the auxiliary feedwater reservoirs are empty, the steam generators can continue cooling the primary circuit for several hours only. There are limited possibilities to refill the AFW tank.
- If the cooling via steam generators fails, the primary circuit begins to boil and steadily loses its water volume. This results in uncovering and later to melting of the fuel, the relocation of the corium towards the bottom of the reactor pressure vessel (RPV) and the piercing of the bottom of the RPV. Without operator intervention this process takes between 2 and 3 hours.
- During a severe accident when the core has melted through the reactor pressure vessel and residual heat removal has failed, pressure in the containment rises. Venting could prevent containment collapse, but none of the four units is equipped with filtered venting. A feasibility study will be carried out to fit a filtered vent. A second study assesses the residual risk of hydrogen accumulation in the spent fuel pools buildings.
- In case of a total SBO and/or loss of ultimate heat sink, Electrabel plans to use new non-conventional means (NCM) to refill the steam generator and the spent fuel pool, to avoid the overpressure in the reactor building and to restore the electrical power supply to instrumentation and control panels. However, the operability and practicability of the NCM are not proved.
- Furthermore, Electrabel has to reinforce its emergency organization. The full implementation of the new emergency organization will be effective in 2013.

WEAKNESSES THE BELGIAN STRESS TESTS IGNORED

With respect to the limited number of initiating events considered at the design phase, the units of Doel 1/2 have significant design deficits [FANC 2011]:

- Not all of the first level safety systems[15] (operated from the main control room) are physically separated and/or design basis earthquake resistant. The accident leading to the potential unavailability of multiple first safety systems are covered by the second level system that are operated from a separated control room. But the second level systems are not housed in a bunkered building and the second level systems are mainly manually operated from

15 The first level safety systems intended for incidents and accidents of internal origin and earthquakes, and the second level emergency systems dedicated to external hazards.

emergency control room.

- Doel 1 and 2 share the control room and several first level systems. This increases the probability that both units are affected in case of an incident.
- The low level safety injection pumps have duties in the normal operation (no independence between the levels of the-defence-in-depth concept, (see Chapter 3).
- The physical separation of the electrical power supply and instrumentation cabling is limited.
- The fire extinction water system has no seismic design (with exception of the filling of the second level feedwater).
- The number of redundant safety systems is low compared with current state-of-the- art even compared with Doel 3 and 4.
- The spent nuclear fuel is stored in pools in the nuclear service building instead in a bunkered nuclear fuel building.
- The inner containment consists of a steel bulb instead of a pre-stressed concrete.[16] This is a major weakness in the case of a core melt accident, because steel fails quicker than concrete.

The overall concept of defence in-depth and therefore the prevention of accidents are not sufficient. These design weaknesses can potentially aggravate or even trigger an accident.

In addition to the design deficits the operational practices also show significant weaknesses: In March 2010, an IAEA Operational Safety Review Team (OSART) visited Doel 1/2 to review operating practices. 15 safety issues were identified by the team. According to OSART-follow up mission (5 to 8 March 2012), there are still some important issues [FANC 2012c; IAEA 2010b]:

- Analyses for some events are not being performed to the required depth and rigor and they are not being completed in a timely fashion. Without thorough and timely analysis, data cannot effectively be used to prevent repeat events, or more serious events.
- The plant has not fully updated the Safety Analysis Report (SAR) to reflect the current status and the licensing basis of the plant. The last update of the external hazard chapter was in 1992.
- In certain plant areas inadequate conditions exist due to lack of attention and insufficient maintenance workmanship. Deficient material conditions could lead to deterioration of the equipment and systems, resulting in their unreliability.
- In some areas of the electrical building, cable separation schemes and compartmentalization are inadequate. Thus, the risk of an electrical fire is increased.

The Belgian Federal Government demanded terrorist attacks (aircraft crash) and other man made events (cyber attack, toxic and explosive gases, blast waves) to be included as possible triggering events in the Belgian stress tests program. The assessment of these man-made events were however not in the scope of the EU stress tests programs, and are thus developed in a separate national report [FANC 2012b]. The results of the main topics are:

- In case of an aircraft crash (accidental or intentional) significant damage can occur to the external concrete structure, with the possibility of projectiles penetrating into the containment. The extremely likely failure of the cooling system would result in a severe accident of the most hazardous category: core melt with an open containment. The radioactive releases

16 The secondary (outer) containment consists of a reinforced concrete cylinder on which a reinforced concrete hemispherical dome is placed and encapsulates the primary containment, thus protecting it against accidents.

would be very high and occur particularly early.

- The implementation of additional physical obstacles around the sites is being studied, by Electrabel. Furthermore, the capacity for extinguishing a kerosene fire and a way to replenish the pools with water in the event of cracks in the foundation is being investigated.
- FANC formulated additional requirements: Extending the emergency plans and procedures for each unit; necessary improvements need to be implemented, based on a comparison of the existing non-conventional emergency means with the “Extensive damage mitigation guidelines” of the US Nuclear Regulatory Commission.
- The emergency control room has to protect from toxic gas and the main control has to reinforce of air-tightness to guarantee the habitability. Furthermore the ventilations systems are not equipped with explosive gas detectors.
- According to Electrabel, the loss of safety functions resulting from cyber attack is impossible in all units. But this conclusion is only based on engineering judgment. Therefore FANC recommends performing a security evaluation (with the assistance of external IT-experts) in order to decrease the risk against cyber-attacks.

CONCLUSION

A major deficiency is the lack of a filtered venting system. Filtered venting constitutes a key system in mitigation of severe accidents. Several countries started to require filtered venting systems decades ago.

The safety margins regarding flooding are not sufficient, although the embankment can fail and the protection of the plant against flooding the buildings relies on sandbags or mobile barriers only. The safety margins regarding an earthquake have not been assessed sufficiently.

Material degradation, the mentioned design weaknesses and operation shortcomings of the very old Doel 1/2 can significantly aggravate the development of an accident caused by an external impact.

Another major weakness of Doel 1/2 is the vulnerability against air craft crashes. When Germany was taking its decision on the closure of eight nuclear power stations last summer, one of the important arguments was that their protection against terrorist attack was very low.

In November 2011, Belgian's political parties have reached a conditional agreement to phase out nuclear power by 2025, if they can find an adequate supply of energy from alternative sources by that time. The three oldest reactors (including Doel 1/2) are set to be shut down by 2015, with the rest taken off the grid by 2025 [ED 2011]. It has to assumed, that FANC will not require comprehensive and expensive measures for the remaining time.

The Doel NPP is situated close to the centre of Antwerp. In case of a severe accident, the evacuation of all the people on time is nearly impossible. Because of the lack of filtered venting, the threat of a relative high radioactive exposure exists.

To assess the risk, both the probability and the potential consequences of a severe accident need to be considered: Doel 1/2 poses a high risk to the environment.

Considering all facts, we recommend to shut down Doel 1/2 immediately.

Regarding units Doel 3 and 4, a transparent action plan for dealing with the identified deficiencies should be defined. The units need to be taken out of operation until all measures of high priority (e.g. filtered venting) are implemented.

Summary of the Belgian Country report including recommendations of the peer review team see on chapter 4.3 NPP Tihange.

4.3 NPP TIHANGE, BELGIUM

The Tihange nuclear power plant (NPP) is operated by Electrabel, a subsidiary of the GDF-SUEZ Group. The site is located on the Meuse river, at 25 km southwest of Liege (about 200,000 inhabitants) and at about 80 km southeast of Brussels (region Brussels: 1 million inhabitants).

The Tihange NPP comprises three pressurised water reactors (PWR): Tihange 1, commissioned in 1975, Tihange 2 (1983) and Tihange 3 (1985). Tihange 1 is a Framatome 3-Loop reactor with a net capacity of 962 MWe; Tihange 2 (1008 MWe) and Tihange 3 (1042 MWe) are Westinghouse 3-Loop reactors.

The following chapter discusses, particularly, the weaknesses of the old unit **Tihange 1**.

The Belgian regulatory body, the Federal Agency for Nuclear Control (FANC), published the Belgian stress test report.

WEAKNESSES THE BELGIAN STRESS TESTS DESCRIBES [FANC 2011]

- The Probabilistic Safety Analyses (PSA) for Tihange NPP did not take into account the risk of fire and flooding; furthermore the PSA did not consider any event related to the spent fuel pools.
- Tihange 1 was designed to withstand a Design Basis Earthquake (DBE) characterized by peak ground acceleration (PGA) of 0.1g. During the first periodic safety review (1985) the PGA has been raised to 0.17g.
- In April 2011, following the accident in Fukushima, Electrabel commissioned a probabilistic seismic hazard analysis (PSHA) using a state-of-the-art methodology. This PSHA resulted in an increase of the PGA to 0.23 g.
- FANC require a more elaborated study, e.g. with due consideration of results arising from the EC-project SHARE (seismic hazard harmonization in Europe). The peer review recommends that FANC monitors the completion of the updated PSHA, the implementation of the consequential measures and the updated assessment of safety margins.
- The seismic margin review shows that in total 21 systems, structures and components (SSC), e.g. main switchboards and transformers, have a low probability of resisting an earthquake exceeding the review level earthquake (RLE) of 0.3g. FANC requires a detailed action plan for the required improvement.
- Strengthening of the electrical building to enhanced protection against external hazards is necessary, however, this will only be assessed (not implemented) in 2012.
- The effect of a fire induced by an earthquake is not considered.[17]
- In case of a design basis earthquake (DBE), the autonomy of the emergency diesel generators of the second level safety systems is only 7.5 hours.[18]
- According to the plant visit team, the seismic instrumentation appeared to “offer an opportunity for improvement.”
- The original design basis flood (DBF) was fixed with reference to the practice used in civil engineering, which is the flow rate derived as the highest historically recorded flood level of the river increased by 20% (2220 m^3/s).

17 The peer review team recommends to analyse the impact of failure of a fuel tank containing 500 m^3 fuel, which is not seismically qualified.

18 The peer review team recommends increasing the autonomy of these EDGs.

- The important floods in 1993 and 1995 in the Meuse valley reached nearly this flow rate, thus a reassessment of the flood risk has been conducted – but with the same outdated methodology. The calculated flow rate (2615 m³/s) corresponds with a water level of the same height as the site platform elevation.
- During the re-assessment within the latest Periodic Safety Review (PSR), new DBF parameters have been derived using the probabilistic approach, according international standards. The corresponding flood rate of a return period of 10,000 years (3488 m³/s) could result under exceptional circumstances combining rapid melting of snow and a long period of heavy rain.
- Corresponding water heights of the decamillennial flood would largely exceed the site platform elevation (up to 1.70 m), causing flooding of the three units and loss of safety related equipment, including all on site AS power sources and both primary and alternate ultimate heat sink.
- Tihange 1, which is located most upstream, will be the first unit to be hit. Already at a flow rate with a return period of about 400 years (2800 m³/s), the unit is completely surrounded by water and all buildings except the reactor building will be flooded (first cliff edge effect). The systems providing the cooling of the pool are out of service, likewise the shut down cooling system. (The ultimate means circuit (CMU) equipment has to take over).
- The second cliff edge effect is a flow rate that occurs statistically nearly every 600 years (2,900 m³/s), corresponding to the flooding of equipment in Tihange 1. Even the auxiliary feed water system gets lost in Tihange 1. (Only the CMU could feet the steam generators and guarantee core cooling.)
- The third cliff edge effect occurs at 3,000m m³/s: The auxiliary feedwater system of Tihange 2 fails. (CMU has to use for cooling the fuel).
- The fourth cliff edge effect occurs for flow rates between 3000 and 3300 m³/s: Loss of cooling for Tihange 3. (Only CMU could cool the fuel in the reactor and in the pool).
- Electrabel has proposed to implement three levels of defence against flooding, which have in principal been accepted by the regulatory body:
- The first level is a peripheral protection of the site, which shall consist of a wall[19] of a height greater than of the Meuse in case a decamillennial flood and shall be installed not before 2014.
- A local protection of the buildings in order to cope with a local failure of the protection wall triggering a sudden inrush of water on the site. This second level of defence shall consist of coffer dams and other sealing devices that will be installed during the flooding alert period.
- When conventional equipment is rendered unavailable through flooding, the CMU equipment preinstalled during the alert should be used, which is the third level of defence. The robustness of the currently installed non-conventional means (NCM), i.e. the so-called CMU should be further improved.
- Since the CMU is currently needed for floods with a return period exceeding 100 to 400 years, FANC requires determining specific provisions as applicable to equipment important for safety; the technical characteristic of these NCM should account adverse conditions.
- A dedicated emergency strategy has been requested by FANC. The task of the emergency team is to initiate the shutdown of the units and the deployment of the CMU equipment. As

19 The wall would include coffer dams which, in case of threat of flooding could be used to close the opening necessary for normal operation of the NPP.

soon as a flow rate prediction of 2500 m³/s, the Internal Emergency Plan starts.

- FANC requires means for onsite transport of personnel and equipment towards the units, while the site is flooded. Movement on the site is only possible by boats.
- Fire or explosions potentially induced by the flooding are not been examined yet. FANC requires taking additional measures because the automatic fire extinction system is lost during the flooding.
- The peer review team recommends implementing all suggested measures for the Tihange site since the analysis shows relatively low protection of the site against flooding. The team recommended furthermore that Electrabel shall include a safety margin for the flood protection wall to adequately cover uncertainties associated with a 10,000 year flood.
- A point neither FANC nor Electrabel consider is that a flood may transport debris of all types which may physically damage structures, obstruct water intakes or damage the water drainage system (see chapter 2.1.1.2).
- Given the fact that tornadoes of high intensities were observed in the past years in the neighbouring countries, FANC requires to confirm the robustness of the second level system of Tihange 1 in case of a beyond design tornado (wind speed exceeding 250 km/h). Electrabel has claimed that the probability of these tornadoes is very low.
- FANC requires the reassessing of the capacity of the sewer system regarding both short-duration heavy rains and long-lasting rains, and a redefinition of 100-yearly rains. However, the peer review team recommends the derivation of design basis parameters with 10,000 years return periods; attention should be dedicated also to extreme temperatures.
- In case of loss of the primary ultimate heat sink combined with a total Station Black-out (SBO) the units can no longer use the water from the alternate ultimate heat sinks (groundwater or deep water intakes) due to lack of electric power.
- In case of a total SBO, only the turbo-pump of the auxiliary feedwater (AFW) system remains available to feed water into the steam generators. After three hours the first cliff edge effect appears: the auxiliary feedwater reservoirs are empty.[20] If the SBO happens about one hour after the reactor emergency shutdown, this time spam is slightly prolonged to 7 hours. There are limited possibilities to refill the AFW tank.
- If the cooling via steam generators fails, the primary circuit begins to boil and steadily loses its water volume. This results in uncovering and later to melting of the fuel, the relocation of the corium towards the bottom of the reactor pressure vessel (RPV) and the piercing of the bottom of the RPV. Without operator intervention this process takes between 2 and 3 hours. In case the primary system is open, the reactor water starts to boil within 30 – 60 minutes.
- In case of a total SBO and/or loss of ultimate heat sink, Electrabel plans to use new non-conventional means (NCM) to refill the steam generator and the spent fuel pool, to avoid the overpressure in the reactor building and to restore the electrical power supply to instrumentation and control panels. However, the operability and practicability of the NCM are not proved.
- FANC requires to asses that in case of total SBO, whether all containment penetrations can be closed in due time and whether the relevant containment isolation systems remain functional, in particular during outage situations.
- During a severe accident when the core has melted through the reactor pressure vessel and residual heat removal has failed, pressure in the containment rises. Only venting could

20 At Tihange 2, the tank allows the feeding of the steam generators during 17 hours, at Tihange during 23 hours.

prevent containment collapse. Filtered containment venting is considered state-of-the-art for some years, but none of the three units of Tihange NPP is equipped with a filtered venting system. A feasibility study will be carried out to fit these devices. The peer review team emphasizes its importance and recommends considering sub-atmospheric pressures in the containment in that matter.

- A second study will be conducted to assess the residual risk of hydrogen accumulation in the spent fuel pools buildings. The peer review team recommends, regardless of the outcome of this study, the considering of the installation of passive autocatalytic hydrogen recombiners (PARs).
- Electrabel has to reinforce its emergency organization. The full implementation of the new emergency organization will be effective in 2013.

WEAKNESSES THE BELGIAN STRESS TESTS IGNORES

With respect to the limited number of initiating events considered at the design phase, Tihange 1 has significant design deficits:

- Only partially physically separated redundant safety systems. Because of this a fire has the potential to damage all these systems simultaneously.
- The safety injection system pumps have duties in the normal operation, i.e. there is no independence between different levels of the-defence-in-depth (see chapter 3).
- The second level emergency systems had not been considered in the initial design. As a result of the periodic safety review, an emergency system was installed to respond to several accidental scenarios of external origins. However, the second level emergency system only includes two emergency power supplies, one water cooling circuit and an injection pump to the primary pumps seals.
- The thickness of the basement is only 2.15 m. Thus, the time for a potential containment basement melt through is relative short.

The overall concept of defence in-depth and therefore the prevention of accidents are not sufficient. These design weaknesses can potentially aggravate or even trigger an accident.

Tihange 1 is in operation for nearly 40 years (37 years). This means that ageing of materials is a major safety issue in the plant. It has to be expected, that the frequency of ageing related incidents will increase. These incidents have the potential to trigger, but particularly to aggravate accidents. Incidents could also indirectly be caused by ageing: If degraded components are replaced, defective mounting or other errors will be possible. This has been shown the experience at nuclear power plants around the world.

Recently, two safety relevant incidents have been occurred at Tihange 1 that are probably direct or indirect related to ageing. Both incidents are categorized to level 1 of the International Nuclear Event Scale (INES) [FANC 2012c]:

- On 16th of January, 2012, it has been noticed during the monthly check that an emergency diesel generator was out of order since seven days. The failure was caused among others by an installed component which was not supplied correctly.
- On 7th of February, 2012, it has been found that a group of heating elements of the pressuriser has been out of order.

The Belgian Federal Government demanded terrorist attacks (aircraft crash) and other man-made events (cyber attack, toxic and explosive gases, blast waves) to be included as possible triggering events in the Belgian stress tests program. The assessment of these man-made events were not in the scope of the EU stress tests programs, and are thus developed in a separate national report

[FANC 2012b]. The results of the main topics are:

- In case of an aircraft crash (accidental or intentional) significant damage can occur to the external concrete structure, with the possibility of projectiles penetrating into the containment. The extremely likely failure of the cooling system would result in a severe accident of the most hazardous category: core melt with an open containment. The radioactive releases would be very high and occur particularly early.
- For Tihange 1 a study is undertaken for the construction of a new "bunkered" building resistant to an aircraft crash and which would house the second level emergency systems.
- The implementation of additional physical obstacles around the sites is being studied, by Electrabel. Furthermore, the capacity for extinguishing a kerosene fire and a way to replenish the pools with water in the event of cracks in the foundation is being investigated.
- FANC requires extending the emergency plans and procedures based on a comparison of the existing non-conventional emergency means with the “Extensive damage mitigation guidelines” of the US Nuclear Regulatory Commission.
- The main control room at Tihange 1 has no automatic isolation of the ventilation in the event of the detection of toxic gas. This endangered its habitability. Furthermore the ventilations systems are not equipped with explosive gas detectors.
- FANC recommends performing a security evaluation (with the assistance of external IT-experts) in order to decrease the risk against cyber-attacks.[21]

CONCLUSION

The Tihange NPP currently does not comply with the requirements regarding flood protection. Design basis flood with statistical return period up to 10,000 years has to be implemented according international state-of-the-art. In case of this decamillennial flood the water level on the Tihange site is nearly two meters and all safety systems of the three units are flooded and not operational. It is not believable that the staff moving by boats around the site with so far not appropriate equipment will be able to prevent severe accidents of all units, particularly of Tihange 1. Even if the reactor is shut down precautionary there are only hours to implement water and power supplies. The current flood protection as well as the possible consequences remind of the disastrous accidents at Fukushima NPP in 2011.

The Tihange site is currently only protected by its design against a flood with a statistical return period up to 400 years. Tihange 1, which is hit by flooding at first, is the less protected unit of the site. Damage to all safety relevant equipment would be caused already by floods with return periods of 600 years.

Tihange 1 shows a lot of further shortcomings: Another major weakness is the vulnerability against air craft crashes. The seismic margins have not been assessed sufficiently yet, however, there are identified a lot of deficiencies. A fire, e.g. triggered by flooding or earthquake, is also a potential danger.

Material degradation and design weaknesses of the very old Tihange 1 can significantly aggravate the development of an accident caused by flooding or other external or internal events.

Because of the lack of filtered venting, which is a key system in mitigation of severe accidents, the probability of high radioactive release in case of such a severe accident is very high.

Tihange NPP is situated close to the centre of Liege. In case of a severe accident with a major

21 According to Electrabel, the loss of safety functions resulting from cyber attack is impossible in all units. But this conclusion is only based on engineering judgment.

radioactive release, the evacuation of all the people on time is nearly impossible. Furthermore the capital city Brussels will be affected by a severe accident in case of specific wind directions.

Both the probability and the potential consequences of a severe accident are relatively high, therefore the risk of Tihange 1 unjustifiable high. Considering all facts, we recommend to shut down Tihange 1 immediately.

Tihange 2 and Tihange 3 need to be taken out of operation until all measures of flood protection are implemented; a transparent action plan for dealing with the identified shortcomings should be defined.

KEY RESULTS OF THE PEER REVIEW

EARTHQUAKE

The method chosen to estimate safety margins and cliff edge effects does not assess the seismic robustness of the structures, systems, and components (SSCs), but rather looks at the probability of the SSCs to withstand a certain Review Level Earthquake (RLE). The RLE values were estimated to envelope those calculated by the recent PSHA (RLE for Tihange = 0.3g, RLE for Doel = 0.17g).

In general results demonstrate the capacity of Tihange and Doel NPPs to resist also earthquakes of higher intensity than the one considered in the design; although for a limited number of SSCs it is unlikely that those will sustain the RLE. Those components that showed the resistance to RLE with low probability will be subject to more precise calculations and modifications will be realized.

A feasibility study for strengthening the Electrical Auxiliary Building at Tihange 1 is required.

During the country visit the review team was informed that **the fuel tank of Tihange 1, containing 500 m3 of fuel is not seismically qualified**. The recommendations include analyses to verify whether the impact of failure of this tank (e.g., fires, flood) induced by an earthquake is covered.

The peer review resulted in the recommendation that the national regulator monitors the completion of the updated PSHA, the implementation of the consequential measures and the updated assessment of safety margins.

FLOOD

The **Doel NPP** is not considered as being at risk of flooding due to the fact that the NPP is situated on an elevated platform and, secondly, the nearby river has an artificial embankment, which serves as a barrier for the site. Protection against external flooding of the Doel site is adequate also for floods with 10,000 years return period.

For Doel NPP the buildings which are flooded first or quickest in the case of wave overtopping or embankment failure were identified. For the most important SSCs and their physical location, it was checked whether there are any thresholds, plinths, etc. present to protect against the consequences of flooding in case of wave overtopping or embankment failure.

Safety margins for the Doel site are largely based on the topography of the site. Even in case of the river embankment failure near the site when a 20 cm water layer would be expected on the site at least one safety system level would remain unflooded (1st or 2nd level) for each unit. Organization and relevant safety equipment available shall provide an adequate response to an unexpected flood. A reinforcement of the embankment is considered for Doel.

DBF for **Tihange** site was originally derived as the highest historically recorded flood level of the surrounding river increased by 20% (i.e. 2200 m^3/s). Based on the flood in 1995 this value was revised to 1995 flood + 20% margin (i.e. 2615 m^3 /s). During the latest reassessment within the latest PSR, new DBF parameters have been derived using the probabilistic approach. Values with return periods of 10,000 years are taken as new design basis values. During the country visit it was reported, that for Tihange the new DBF value has been assessed to reach 3488 m^3/s as the best estimate value.

The results for safety margins of Tihange NPP show that there are weak margins for floods exceeding those with 400 years return period. Significant damage to equipment would be caused already by floods with return periods of 600 to 1,000 years, aggravating the consequences with increasing return periods (higher river flow rates). The value at which no safety related systems would be operational is clearly estimated. Related cliff edge effects are described and linked to respective river flow rates.

The conclusion of the peer review team is: Taking into account the relatively weak safety margins and the reconsideration of DBF values, it is recommended that the national regulator should focus on the implementation of all safety improvements proposed by the licensee, as well as those prescribed by the regulator.

EXTREME WEATHER

Conclusion of the peer review team on this specific area:

The design parameters for extreme weather conditions are mainly based on historic data and therefore on a return period in the order of 100 years. The derivation of design basis parameters with 10,000 years return periods is recommended to be considered. Attention should be dedicated also to extreme temperatures.

LOSS OF POWER AND LOSS OF UHS

The basic safety principles, such as defence in depth, redundancy of important safety equipment, their physical or geographical separation (however this principle is not fully implemented in the Doel 1/2 units), as well as their diversification, were applied from the design phase. A number of systems are shared by the Doel 1/2 twin reactors, while strict physical separation is applied between the redundant safety systems at all other units.

Doel 1/2 has some deficiencies because during the design phase several external and internal initiating events were omitted. These deficiencies have led to the implementation of specific safety systems, such as the emergency system building which is seismically qualified.

SEVERE ACCIDENT MANAGEMENT

The Long Term Operation (LTO) project for Doel 1/2 and Tihange 1 units also includes a design upgrade related to severe accident management – filtered containment venting. The pre-feasibility studies for the implementation of Containment Filtered Venting System for these units are on-going. This topic will also be assessed for all other units in the framework of the stress test project. During the country visit the peer reviewers recommended to consider the case of sub-atmospheric pressures in the containment after venting. The peer review recommends also the installation of Passive Autocatalytic Combiners in the SFP buildings to reduce the residual risk of hydrogen generation and accumulation.

CRITICAL COMMENT

The oldest reactors Doel 1/2 and Tihange 1 are now 36 and 37 years resp. in operation. This is a significantly longer operation time compared to the closed reactors in Germany. LTO would require an intensive investigation on material properties after such a long operation term. It is clear that these reactor units cannot be improved to achieve a safety level equivalent of modern NPPs.

4.4 NPP GUNDREMMINGEN, GERMANY

The Gundremmingen nuclear power plant (NPP) is equipped with two similar boiling water reactors (BWR) of identical design (German construction line 72). The power output of the reactors is relatively high: Gundremmingen B net capacity 1284 MWe, unit C 1288 MWe. Commercial operation started in 1984. The NPP is owned by RWE (75%) and EON (25%) and operated by Kernkraftwerk Gundremmingen GmbH (KGG).

The site is located at the River Danube about 90 km northwest of Munich. The shortest distance to Austria is about 100 km.

The German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) prepared the German stress tests report.

Preliminary notes: (1) The German stress test report contains little information and almost no assessments by the BMU. The "Regulatory body" in Germany is composed of authorities of the Federal Government (BMU) and authorities of the Länder governments. Licensing and supervision, inspection and enforcement as well as plant specific safety assessments and reviews of nuclear power plants are executed by the Länder.[22]

(2) Directly after the Fukushima accident, German nuclear power plants were subjected to a two-month safety review by the Reactor Safety Commission (RSK).[23] Furthermore an Ethics Commission “Secure Energy Supply” re-assessed the risks associated with the use of nuclear energy. On 6th August, an Amendment of the Atomic Energy Act entered into force reacting on the results of the RSK safety reviews and the Ethics Commission. This law terminated the operational licences for the seven oldest NPP and Krümmel NPP the same day. The licences for the other plants will expire on a step-by-step basis between 2015 and 2022 at the latest. Gundremmingen B has to stop power operation in 2017, Gundremmingen C in 2021 [BMU 2011].

WEAKNESSES THE GERMAN STRESS TESTS DESCRIBED [BMU 2011]

- The units are designed to withstand a design basis earthquake (DBE) with an intensity of IMSK=VII, corresponding to a PGA value of 0.1 g. However, the last re-evaluation of the site specific hazard is nearly twenty years old (1993) and thus completely outdated. According to results of the German Research Centre for Geosciences (Potsdam) higher DBE are possible at individual German NPP sites.
- The evaluation of the DBE is based on an exceedance probability (10^{-4}) that is one magnitude lower than those usually used in Germany and required in the revised German nuclear safety standard (to be published in 2012). Furthermore, it has to be assumed that conditions of low-power and shutdown operation are not considered.
- The DBE based on a state of the art reassessment would be probably higher, but safety margins regarding seismic loads are not proven sufficiently.
- The calculated design basis flood (DBF[24]) is 33 centimetres higher than the grade elevation. Thus, the premises will already be flooded in this case. Protection of safety relevant building should be assured, for example, with seals of cable penetration.
- Safety of the plant during the course of a longer-lasting flood as well as the protection of canals and buildings regarding the intrusion of water and the floating resistance in the case of

22 The BMU stated several times: The Länder authorities basically confirm the information and assessments provided by the licensees. This holds in particular for the information regarding the licensing basis. In general, the assessments of safety margins are plausible, but cannot be verified in line with the normal regulatory standards.

23 During the safety review, the operators had to shut down the nuclear power plants commissioned prior to 1980.

24 DBF is defined to be the flood with an exceedance probability of 10,000 per year.

a higher level flood is not assured.

- Because no exceptional loads are expected from precipitation events by the operator, the NPP has no specific design additional of the flood protection. Currently Germany does not have any specific requirements regarding extreme weather conditions in force.[25]
- In case of a total station black out (SBO), each unit has an additional residual heat removal system (AHRS) – but the functionality of the AHRS relies on one emergency diesel generator (DG). The DG is protected against site-specific earthquakes (DBE), which is, as mentioned above, probably lower than required. The AHRS could ensure the heat removal from the reactor coolant circuit and the spent fuel pool only for ten hours.
- The operator claims loss of primary ultimate heat sink and the AHRS to be extremely unlikely; and for this case accident management measures are available (depressurization of the reactor cooling system, water injection from different sources e. g. injection by mobile pumps, heat removal by filtered containment venting). But in this case the core cooling is only ensured for 15 minutes.
- In case of a total station black out and loss of heat sink, accident management measures have to ensure heat removal from the spent fuel pool. The decay heat can be removed by vaporisation of the spent fuel pool coolant and injection of water. The water can be provided from different sources with fire fighting water pumps. But this has to be installed within 12 hours.
- In general the feasibility and operability of accident management measures (e. g. injection possibilities for the cooling of fuel assemblies) under adverse conditions are not proven. A further development of the accident management concept under external hazard conditions is required for all German NPPs by BMU.
- Because of the installed filtered containment venting system, further Severe Accident Management Measures for restricting the activity release into the environment are not defined. The containment filtered venting system is shared by both units.
- No specific Severe Accident Management Measures for restricting releases or preventing hydrogen explosions after severe damage of spent fuel in the pools are available. Up to now it was assumed severe damage is excluded by prevent measures.
- Severe Accident Management Guides (SAMGs) to cover beyond design basis accident scenarios are not established. In Germany, the systematic implementation of SAMG is pending.

Weaknesses the German stress test Ignored

- The safety of the NPP relies on completely outdated rules and regulations (1977 – 1996). The new “Safety Requirements for Nuclear Power Plants” have been pending for years in Germany. Therefore accident prevention does not meet state of the art requirements. This is problematic in particular because of ageing related material defects.
- Up to 38 percent MOX (Mixed-oxide) fuel is used in the reactor core. MOX fuel characteristics during reactor operation are different to uranium fuel. In particular reactor control is more complicated; the effectiveness of control rods decreases.[26] MOX fuel elements have a lower melting point and a reduced thermal conductivity. In case of an accident thus a heating of the

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25 Research projects to evaluate the potential impact of extreme weather conditions on German NPPs are currently performed on behalf of the BMU. Depending on the results of these activities regulatory actions (new requirements and revision of safety standards) will be considered

26 In order to limit these effects, the number of MOX in the reactor core is limited.

reactor core proceeds faster and the core melts. Also accident consequences increase: MOX contains considerable more long-lived alpha emitters. Thus, the potential radiation exposure by inhalation and by contaminated food will increase significantly.

- In recent years many fuel defects were detected, the most recent in March 2012; the reasons are unknown.
- A crash of a Boeing 737 against the reactor building can cause a severe accident. In case of (1) a major destruction of the reactor building or (2) a damage of the control room by fire and debris combined with leakages in the cooling system, a severe accident could occur. This is the result of a study on behalf of the BMU. According to the Federal criminal police office, the probability of a terror attack against a nuclear power is low; however, a terror attack has to be taken into account [BMU 2002].
- The spent fuel pools are located inside the reactor building, but above and outside the containment (like in the Fukushima Daiichi NPP). In case of a severe accident, there is no barrier to the environment. Also the vulnerability against a terror attack is high. In total 3219 spent fuel assemblies could be stored in the spent fuel pool of each unit. These are four times more than in the reactor cores (784). Currently, the fuel pools are nearly full. During a severe accident about 10 percent of the caesium-137 inventory of the spent fuel pool would be released into the environment [ALVARAZ 2003]. This would be about 240,000 Terabecquerel (TBq) of caesium-137, i.e. considerable more than was released at the Chernobyl accident (about 100,000 TBq) and at the Fukushima accident (about 40,000 TBq) [FAIRLI 2006; STOHL 2011].

CONCLUSION

The evaluation of the design basis earthquake (DBE) was not appropriate; safety margin are not demonstrated. In case of a design basis flood (DBF) the site is flooded. Besides earthquake and flooding, terror attacks could cause a severe accident. A large amount of radioactive release has to be feared not only from the reactor core but also from the spent fuel pool that is located outside the containment. The used MOX fuel would increase the potential consequences considerable.

The prevention of a severe accident relies on accident management measures that take not into account accident conditions, so the feasibility is not proved. However, a severe accident is excluded by the operator; mitigation measures are almost not defined.

In case of a severe accident not only the operator is unable to cope, but also the authorities. According to a recent study of the Federal Office of Radiation Protection (BfS), the emergency response would fail in case of a severe accident. Large regions would be contaminated and entire cities would have to be evacuated. Such evacuation plans do not exist [SPIEGEL 2012].

KEY RESULTS OF THE PEER REVIEW

EARTHQUAKE

For most operating plants of the design basis earthquake (DBE) was re-assessed during the siting process for the interim spent fuel storage (2002). However for Gundremmingen the last seismic re-evaluation was in 1993. No seismic PSA has been performed.

The regulator intends to extend the requirement regarding seismic monitoring systems to all plants, but no schedule for implementation was provided.

The margins as well as the cliff edge effects for seismic events have not been determined. Nevertheless, national assessment concludes that safety margins are available due to both, the conservative hazard assessment methods and robustness of the design itself. The peer-review process identified that the design concept where the plants need to be resistant against an aircraft crash is a strong safety feature, because it offers additional level of protection for other external hazards including seismic events.

The peer-review team was not informed about further enhancements. The reviewers stated that this might change as a result of the assessments following the discussions of the German Reactor Safety Commission (RSK) and GRS Information notice, which are ongoing.

FLOOD

The level of a flood up to an exceedance probability of 100 per year is based on the historical records at the sites. The extension to exceedance probabilities of 10,000 per year, which is required for the design basis flood (DBF), is obtained by extrapolating the historical records.

For flooding beyond the DBF, the margins have been assessed, by estimating the difference between the water level of the DBF and the height of openings that could lead to flooding of safety related buildings. Sufficient margins have been determined for safety related buildings and structures. No cliff edge effects have been identified.

Positive safety features include the requirement that all upstream dam failures need to be taken into account when estimating the DBF. Permanent measures have to be in place for such events.

An on-going RSK work program reviews the protection of canals and buildings with respect to the intrusion of water and the buoyancy effect in case of a higher-level flood with a postulated flooding of the plant site. This is important for Gundremmingen, because the site is flooded in case of DBF.

It is noted in the peer review report, that the accessibility of plants during floods has been evaluated. There is also an on-going review of the accessibility of buildings in case of long-duration flooding on the site.

EXTREME WEATHER

The design basis for extreme weather is based on national engineering standards rather than on specific nuclear related requirements. Specific assessments of the extreme weather conditions beyond those that are the design basis have not been undertaken. This is due to the fact that such conditions are not believed to be possible in Germany.

The margins to the extreme weather beyond design basis have not been estimated within the process. In general, the margins are likely to exist mainly due to the design of German NPPs covering also events such as explosions and aircraft crashes that are enveloping the conditions of extreme weather events (e.g. wind and tornadoes). BMU has initiated research activities in the area of extreme weather, which may result in new requirements.

The peer review team recommends considering the assessment of margins with respect to extreme weather conditions exceeding the design basis (e.g. by extending the scope of future PSRs).

LOSS OF POWER AND LOSS OF UHS

One additional emergency diesel generator (EDG) for each unit is available within the additional residual heat removal system to cope with loss of off-site power and loss of the ordinary back-up AC power sources situation. This EDG is protected against earthquake and flooding. The residual heat removal from the spent fuel pool, which is outside the containment, occurs by evaporation. The evaporation losses can be made up only by mobile pump(s).

Both operators and the regulator considered the use of Accident Management (AM) measures to cope in the most challenging scenarios and plants modes, including the ENSREG stress tests scenarios. These AM measures are described in the plant specific accident management manuals and include complete timelines with the grace period and time needed to implement the measures. The peer review team accepted the conclusions of German regulator that no weaknesses have been found related to this topic of the stress tests. Nonetheless, the RSK and GRS have proposed some improvements they are, among others, also concerning NPP Gundremmingen:

- SBO coverage for at least 10 hours,
- Additional emergency power generator available within 10 hours,
- Increasing of spent fuel pools (SFPs) cooling capabilities,
- Multi-unit events analysis.

Severe Accident Management

Procedures to describe separate AM measures following a core melt situation are included in the Accident Management Manual. In the case of fuel damage inside the reactor pressure vessel (RPV) flooding the drywell of the containment is foreseen, but a melt-through of the RPV cannot be prevented for all situations.

It is assumed that due to long grace periods and the adoption of appropriate measures to ensure cooling of the fuel elements in the pool, damage to fuel elements can be excluded severe accident conditions. Mobile equipment will be used to avoid fuel uncover.

Recently, the RSK started a discussion on the implemented severe accident management measures in Germany. This resulted in the publication of new and extended recommendation in October 2010. However, during the peer review process, the following weak points / deficiencies have been identified:

- The accident-proof instrumentation is battery secured for only 2 to 3 hours,
- Consequences of fuel meltdown in the spent fuel pool have not been considered,
- No low power or shut down SAMGs exist,
- No long-term SAM procedures have been developed so far.

Based on the results of the plant-specific safety review of German NPPs in the light of the events in Fukushima, the RSK concluded that comprehensive further efforts are necessary, among others concerning Gundremmingen:

- Development of the AM concept under external hazard conditions,
- Long-term energy supply (in particular to mobile systems),
- Long-term heat removal from reactor core and spent fuel pool,
- Long-term heat removal from the wetwell,
- Use of AM measures under long-term SBO conditions
- Development of AM measures to protect the building structure surrounding the spent fuel pool, against hydrogen combustions or to prevent them.

<u>Conclusion of the peer review team:</u> It appears that severe accident management concept in Germany is focused on the use of preventive measures to be followed by mitigative measures such as filtered containment venting systems. The AM measures and the related plants internal organization appear well structured, adequate to cope with accidents with different levels of severity. Although guidance exists in the Accident Management Manuals, SAMGs have been not developed. The development of generic SAMGs is scheduled for completion by the end of 2012, following time plant specific adjustments will be made. The reviewer expects that SAMGs will be available.

The reviewer stated that the operability of instrumentation essential to severe accident management should be systematically evaluated for severe accident conditions. This should be achieved in the course of SAMG development. Improvements in this respect might be necessary.

CRITICAL COMMENT

The protection against earthquake or flooding is probably not sufficient. However, the prevention of severe accidents relies on outdated accident management measures that do neither take into account external hazard conditions nor the need of long-term heat removal. Consequences of fuel meltdown in the spent fuel pool have not been considered, thus mitigation measures are not defined. In case of a severe accident a large radioactive release is possible, particularly from the spent fuel pool that is located outside the containment and stores also MOX elements.

4.5 NPP KRŠKO, SLOVENIA

The Slovenian stress test report was published by the Slovenian nuclear safety administration (SNSA); the authority used Krško operator´s progress report, added its own executive summary and conclusions and sent it as the National Slovenian Report to the European Commission.

Krško NPP is located in the industrial zone on the north-western brim of an alluvial valley surrounded by hills from 200 m to 700 m, east-southeast of the town Krško on the left bank of the Sava River. The average altitude of surrounding area is about 154.5 meters above Sea level. The plant itself is located at 155.20 m on a plain. The surrounding area of the site is sparsely populated. Except for a few small towns, Krško included, the area is mainly rural. 27,700 inhabitants live in the 10 km radius around the plant. However, in a circle of 25 km, 55,000 live in Slovenia and 147,700 people in Croatia.

The plant is in operation since 1983; NEK is a 2-loop Westinghouse PWR, with an electric output of 700 MWe. Both original steam generators have been replaced with Siemens SG in 1999.

The Krško NPP is located in a seismically active region. For siting and design of Krško NPP the US NRC nuclear regulation and standards were applied. In line with these standards the peak ground acceleration (PGA) of 0.3 g was used for Safe Shutdown Earthquake (SSE).

Seismic hazard assessments in 1994 and 2004 led to raising the PGA values for the SSE: In 1994 to PGA= 0.42g and in 2004 to PGA= 0.56g. As part of the seismic PSA investigations detailed walk downs were performed to identify the seismic vulnerabilities. Due to the new 2004 seismic hazard assessment frequency has increased substantially. A new screening target has been set with High Confidence of Low Probability of Failure (HCLPF) value of about 1.0 g. The first periodic safety review represented a significant review process, where seismic issues were identified, evaluated, and new actions were set up for plant seismic improvements. One of the most important improvements will be the installation of a third seismically classified emergency diesel generator, which will be completed in 2012.

Seismic margins with weak points and cliff edge effects are evaluated by means of identifying success paths from the safety analysis report, safety studies, and other investigations. Evaluation of seismic core damage margin, seismic margin for containment and spent fuel pool (SFP) integrity and cliff edge effects is presented.

The spent fuel storage pool is located in a separate building. This fuel handling building is an integral part of the plant with a reinforced concrete structure. It is designed in accordance with the seismic and other criteria for safety structures. The spent fuel pool within the fuel handling building is lined with stainless steel to prevent leakage of water.

The SFP integrity is ensured for PGA up to approximately 0.9g. For earthquakes exceeding the PGA of 0.9g, structural failures of SFP and pipes cannot be excluded and fuel uncovers are considered as likely to occur.

Seismic events with PGA over 0.8 g were assessed as very rare at the NEK site, with the return period of the order of 50,000 years or more.

The Krško NPP site is located in an area prone to flooding. The design basis flood for Krško NPP is the 10,000 year flood. The Krško NPP is located in the Krško-Brežice Basin, on the left bank of the Sava River. The right bank of the Sava River above the Krško NPP and the left bank of the Sava River below the Krško NPP are extensive inundation areas that are flooded in events with high river flow.

The operators report concludes that the plant design with its additional flood protection dikes constitutes adequate protection against design basis flood.

WEAKNESSES THE SLOVENIAN STRESS TEST DESCRIBED [SNSA 2011]

- Earthquakes of a PGA ranging above 0.8g or higher are a hazard for the reactor core, mechanical damage could disturb the core geometry and thus the insertion of the control rods. Partial core melt is not excluded in such a situation. In this PGA range also containment spray and low pressure emergency cooling would be unavailable. Late radioactive releases cannot be excluded. Severe accident management measures are foreseen to mitigate the release.
- Seismic events resulting in early radioactivity releases to the environment would be likely to occur when the PGA significantly exceeds 1g.
- Earthquakes which would endanger the spent fuel pool are exceeding PGA= 0.3g. Normal cooling system cannot be expected as being operable in this situation, after 76 hours fuel will be uncovered. In the range of PGA= 0.3 -0.9g parts of falling objects could damage fuel assemblies, but it is not expected that the fuel would become uncoolable in the pool. If the water level in the pool drops to the top of the fuel assemblies, the loss of shielding makes the access of the NPP personnel to the SFP very problematical.
- If the SFP sprinklers are ineffective, overheating of the fuel is expected. When the water level drops to 30% below the assemblies' height, degradation of the assemblies starts and results in exothermic reaction of zirconium oxidation and hydrogen production.
- Regarding flood the cliff edge effect occurs when a flood with a river flow 2.3 times larger than the design basis flood and 1.7 times larger than the assumed maximum flood hit the plant plain. Such a flood is assumed to have a return period of 1 million years.
- The NEK control room has been improved. However a new backup control room will be constructed. Since NEK has only one water intake construction, an alternative seismically qualified UHS independent from the river Sava is planned.

WEAKNESSES THE SLOVENIAN STRESS TEST IGNORED

- A strong earthquake (PGA > 0.9g) causes fuel damage in the reactor core and in the spent fuel pool more or less simultaneously. The report assesses those two events separately. The report does not inform about the availability of sufficient equipment and personnel to manage SFP cooling and to prevent or mitigate core damage when a mechanical impact of the earthquake hinders the control rods falling down properly.
- The challenge of the maximum probable flood is loss of offsite power. The Krško NPP is in the process of upgrading its existing flood protection by raising the flood protection dikes upstream of the Krško NPP along the left bank of the Sava river and the Potočnica creek. After implementation of that modification the Krško NPP would not become isolated on an island even during the probable maximum flood. An earthquake could cause a flood wave followed by a dike break and flood the site. To rely only on dikes for protection of the NPP is irresponsible.
- Common cause failures caused by external events are mentioned, but a systematic assessment is missing.

CONCLUSION

The Krško site is not suited as a site for an NPP; main hazards for the plant consist of earthquake and flood. Cliff edge effects caused by beyond design earthquake, flood or combination of both cannot be excluded solely arguing the low probability of occurrence. Krško was in operation for 30 years, currently the permitting procedure for lifetime extension is being prepared; the permit should be issued next year. Material fatigue due to ageing the National Report does not consider as a factor in the accident sequences. The same applies for common cause failures.

KEY RESULTS OF THE PEER REVIEW

EARTHQUAKE

Recent PSHAs result in a PGA of 0.56g which is nearly twice the original SSE accelerations of 0.3g. This seems to be plausible, considering that Krško NPP is located in a seismically active region.

Peer review conclusion: Evidence showed that the Krško NPP can accommodate the new value of PGA of 0.56g or 0.6g. The design basis has not been updated. However, the increased seismic hazard will be considered for future projects, e.g. the third DG building.

The report refers to several active faults, which were identified in the immediate region of Krško. Analyses included paleoseismic investigations, concluding that no relevant paleoseismic tracks were found. According to information provided in the course of the country visit, soil liquefaction might start at 0.8g.

Recommendation of the peer review: Estimation of seismic safety margins is based on the identification of success paths and on the seismic probabilistic safety assessment (SPSA). The margins are found to be substantial with regard to the DBE. However, there seem to be decreased margins with regard to the updated seismic hazard assessment. **The peer review recommended that the regulator should consider requesting an update of the seismic design basis for future design modifications and consequently the associated PSA model.**

FLOOD

The Krško NPP is assessed to be robust for the defined flood events (design basis flood, probable maximum flood, extreme flood). An additional flood-protection measure (increasing the dike height upstream from the plant) is in progress.

Safety margins were evaluated in an advanced manner. Methods and input data are specified in a limited manner in the SI-NR, but have been reviewed in the course of the country visit.

EXTREME WEATHER

The SI-NR provides only limited information. The design basis and the safety margins are presented in a simplified way, mainly providing information on the meteorological history of the site. Based on the first PSR, the SI-NR concludes that the design of the plant against extreme weather conditions is sufficient and no action is needed. During the peer review sufficient information was provided and is documented in the updated safety analysis report.

Loss of Power and Loss of UHS

According to the information provided in the SI-NR, the present accident management organisation appears to be well structured and adequate to cope with different levels of severity in the event of an accident, including severe core damage. WENRA safety reference levels have been implemented in the Slovenian regulatory process.

Moreover, the Slovenian regulatory assessment also applies selected USNRC regulations, standards, guides and good practices. An especially noteworthy characteristic of the Slovenian SAM organization is the validation of the SAMG using a full-scope simulator.

Several provisions are already in place to support SAM with the use of mobile equipment. It is important that upgrading measures identified to improve SAM capabilities (e.g. installation of PARs, filtered venting, new emergency control room, third engineered safety features train) are implemented as planned.

SEVERE ACCIDENT MANAGEMENT

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The current plan of the licensee envisages a plant life extension beyond 2023 for an additional 20 years. The relevant submission is expected to be approved by the regulator by 2013. This life extension beyond 2023 will be accompanied by two additional cycles of PSR.

For the life time extension a large number of improvements is foreseen. The action plan, expected to be implemented in 2016 comprises the following measures:

- additional seismic strengthening of the DG3 and MD3 bus
- additional seismically-qualified (2×SSE) SI and AFW pumps
- alternative seismically-qualified (2×SSE) UHS
- installation of a special safe-shutdown control room
- alternative seismically-qualified (2×SSE) means of cooling the SFP.
- Realization of an additional third train of engineered safety features comprising the already mentioned third DG, MD bus, a high-pressure safety-injection pump and a feed-water pump. This train will be located in the already constructed building protected against external events.
- Alternative UHS,
- Installation of a special emergency control room, to be located in the above mentioned new building,
- A new technical support centre with enhanced habitability requirements.
- Alternative means of cooling the SFP and of decay-heat removal,
- Filtered containment venting,
- Passive autocatalytic re-combiners for hydrogen control in the containment.

CRITICAL COMMENT

In 2013 the NPP Krško will have operated 30 years; there is a lifetime extension planned for 20 years or even more. A lot of safety improvements are needed to guarantee safe operation during the extended lifetime. Since the site is not suited for an NPP, Slovenia should consider if the investment in safety improvements at Krško would not be better used for alternative clean energy as wind, solar and small water power.

4.6 NPP Mochovce, Slovak Republic

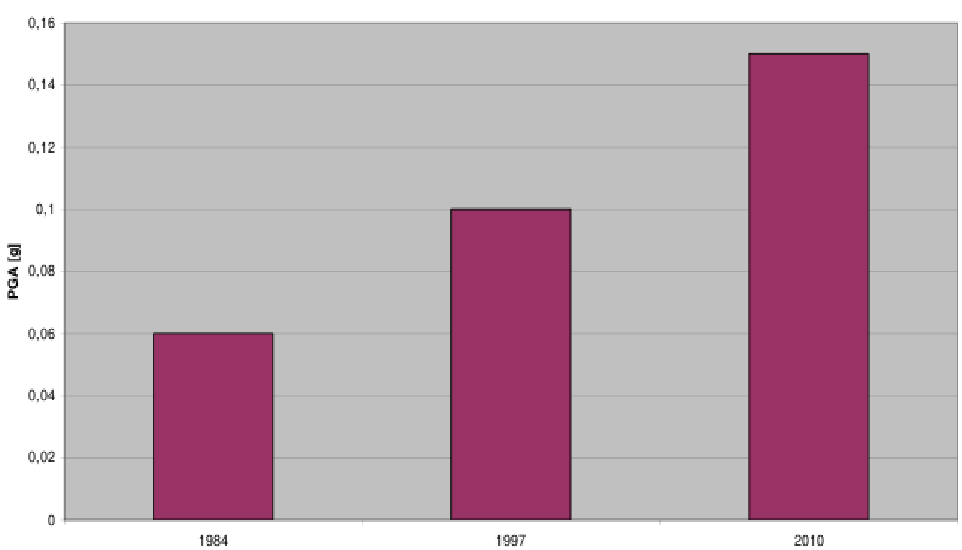
NPP Mochovce is situated 90 km north-east of Bratislava. The Nuclear regulatory authority (UJD) prepared the Slovak stress test report (UJD 2011), based on the operator report results.

NPP Mochovce consists of two pressurized water reactor (PWR) units VVER 440/V213, operating since 1998 and 1999 respectively, and two units VVER440/V213 under construction. EMO 1/2 has a thermal power of 1470MW and an electric output of 470MWe. EMO 3 is scheduled to start in 2013, unit 4 to follow in 2014.

The VVER 440/V213 is a PWR of Soviet design with 6 loops. Significant time margins are available for core cooling even during loss of power supply and loss of ultimate heat sink. This is possible because of the large thermal inertia due to low power and comparably large amount of water both in the primary and the secondary system, as well as a large volume of water inside in the pressure suppression system [UJD 2011]. The VVER 440/V213 is not equipped with a full pressure containment, which is a common feature of most pressurized water reactors.

The confinement of VVER 440/V213 reactors consists of compartments, which enclose the essential primary circuit components: steam generator, pipelines, pumps, shut off valves and reactor pressure vessel. The VVER 440/V213 confinement on its own does not prevent radioactive steam from leaking into the environment, the steam needs to condense in a special pressure relief system – the Bubbler Condenser to prevent overpressure in the confinement system. A failure of the relief system can cause the confinement to burst at its weakest point and release radioactive material into the environment. In recent years studies on the behaviour during severe accidents were commenced. The results are to be used for the development and improvement of the SAMG - Severe Accident Management Guidelines. Safety analyses showed that the confinement and in particular the Bubbler Condenser have very low or even no safety margins under certain conditions and for certain components. In case of EMO 3/4 improvements are planned (e.g. installing recombiners for hydrogen removal). [WENISCH et al 2009] The spent fuel pool (SFP) is located outside the containment barrier in the reactor hall. During reactor operation, the SFP is covered.

Earthquake is a major hazard for the Slovak NPPs, also for NPP Mochovce:



The original design values were determined as being 6.5 - 7 MSK 64, PGA = 0.06 g); the result of the re-evaluations was the new value for SL2, which was PGA=0.143. UJD SR decided to increase the design-basis earthquake for Mochovce site to include a safety margin and required PGA = 0.15g. This value is used for completion of EMO 3/4 and the upgrading of EMO1/2 currently being undertaken. (UJD 2011)

Comprehensive upgrades were performed, and the seismic monitoring system was renewed. However, it is important to understand that upgrading measures do not necessarily establish the condition, which an adequate original design basis would guarantee. [WENISCH et al 2009]

UJD considers flooding as not relevant; no protection against external sources of the site flooding is needed because of distance to the Hron River. The existing dams in the River Hron do not pose a hazard to NPP Mochovce. Earthquake is the main hazard for Slovak NPPs.

Weaknesses the Slovak Stress Test Described [UJD 2011]

- If the Velke Kozmalovce water reservoir dam fails, the service water source will be endangered due to the flooding of the service water pumping station providing the service water to the Mochovce site. In case of complete loss of the service water supply, the water inventory in the cooling tower pools and in cooling water channels will be used. At the Mochovce site, an additional water reservoir is available and can be used. An alternative water source is missing.
- Internal floods could be caused by an earthquake. Sources of internal flooding are:
- Large break of the feed-water pipeline in the lengthwise electrical building
- Large break of circulating cooling water system in the turbine hall,
- Large break of feed-water pipeline and condensate pipeline in the turbine hall.
- All buildings containing electrical equipment could be affected by an internal flooding, in case of a pipe or tank rupture.
- All VVER 440/213 reactors are implementing severe accident management measures. These measures concern hydrogen removal from the confinement and from spent fuel pools, as well as cooling of the reactor pressure vessel by flooding the reactor shaft with water from the Bubbler Condenser. Thus the overheated core is cooled and retained in the vessel, even if the core is melting.
- Vulnerability of the equipment needed for plant shutdown and cool-down is verified for the design basis earthquake. However, there is no analysis of the resilience of this equipment in beyond design earthquake conditions available. Therefore cliff edge effects cannot be excluded.
- Reliability of AC emergency power supply under station blackout (SBO) conditions and diesel generator (DG) for charging batteries is not sufficient, modifications of the pump of the borated coolant system should be enable their use during SBO.
- Several connections for mobile pumps are not installed yet, which would provide cooling water for specific systems (spent fuel pool, emergency feed water, steam generator feed water)
- Resilience of reactor coolant pump seals at long term cooling failure due to loss of ultimate heat sink (UHS) is not guaranteed for more than 24 hours.

Weaknesses the Slovak Stress Test Ignored

- The Slovak stress test evaluates only the minimum of natural events and other sequences causing the loss of the ultimate heat sink (UHS) and station black out (SBO).
- The reactor buildings do not provide sufficient protection to the plant against external impacts like airplane crashes or explosions. Indirect consequences of man induced events are not assessed. Roofs of buildings where heavy machinery is used (fuel loading machine reactor

hall, crane in the turbine hall) are not necessarily qualified for beyond design impacts of natural and man-made events.

- Several problems occurring during accidents are supposed to be solved by the fire brigade. The report does not cover the question whether the fire brigade´s buildings and equipment are resilient against a Beyond Design Earthquake, or how the fire brigade handles radiation on site.
- An earthquake could cause fire to break out however the stress test does not examine this issue. This is particularly dangerous for the electric systems, when cable fire can easily cause failure of several necessary systems. After this failure, events at the NPP can get out of control and lead to a core melt accident with high level radioactivity releases. Additional fire-fighting equipment was installed, however, this does not reach the same safety level as structural separation of those redundant systems would have if taken into account already during plant design phase.
- Habitability of the main control room in case of an accident in the spent fuel storage was not assessed.
- Hydrogen concentration due to a severe accident in the spent fuel pool was not assessed.

CONCLUSION

Obviously the Mochovce nuclear power plant has a number of safety deficits stemming from the original design. The idea is to remedy those deficits by implementing upgrades. However, this goal cannot be achieved. On the one hand, there are safety deficits, which cannot be upgraded, e.g. thickness of the confinement walls. Taking into account the existing risk of terrorism, it is irresponsible to operate a nuclear power plant with such a high vulnerability to external attacks.

On the other hand this high number of upgrading measures significantly increases the already enormous complexity of the nuclear power plant. The nuclear regulator UJD points out, that the planned upgrading measures at Mochovce NPP constitute a fairly high amount of modifications, additionally mutual interaction of many modifications need to be taken into account. In emergency cases this can cause necessary safety systems to be unavailable or only partially available due to failed activation. Moreover it is extremely difficult for the operating team to intervene in such a complex plant.

It has to be expected, that even more safety deficits of the old and out-dated reactor type VVER 440/V213 will be recognized in the future and require further safety upgrades. Experience shows that safety deficits are not removed immediately, but months or years later. [WENISCH et al 2009] Like all VVER 440/213 reactors, the Mochovce NPP is implementing Severe Accident Management measures and guidelines. These measures concern hydrogen removal from the confinement and from spent fuel pool, as well as cooling of the reactor pressure vessel by flooding the reactor shaft with water from the Bubbler Condenser. Thus the overheated core is cooled and shall be retained in the vessel even it had started to melt. These measures are an attempt to achieve safety objectives required by WENRA. **Safety objective 3** is an ambitious requirement, to control core melt accidents and reduce radioactive emissions to the environment. However it must be mentioned, that if the vessel fails, the reactor cavity door[27] is not designed to retain the molten core. In this case the release of a large amount of radioactive material to the environment cannot be prevented.

27 Door is for maintenance jobs

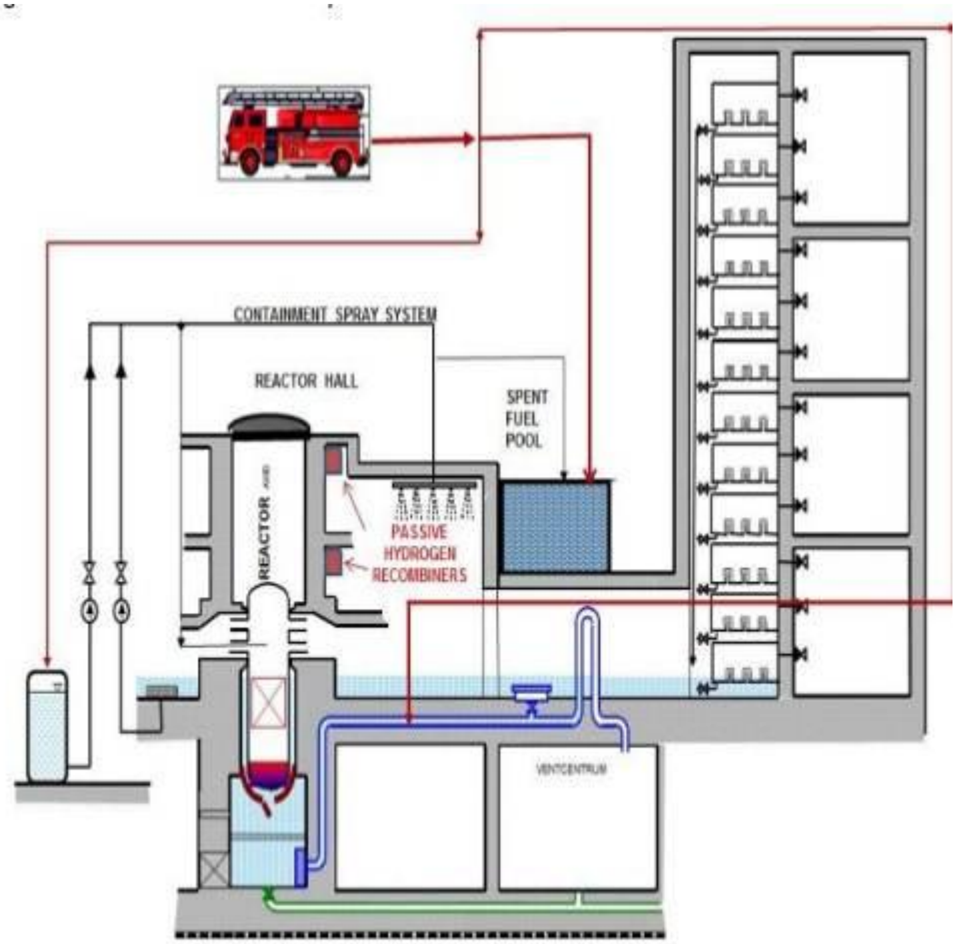


Figure 2: *Severe Accident Management: Retention of corium in the vessel; ex-vessel cooling by flooding of the reactor cavity. (source: National report Slovak Republic)*

KEY RESULTS OF THE PEER REVIEW

EARTHQUAKE

In relation to the DBE levels, the conservative assessments – as based on the latest scientific information in the field – yielded in setting relatively high DBE PGA values for both sites. The corresponding reinforcements are already fully implemented at the Bohunice site and are to be implemented before the next PSR at the Mochovce site.

During the visit to the Mochovce site, the review team identified some cases where some components of no primary safety feature potentially may have indirect influence on some safety functions. The review team recommended the licensee and the regulator to review all such cases shall.

The original DBE assessments for Slovak NPPs have been questioned and subsequently re-evaluated in several steps in accordance with the development of methodologies, data and safety requirements.

All NPPs comply with the minimum DBE level defined by the IAEA standards. Beyond design basis capabilities of the plants are discussed in the report but particular safety margins to cliff-edges are not quantified because the refined analysis of the vulnerability of the key equipment were not available at that time. However, the conservative approach to design of particular components, which was used during calculations allowed to demonstrate additional margins. Studies have been completed for civil structures to prove this. The robustness of the plant against earthquakes has been significantly increased recently and is considered adequate in accordance with the current requirements. In addition there are some safety upgrading measures envisaged.

The peer review recommended to the Slovak regulator to consider monitoring the implementation of measures for a quantification of margins. To assure a timely completion of the measures for seismic resistance of the relevant SSCs of EMO 1&2 for the newly defined Review Level Earthquake (PGA of 0.15g) the team recommended to the regulator to consider prioritization of the seismic upgrading measures, e. g. in respect to the fire brigade building.

FLOOD

The flooding against which the plants are designed is defined for each plant. All relevant sources of flooding were considered. Strong rainfalls were defined as the only potential sources of flooding. The approach used for the assessment of the Design Basis Flood (DBF) appears to be reasonable in compliance with the international standards. The protection against the DBF is adequate mainly due to relatively high altitude difference between the sites and nearby rivers. Large margins were proven at Mochovce site.

EXTREME WEATHER

Due to the lack of information on resistance of SSCs to the beyond design weather conditions in plant documentation, engineering judgement is applied to estimate the plant response. Postulated external events and their characteristics caused by extreme meteorological conditions are considered complete and specified in line with international practice. Originally postulated events were extended to include tornadoes. Specifications of the events are estimated for up to 10,000 year return period. Safety margins between the threats and design values were performed: for extreme wind and snow a 20% margin was confirmed. For extreme temperatures – proved margins vary for different parameters between 5% and 20%.

For Mochovce NPP more detailed confirmation is needed regarding the detailed meteorological study and assessment of the impacts using state of the art methods. Precise quantitative specification on the cliff-edges is therefore not available at present.

Evaluations of the effects of extreme meteorological conditions in the report are mostly qualitative, based on operating experience and on engineering judgment. Evidence has been provided on the existence of additional safety margins but their quantification is pending. Due to the lack of information in the plant documentation on resistance of SSCs to the beyond design weather conditions, engineering judgment is applied to estimate the plant response and assess the safety margins. Further work is required in this respect.

The peer review recommended to the Slovak regulator to consider monitoring the implementation of the measures for strengthening the level of protection of the plants against extreme weather conditions.

LOSS OF POWER AND LOSS OF UHS

The National Report indicates that a comprehensive complementary safety analysis has been performed by the licensee for all NPPs. Measures are proposed to increase safety capabilities in case of LOOP, SBO and loss of UHS, and have been assessed by the National Regulator including own calculations and modelling.

As a reaction to the Fukushima event, the utilities and the National Regulator carried out walk-downs and inspections on structures, systems and components relevant to LOOP, SBO and loss of UHS. As a result of these special reviews and assessments, measures to improve the safety of the plants have been planned and prioritized. Special tests also were performed (for example: test of feeding steam generators using the fire truck high pressure pump, test of water supply to SFP from bubble condenser trays) to validate the possibilities to apply dedicated measures. Some measure designed to increase robustness were implemented immediately (e.g. the procurement of fire truck containing high- pressure pump).

In the peer review there is long list of measures identified to increase robustness against beyond design basis events, these comprise among others:

- increase reliability of emergency power supply by new air-cooled diesel generator for SAM
- provide .mobile diesel generator for charging batteries
- provide mobile high pressure feed-water pump for each unit for injection into steam generators
- provide for power supply of containment bubble condenser trays drainage valves and hydro
- accumulator isolation valves from the vital power supply system (Mochovce NPP)
- consider the possibility to control selected valves without vital power supply by means of small
- portable motor
- provide for mobile pumps for essential service cooling water make-up from circulating water

SEVERE ACCIDENT MANAGEMENT

Symptom based SAMG have been developed between 2002 and 2004. As explained below, this development and also other projects lead to identify the need of some hardware modifications to practically eliminate situations of large releases. These modifications are part of the on-going SAM implementation program which has been approved by the regulatory body.

There are large ongoing upgrading programs in the area of accident management, the following table summarizes the current status of implementation:

Title of subproject SAM	Mochovce 1/2
Reactor Cavity Flooding	2011/2012
Primary circuit depressurization	2013/2014
Containment hydrogen management	2012/2013
Containment vacuum breaker	2014/2015
Alternative coolant system	2014/2015
Alternative power supply system	2013/2014
I&C Post accident monitoring system	2014/2015
Containment long term heat removal	2015

<u>Peer review conclusions</u>: The national report and the information gathered during the country visit

provide satisfactory information on the severe accident management program developed for the Slovak NPPs. The peer review pointed out that the severe accident requirements are included in the Slovak regulations and verification of compliance is part of the PSR process.

The hardware modifications proposed by the utility are based on a consistent set of analysis (deterministic and probabilistic studies) and correlated to advanced safety concepts for severe accident management. It is however important that the SAM modification will be implemented according to the proposed schedule. The peer review suggested locating the special equipment for SAM in dedicated locations qualified against external hazards.

With the strategy developed for the Slovak VVER 440, Reactor Pressure Vessel (RPV) failure is considered very unlikely after the modifications for molten core in-vessel retention. Nevertheless, investigation to limit the consequences in case of RPV failure could be considered in further steps. The verification of tightness of all containment penetrations (e. g closure above the vessel, above the SG) in severe accident conditions should be further examined (resistance of seal in particular).

The strategy of long term management of containment pressure without any containment venting system should lead to further verification to check the real feasibility of long term containment heat removal in severe accident conditions.

CRITICAL COMMENT

It seems that the operators of VVER 440 /213 reactors are planning a range of large improvement. For a plant which was originally designed for 30 years of operation and half of its design live is over such a huge improvement program is an expensive investment and thus likely will lead to life time extension. In any case it should be proved that the planned improvement program achieves a substantial mitigation of the severe accident hazard.

4.7 NPP MUEHLEBERG, SWITZERLAND

The Swiss National Report reviewing the operators' reports was prepared by the Swiss Federal Nuclear Safety Inspectorate (ENSI).

NPP Muehleberg has a thermal power of 1097 MW and an electric output of 373 MW. The NPP is in operation since 1972. Muehleberg is a General Electric Boiling Water Reactor (BWR/4) with a Mark I containment. Mark I double containment is a steel construction in the form of a light bulb with a concrete enclosure. Muehleberg has a filtered venting system to relief the steam pressure in the containment in case of a loss of cooling accident.

***Figure 3**: KKM Mark 1 Containment [Source: KUEHNI 2011]*

Muehleberg NPP is located at the River Aare downstream and West of the city of Bern. The Wohlensee dam and the hydroelectric power plant are situated upstream of the Muehleberg NPP at a distance of 1 km.

Periodic Safety Reviews (PSR) of Swiss NPPs, which are required by law every 10 years have resulted in numerous refurbishment and back-fitting measures in the past. Major back-fitting measures that are particularly important in connection with the EU stress test are listed below

[ENSI 2011]:

- Since the 1980s, systems to prevent **hydrogen explosions** have been gradually back-fitted in all nuclear power plants, this includes passive autocatalytic re-combiners, H2 mixing systems, H2 ignition and N2 systems to inert the primary containment,
- Since the 1990s, all nuclear power plants were equipped with a **filtered containment venting** system to mitigate the consequences of a severe accident, with independent **special emergency systems** in a separate, bunkered building. These systems serve in particular to provide protection against natural and man-made external events; Muehleberg NPP was back-fitted with such a **bunkered emergency building.**
- **Severe Accident Management Guidance** (SAMG) has been developed over the past ten years on the basis of the Level 2.

Probabilistic Safety Analyses (PSA) were performed for all Swiss NPPs, both for power and non-power operation. The safety systems used in the Swiss nuclear power plants were subdivided into three "safety trains", where each can bring the plants into a safe shutdown state in case of accidents [ENSI 2011]:

Safety train 1: It consists of the conventional safety systems which are used to control accidents due to internal events (such as LOCA – loss of coolant accidents – or internal flooding) and, depending on the original design concept of the nuclear power plant, external events related to natural causes, such as earthquakes and external flooding.

The conventional safety systems of the old Muehleberg NPP are not entirely designed to withstand earthquakes. The old plant was not built to comply consistently with the design principles of functional independence, physical separation and level of automation, though the single failure criterion by means of redundant system trains is fulfilled.

Safety train 2: The special emergency systems ("Notstandsysteme") constitute another safety train which is primarily intended to control accidents initiated by external events. Special design features of the special emergency systems include their functional independence and physical separation from the conventional safety systems, and self-sufficient operation of at least 10 hours without manual intervention. In all Swiss NPPs, control and monitoring of the special emergency systems is handled via an emergency control room ("Notsteuerstelle"), which is also specially protected and both physically and functionally separated from the main control room. The special emergency core cooling systems of the Boiling Water Reactor (BWR) Muehleberg do not have a high-pressure injection, but they a venting and low-pressure injection as redundant functions.

Safety train 3: The preventive accident management measures implemented in all nuclear power plants constitute the third safety train. This train consists exclusively of manual measures that are to be implemented locally by operating staff; they are stipulated in specific emergency procedures, are ordered by the emergency staff and are carried out with the deployment of either permanent built-in or mobile equipment. At Muehleberg NPP it was determined that a blockage of the special emergency intake structure by bedload during a flood cannot be excluded. The operator decided to carry out the following back-fitting measures: three intake pipes (periscopic pipes) were installed in two physically separate positions on the special emergency intake structure, so that an adequate inflow of cooling water was guaranteed. If the emergency intake structure is blocked, as an alternative a fixed injection point for mobile fire brigade pumps was installed where cooling water can be fed into the emergency cooling system.

Weaknesses the Swiss Stress Test Described [ENSI 2011]

The periodic reviews lead to continuous safety improvement. As a result, only a few deficiencies are left over to be improved in the next years; the deficiencies of NPP Muehleberg are listed below:

- a diversified, separated heat sink is lacking
- the instrumentation for measuring the filling level and temperature at the spent fuel pool is not accident-proof, an alternative cooling system for the SFP is missing as well as a physically separated additional feed option for the SFP for use in accident management.
- The Muehleberg NPP has no installations to prevent hydrogen explosions in the reactor building. If the used fuel in the SFP would be overheated, the chemical reaction of the fuel cladding with steam would result in hydrogen released in the upper part of the reactor building. If the ventilation system cannot discharge the hydrogen, it could build up and result in a detonation. To prevent hydrogen explosions passive catalytic recombiners should be installed. ENSI has requested Muehleberg NPP to report about the protection against hydrogen detonation until end of March 2012.
- Some issues were detected during the checks for the EU stress tests: For Muehleberg NPP they concern the seismic instrumentation for automatic scram, seismic robustness of the containment isolation, impacts of total debris blockage of hydraulic installations, extreme weather conditions, Severe Accident Measures and deployment of mobile equipment.

WEAKNESSES THE SWISS STRESS TEST IGNORED

- Muehleberg NPP is an old Boiling Water Reactor (BWR), which does not fulfil the Swiss nuclear regulator´s requirement of strict physical separation of redundant safety systems. All pumps and heat exchangers for the emergency core cooling and decay heat removal systems are installed in the Torus annulus space without physical separation, because there is not enough space for walls. All these pumps are on elevation -11m, which is below the water table of River Aare. However internal flooding in the reactor building could result in a cliff edge effect. The system for pumping-out the containment has a small capacity and is not seismically qualified, thus all emergency cooling systems could fail.
- According to the Seismic Hazard Assessment of Switzerland of the year 2004, earthquakes with a maximum magnitude of 7.5 can occur all over Switzerland. Using a specific predictive spectral ground motion model for Switzerland, expected ground motions were assessed in units of the 5% damped acceleration response spectrum at frequencies of 0.5-10 Hz for all of Switzerland. Maximum spectral accelerations at 5 Hz frequency reaching 150cm/s^2 (0,15 g) for a return period of 475 years and 720 cm/s^2 (0.72g) for 10,000 year. Events of magnitude 6-7 are now believed to be possible in all regions of Switzerland, but in areas of low seismicity, their return periods may be > 10 000 years to be known from the historical or even the paleo-seismic records. [GIARDINI et al. 2004]
- According to the IAEA safety requirements it would be recommended to assess the earthquake with the maximal PGA not with the 10,000 year return period but with the 100,000 years period. “In highly active areas, where both earthquake data and geological data consistently reveal short earthquake recurrence intervals, periods of the order of tens of thousands of years may be appropriate for the assessment of capable faults. In less active areas, it is likely that much longer periods may be required.” [IAEA 2004]
- In case of a severe earthquake, which causes a seismic surge, the intake of the emergency cooling water supply system could be blocked by debris; even if river water can pass the periscopic pipes the heat exchangers in the cooling system could be clogged with dirt. The consequence of this event would be a station blackout and a total loss of decay heat removal. This problem is not sufficiently considered in the report. A second diverse system for decay heat removal with another source of water would be a considerable improvement.

- Muehleberg NPP has been in operation for 40 years; ageing of components and equipment weakens the material; fatigue, abrasion and cracks could pose a hazard, for example due to an earthquake. Small failures could develop into breaks (pipes and tanks), pumps and valves and other equipment could fail. Also building structures suffer from material fatigue. The ENSI report ignored the problems caused by the ageing phenomena.

CONCLUSION

The ENSI report is based on the operators' assessment of their plant; ENSI explains that the Swiss NPPs have implemented many safety measures long before the Fukushima accident. Since the Swiss regulators have demanded most of these measures, the stress test report considers the safety level and the safety margins of the Swiss plants in principle as sufficient.

For Severe Accident Management mobile equipment plays an important role, mobile equipment is stored on-site and in an emergency centre, from where helicopters could transport equipment to all Swiss NPP sites.

For the Muehleberg site it is not possible to exclude that a seismic event exceeds the seismic design basis. The PEGASOS project results indicate that the current design maximum PGA of 0.15g for safety significant buildings and systems of Muehleberg NPP could be exceeded at a frequency of approx. 6 x E- 4 per year, which is not extremely rare. At present Muehleberg has no diversified Ultimate Heat Sinks (UHS) and lacks protection against hydrogen explosions in the reactor building (from SFP).

A total loss of cooling due to internal flooding is possible at Muehleberg NPP, because all pumps of the emergency cooling and decay heat removal systems are in the annular space without any physical separation.

The generic assessment for BWR-4 / Mark I containments estimated that the end cap lifts at 0.6 MPa (a rather modest pressure). The end cap is also expected to fail at about 370ºC, a temperature easily achieved in the drywell if the reactor vessel fails, depositing core debris on the drywell floor. In this case, it seems likely that the drywell end cap would lift, resulting in containment failure. This is what happened in the case of the Fukushima accident in 2011

KEY RESULTS OF THE PEER REVIEW

EARTHQUAKE

For NPP Muehleberg (KKM) the largest seismic hazard is the failure of the Wohlensee dam. According to current knowledge the seismic safety margin for the Wohlensee dam wall is the limiting parameter for cooling down the reactor core. The dam is very old and in case of a severe earthquake the dam is expected to break. After a breach of the Wohlensee dam the expected consequence is a clogging of the cooling water intakes of the special emergency system. In this case cooling of the reactor could fail.

The flood height was already covered by the original design. ENSI already requested an alternate heat sink diverse to the water intake of the river Aare (2015), so that a breach of the Wohlensee dam will not be limiting in the future.

Switzerland finalizes currently a reassessment of seismic hazard and the results are expected by the end of 2012. Retrofitting of SFP cooling systems at NPP Muehleberg is planned.

FLOOD

All sites consider the consequences of river water discharge, which substantially exceeds the 10^{-4}/y flood (by 20% or more). No cliff-edge effect is identified for any plant in this regard. Only NPP Muehleberg states that a simultaneous breach of several dam walls may trigger a cliff-edge effect as this situation could possibly lead to higher flood levels than those on which the design is based.

NPP Muehleberg has implemented a number of improvements of the robustness against external flooding.

EXTREME WEATHER

PSAs have only been undertaken for beyond design extreme winds and tornadoes. Margins for other extreme weather events and combinations thereof have not been considered. The peer review group concludes that such margins might exist because the Swiss NPPs cover external events such as explosions and aircraft crash. Nonetheless the peer review team recommends considering the assessment of margins with respect to extreme weather conditions exceeding the design bases, e.g. by extending the scope of future PSRs.

LOSS OF POWER AND LOSS OF UHS

Seven layers of electrical supply are available, which is remarkable. Also three safety trains assure different success paths to bring the plant to a safe shut down condition. The implemented design features as reported by the licensees indicate considerable robustness against SBO.

Provisions against loss of UHS are an outstanding strength in the Swiss NPPs. The exception is NPP Muehleberg with only two water intakes, which are not completely alternate cooling sources. ENSI required a seismically qualified alternative cooling source for NPP Muehleberg.

SEVERE ACCIDENT MANAGEMENT

According to the peer review the design and further development of the plants are based on the “defence in depth” concept and in consequence results in good robustness of the plants against severe accidents. More specifically, the SAMGs which are required in regulations seem to have been based on firm foundations, notably on international standards and plant-specific analyses of the potential severe accident phenomena, as well as on knowledge gained from the Level 2 PSA. Hardware provisions for severe accident management seem to be adequate and appear to have been robustly designed, resulting in significant safety margins. ENSI has ordered a number of evaluations and back-fitting.

There is no comprehensive evaluation of the instrumentation required in order to initiate and implement the individual accident management measures (prior to a containment integrity failure). The peer review recommends that the regulator assesses the opportunity of requiring more reliance on passive systems for hydrogen management for severe accident conditions.

CRITICAL COMMENT

NPP Muehleberg has significant deficiencies concerning the design and the “defence in depth” concept. Some of these deficiencies are described above: weakness of the containment, hazard of total loss of cooling, protection against hydrogen generation and accumulation. Therefore we cannot support the conclusion of the peer review report that states “that the further development of NPP Muehleberg is robust against severe accidents”. KKM is one of the oldest plants still operating in Europe and it has obvious design deficits which cannot be eliminated. The German NPPs, which have been taken from the grid in 2011, were not as old as NPP Muehleberg.

4.8 NPP Ringhals, Sweden

The Ringhals nuclear power plant (NPP) is owned by Vattenfall (70 %) and EON (30%) and operated by the license holder Ringhals AB. The Ringhals NPP is situated on the west coast of Sweden about 60 km south of Gothenburg. The distance to the neighbouring country Denmark is about 100 km. With a total net capacity of 3670 MWe, it is the largest nuclear power plant in Sweden. The plant comprises four reactors – one boiling water reactor (BWR) and three pressurised water reactors (PWR):

	Reactor Type	Net Capacity [MWe]	Commercial Operation
Ringhals-1	BWR, ABB Atom	854	01 Jan, 1976
Ringhals-2	PWR, Westinghouse 3-loop	809	01 May, 1975
Ringhals-3	PWR, Westinghouse 3-loop	1057	09 Sep, 1981
Ringhals-4	PWR, Westinghouse 3-loop	950	21 Nov, 1983

The Swedish stress test report was published by the Swedish Radiation Safety Authority (SSM).

Weaknesses the Swedish stress Test Described [SSM 2011]

- Work is currently on-going at Ringhals in order to fulfil the regulation regarding design basis earthquake (DBE).[28] The corresponding requirements entered into force in 2005. In order to allow licensees sufficient time to take the measures and fulfil the requirements, the deadline for taking measures is the end of 2013.
- Rough evaluations of the Ringhals-1 reactor building indicate that the roof can be a weakness. In case the roof cannot withstand a DBE, roof elements may fall down into the spent fuel pool. This can cause damages to the fuel and endanger the possibilities for external cooling.
- Identified deficiencies regarding earthquake are fans on vibration isolators and anchors (for example for cabinets, which houses relays) at Ringhals-2, -3 and -4; control room ceiling at Ringhals-3 and -4.
- No seismic qualification has been performed for spent fuel cooling systems. The alternate cooling modes are possible to use only if the water level in the spent fuel pool is enough to provide radiation shielding.
- SSM assesses that the data is somewhat lacking for demonstrating that functions needed to bring Ringhals-2, -3 and -4 to a safe state during and after a DBE.
- Since there are deviations for the DBE, 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 exceeding the DBE.
- 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.
- For Ringhals-3 and -4 it is not demonstrated that in a case of a blockage of the cooling water intake, the units will be able to safely shut down and maintain safe shut down conditions.

28 The (DBE) is an earthquake with the occurrence frequency of 100.000 per year.

- The Ringhals sites ground elevation is +3 m and hence above the design basis flood (DBF, return period of 10,000 year) level of +2.65 m. Waves are not included in the DBF calculated by the licensee. He claims that no unit is located in direct contact with the sea. However, SSM requires to consider the effect of waves in combination with extreme sea levels 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.
- If the water level inside the units is above +3 m (ground level of the site) the cooling of spent fuel pool, the containment pressure and temperature limitation, the residual heat removal and the emergency cooling system will fail. The licensee plans to eliminate the possibilities for water entering the building (installing new doors, improve sealing's etc.), but only in case of a sea water level up to + 3.3 m.
- In case of a sea water level between +3.3 and +4.0 m (65 cm to 1.35 cm above DBF), large amounts of water will enter the Ringhals units through various openings. The units will hence be significantly affected by the flood and fuel damage is possible. In case of sea water level above +4 m (1.35 cm above the DBF), all doors of the units will break and water will instantly flood all units and fuel damage will occur.
- 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 m below the average sea water level and the time from pump failures until the situation gets critical is very uncertain.
- Ringhals-1 and -2 need plans for snow removal from their roofs, which is an ongoing work. If no snow removal will take place, the maximum consequences of a collapse of the Ringhals-1 reactor building or the Ringhals-2 fuel building is the damage to the fuel in the fuel pools.
- According to SSM, the licensee has not dealt with how severe weather will affect the access to the plant for staff, heavy equipment and supplies. There is not described any procedures to prevent shortcomings due to blockage of access roads connected to extreme weather. Furthermore the condition and consequence of an ice storm for the plant is not evaluated.
- The roofs of the Ringhals-2 diesel building and Ringhals-1 reactor building require measures to withstand hurricanes or hurricane missiles. But only measures at the diesel building are being taken. The consequences of a missile impact on the roof of Ringhals 1 are judged to be acceptable, regarding the 10^{-6} per year probability for the occurrence of the hurricane.
- Ringhals-2, -3, and -4 will not withstand the defined extreme lightning. This has earlier been identified, and is an ongoing work.
- In case of loss of ultimate heat sink supplying the spent fuel pools from the fire-water system could only performed as long as the environment in the spent fuel pool area is acceptable to enter without any protective equipment.
- The primary ultimate heat sink for all units at Ringhals is sea water. The units Ringhals-2, -3 and -4 (PWR) have also 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 at all.
- During total SBO, the units rely on mobile units; however, the number of available mobile units at the site is not sufficient, in particularly in case of simultaneously events at more than one unit. If the mobile unit is unavailable, fuel damaged becomes unavoidable after approximately 16 hours for Ringhals-1 and after approximately 9 hours for Ringhals-2, -3 and -4.

- In case of a total SBO, the reactors rely on the battery power supply for instrumentation and control. Battery power supplies are only qualified for 2 hour of operation without charging.
- In principle the Ringhals 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.
- 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, SSM requires performing an evaluation of the system for the long-term operation.

WEAKNESSES THE SWEDISH STRESS TEST IGNORED

In **July 2009, the Ringhals nuclear power plant has been placed under special investigative measures and conditions by the Swedish Radiation Safety Authority (SSM) to address deficiencies in safety culture at the site.** According to SSM, a series of failures in safety culture have been highlighted during normal supervision of operations since 2005. This could ultimately jeopardise reactor safety.

Ringhals management needs to improve the self-assessment process at the plant, an IAEA Operational Safety Review Team (OSART) concluded March 18, 2010 after a three-week review. According to experts, the process is not an effective management tool in its present form. The team also said there needs to be a better system for evaluating whether training programs for maintenance work are working as intended [NW 2010; NW 2011].

In May 2011, a fire broke out when performing a pressure test of the containment caused by a forgotten vacuum cleaner. The management decided to start this test three days earlier than scheduled, but forgot to inform the personnel. When containment pressure was increased, a short circuit in the vacuum cleaner caused the cleaner to burn. The fire itself did not constitute a security threat and was quickly extinguished. But as a result, there was soot in places which are very difficult to clean [TAZ 2011].

During cleaning after the fire, scrap from welding work was found in the containment sprinkler systems at Ringhals-2 and – later in Ringhals-4 during inspections. Scrap in the containment sprinkler systems at Ringhals-2 and -4 would have caused the systems to function at 85% efficiency, but failure to detect the scrap is worrisome. SSM also demanded a report explaining why testing of safety systems has not been done properly. Considerable work was done on the units in the 1980s and 1990s and it is possible the scrap has been there since then. According to SSM the sprinkler systems were tested with air rather than water. That could be the reason the scrap was not discovered [NW 2011; TAZ 2012].

Checking and maintenance of safety systems is a key to nuclear safety. According to SSM, Ringhals management will have to explain in its report why safety culture problems have persisted at the site [NW 2011]. On November 8, 2011, SSM said that a number of the shortcomings which led to the plant being put under special oversight in 2009 remain and that plant management needs to consider whether the measures it is taking to improve safety culture are “the correct ones and if they are sufficient.” SSM’s decision to continue its special oversight came after a review of Ringhals management’s half-year report to the authority, which is required under the oversight [NW 2011a].

On April 2, 2012, Ringhals-2 has been connected to the electricity grid after one year shut-down, mainly caused by massive cleaning-up operation after the above mentioned fire. SSM approved restart, with conditions to submit several reports on how the unit is operating. The fire generated a substantial amount of ash that is difficult to remove from the containment. The functionality of electric and control systems could be affected [NW 2012].

CONCLUSION

The evaluation of the Ringhals site in the light of the Fukushima accident and in accordance with the ENSREG stress tests specification has revealed a number of shortcomings.

In their current state, the four Ringhals reactors are not able to withstand a design basis earthquake (DBE). The design basic flood (DBF) is not calculated according to the-state-of-art. The reactors have no safety margins, neither in case of an earthquake nor of a flooding. Clearly Ringhals is not to be called “robust” the design basis showing a range of known deficiencies.

A very serious safety issue is the lack of an alternate ultimate heat sink.[29] Any problem with the ultimate heat sink (water intake at the sea), will affect all units. All four reactors and four spent fuel pools would have to be supplied with cooling water.

Another major issue at Ringhals is the insufficient spent fuel pool cooling abilities. They have to be reinforced immediately. Emergency response organisation needs to be reviewed; battery capacities as well as the water supplies need to be increased substantially.

Safety culture at NPP Ringhals has been a serious problem for many years. More cases of undetected sloppiness, like the scrap in Ringhals-2 cannot be excluded. This could result in a partly or total failure of safety systems in the course of an accident.

Due to the existing hazards and insufficient safety culture, the units Ringhals should stop operation as soon as possible – at least until the reinforcement against earthquake and protection against flooding is performed as well as all other known deficiencies are remedied.

KEY RESULTS OF THE PEER REVIEW

EARTHQUAKE

The methodology used for seismic hazard assessment (SHA) is not fully compliant with current international standards and research results as requested in the ENSREG specifications. Based on the discussions during the country visit, SSM agreed to reconsider the existing approach by taking into account the geodetic and paleoseismologic data.

The full compliance in accordance with the current Swedish requirements, originally not designed to withstand seismic loads, is expected in 2013 after the implementation of modifications (e.g. anchoring of mechanical components, emergency power supply). Further investigations are to be performed to evaluate the margins against ground motions exceeding DBE. Some measures to improve the situation during and after an earthquake have been identified, such as alternate ways to cool the spent fuel pools.

The peer review team recommends SSM to consider in timely manner the fulfilment of assessments and the implementation of identified back fitting measures.

FLOOD

In 2005 a new regulation came into force; the licensees have to meet sufficient provisions against flooding by 2013. In addition, the licensees have identified further actions in response to the recommendations issued by World Association of Nuclear Operators (WANO) after the Fukushima accident. SSM states that the only source of flooding considered in Swedish units is from high sea water level. However, the detailed methodology for the definition of high sea level is identified as an

29 The German Regulatory Body is presently analysing the necessity to require a diverse ultimate heat sink for all German plants.

open issue by SSM. According to the report the Ringhals units are assumed to be significantly affected by the flood, the severity of potential fuel damages depend on a number of factors such as which rooms are flooded and the duration of the flooding.

Conclusion of the peer review team: The peer reviewers appreciate that the licensees compile a list of improvements for each NPP and recommend SSM to consider approving and implementing the possible improvements in a timely manner. They consider useful to carry out more detailed flooding risk analysis including cliff edge analysis. In order to identify plant vulnerability against flooding, implementation of a refined external flooding PSA could be suggested to be introduced for Swedish NPPs. The peer review team recommended for the Ringhals site studying the combination of high sea water level and other external phenomena such as swell, strong wind and organic materials.

EXTREME WEATHER

The current Swedish regulation addresses extreme weather without quantification of the loads. Some cliff edge effects and improvements are identified in the national report (e.g. against tornado induced missiles), but it is expected that the further analysis will identify additional measures against extreme weather (ice storms are of particular interest). SSM has requested the licensees that analysed the margins of their plants only by engineering judgement to perform further analysis in this area.

Regarding external hazards, safety also relies on the quality of early warning systems, which are not in place for all sites. Peer reviewers consider that such systems, as well as relevant operating procedures in case of extreme weather conditions, should be implemented at all sites.

Conclusion of peer review team: There is limited information available on extreme weather conditions and the evaluation of the respective hazard. Reviewers' appreciate that the licensees compiled a list of improvements for each NPP and recommend SSM to consider approving and implementing the possible improvements in a timely manner. They further recommend carrying out more detailed external hazard analysis on the basis of the state of the art requirements.

LOSS OF POWER AND LOSS OF UHS

The SSM confirmed that the overall intention of the stress tests had been fulfilled. Identified deviations from the ENSREG stress tests specifications were explained and justified.

Gas turbines are installed as alternate AC power sources. The 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. There are various measures proposed to be implemented in the area of power supply at all levels of defence in depth. The review team concluded that measures identified are appropriate and considers as one of the most important increasing the number of mobile diesel generators at site (each facility should get its own mobile diesel unit with prepared connection points).

If the ultimate heat sink (UHS) is lost, fuel damage can occur in the reactor core and/ or spent fuel pool quite rapidly depending on the level in the fresh water supply tanks systems. There are various measures proposed in the area of UHS. The peer review team concluded that measures identified are appropriate and considers the installation of pipelines to provide fire water to spent fuel pools as one of the most important.

SEVERE ACCIDENT MANAGEMENT

Important measures for SAM were already introduced in Swedish NPP in the eighties; measures are being enhanced in line with new knowledge. However, there are no measures to cope with a multi-unit event. Additionally, the capability to cool the spent fuel pool is very limited. The containment filtered venting systems are not designed for accident scenarios with prolonged duration (> 24 hours). Further weaknesses are:

- Consideration of multiple unit events including long term effects
- Consideration of natural disasters leading to loss of infrastructure
- Concepts to manage large volumes of contaminated water
- Accumulation of hydrogen in rooms or buildings outside the containment.

The assessments done by the licensees and the review carried out by the regulator in the area of SAM show the need of comprehensive improvements as well as the need of further analyses. The peer review team suggested considering the following issues:

- Instrumentation for measurement of water level and temperature in the spent fuel storage
- Enhancement of the accident management programmes (SAMGs, EOPs) for all plant states including spent fuel pools and multi-units events;
- EPO training and drills for extended scope of the AM such as consideration of multi-unit accidents under conditions of infrastructure degradation.

CRITICAL COMMENT

The protection against earthquake, flooding and extreme weather events is not in compliance with current Swedish requirement. Additionally, these risks are not assessed according to the international state of the art. That means there are known risks as well as other unknown, risks.

The available time for intervention measures is very low. It has to be expected that an external event on one hand will affected all units simultaneously, but on the other hand the staff will not be able to cope with a severe accident at all four units at Ringhals site simultaneously. This might result in very serious consequences: Large radioactive releases from the reactor cores and the spent fuel pools.

4.9 NPP Temelín, Czech Republic

NPP Temelín is located in South Bohemia, about 25 km north of České Budějovice. The Czech Stress Test Report was written by the State Office of Nuclear Safety (SUJB), based on reports prepared by the operator. NPP Temelín is in operation since 2000 (unit 1), 2002 (unit 2).

NPP Temelín consists of two units containing pressurized water reactors (PWR) of the type VVER 1000/V320, which has a primary cooling circuit with 4 loops. Each loop has one steam generator (SG) and one main cooling pump (MCP). The VVER-1000 unit has a nominal thermal output of 3000 MWth and a nominal electric output of 1000 MWe. During construction several technical modifications were implemented to achieve “western” safety standards. Those measures included a new I&C system, replacing the original cables with non-inflammable ones and other significant modification in the electrical part; qualification of pressuriser safety and relief valves for working with water and SG safety valve with water and steam-mix, implementation of a reactor pressure vessel (RPV) evaluation program and measures for the protection of the high energy pipe-line at the elevation +28.8 m.

The primary circuit system is enclosed in a full pressure containment, which was built using pre-stressed concrete. The containment is a cylindrical construction with an internal diameter of 45 m, closed with a semi-spherical cap. The inner surface of the protective containment is a hermetically-sealed steel liner. The spent fuel storage pool is also located in the containment. Under normal operation, residual heat is transferred into the atmosphere via cooling towers (2 for each unit).The ultimate heat sink (UHS) is the atmosphere.

In emergency situations the residual heat can be transferred into the atmosphere via the SG and the steam discharge station into the atmosphere or via the essential service water system (ESW) and the cooling tanks with sprinkler system.

The active safety systems have 3 x 100% redundancy; they are independent and physically separated. The passive safety systems (hydro accumulators inside the containment) have 2 x 100% redundancy.

Weaknesses the Czech Stress Test Described [SUJB 2011]

- SUJB requires the original containment to be equipped with a feasible solution of containment venting for Severe Accident Management.
- Lack of alternative replenishment of diesel fuel from a tank for long term operation of the diesel generator (DG).
- Lack of diversity of means to transfer heat to the cooling tower (UHS).
- Analysis is required for heat removal from primary cooling circuit following a loss of ESW.
- A procedure for containment insulation in shutdown state must be developed.
- Hydrogen removal in the containment for severe accident situations is missing.
- Functioning of equipment in BDBA conditions must be verified
- Several improvements are mentioned in the report, concerning emergency procedures and organization, alternative cooling options for the spent fuel pool, habitability of the main control room (MCR) and the emergency control room (ECR).
- Taking into account the reactor's thermal power and the design-basis solution of the concrete reactor cavity, there is no possibility for VVER 1000 units with V320 reactors to ensure any RPV cooling from outside. Measures for retention of core melt outside the reactor pressure vessel is mentioned as a possible safety improvement. When the RPV fails, the core debris would move to the concrete reactor cavity or other parts of containment. Molten core-

concrete interactions could result in containment failure. The applicable strategy in SAMG provides instructions for containment flooding to protect the concrete at the bottom of containment; thus also ensured is an efficient washing-up of fission products leaking from the melt. Cooling the pool of molten materials in the cavity with water may reduce the rate of concrete decomposition and hence postponing containment failure to a later stage of the accident.

Weaknesses the Czech Stress Test Ignored

- SUJB made big efforts to justify the seismic design and prove that the resilience of NPP Temelín is sufficient. According to the National Report the "binding design value of SL2 is PGA = 0.1g, which is comparable to a quake with an intensity of 7° MSK."
- The National Report describes the various methods used to assess the Design Basis Earthquake (DBE).The Report concludes: "There are no tectonic structures within the Czech Republic that would be able to generate strong earthquakes. The evaluation of the historical data and long-term monitoring revealed that the site of the Temelín NPP is seismically very quiet. The results from the network for detailed seismic regionalization substantiate the correctness of the conclusion of the seismic assessment of the location of the Temelín NPP. With 95% likelihood, an earthquake of a magnitude exceeding 6.5° MSK-64 (PGA = 0.08 g) cannot occur within the location of the Temelín NPP." The low probability of 5 % is used as the argument to ignore an earthquake exceeding the PGA of 0.08g.
- Description of accident sequences are complex and do not consider all potential failures of systems and components connected to the event. For example only a part of the switchyard is seismically qualified. Therefore, the analysis should not assume that all switchyards, lines and connections will be available or that all pipes, pumps, and tanks will be intact.
- The containment is equipped with a sprinkler system to maintain stable conditions in case of an accident. Pumps suck the coolant from the containment sump via the SIS exchanger and inject it into the containment with sprinkler jets. The steam condensates on the droplets of water, and leading to decreased pressure in the containment. The water drains into the containment sump from where the heat is transferred into the atmosphere using the ESW system in the ECCS exchanger. The sprinkler system has 3 x 100% redundancy, including all auxiliary systems (cooling, power supply, control and ventilation). Each of the 3 subsystems contains a sprinkler pump, tank with a solution of H3BO3, N2H4, KOH (to capture iodine vapour), water pump, Safety injection system exchanger, connecting piping and system of sprinkler jets.
- Another option is to transfer the heat from the containment by dispersing water in the rooms containing the Main Cooling Pump (MCP) engines. Although this system was designed for extinguishing fire in the MCP engines, it is connected to other rooms in the containment, and therefore may be used with similar effect to the standard containment sprinkler system. Heat removal is probably not assured by the fire protection system as well as KOH and H3BO3 supply.
- Since there is no possibility for cooling the reactor pressure vessel from outside in severe accident conditions, the retention of the melt in the RPV is not assured.
- The baseplate of the containment is on elevation +13 m. In case of a core melt accident, the baseplate could fail after 24 hours.[STRASKY 2011]

Conclusion

The National Report by the Czech nuclear regulator considers only the minimum of initiating events: earthquake, flood and extreme weather and the obligatory assessment of loss of UHS and SBO. For all these issues the worst case is the maximum event in a period of 10,000 years. The

Report (SUJB 2011) mentions only the following impacts:

The breaking of the Lipno dam would correspond to a 10,000 year flood. This event would cause flooding of a major part of the pumping station of the raw water supply. However there is a sufficiently large water reservoir on site to cool down both reactors until reaching cold shutdown status.

The civil structures of the Temelín NPP site are designed to withstand the flooding of maximal rainfall (88 mm in 1 day). The corporate fire brigade has mobile equipment to drain floodwater exceeding the 10,000 year maximal rainfall.

SUJB made big efforts to justify the seismic design and prove the resilience of NPP Temelín as being sufficient: Obligatory value for Temelín SL 2 is 0.1 g. According to the SUJB report this figure already includes a sufficient margin to the maximal peak ground acceleration of 0.08 g. Several international expert studies already found his assessment of seismic risk in Temelín to be insufficient and not reaching the State of Science [UMWELTBUNDESAMT 2001; UMWELTBUNDESAMT 2005]. Initiated by the Joint EU-Czech Republic Parliamentary Committee, the Czech and Austrian experts intensively discussed this topic in 2007/2008. This resulted in implementing two Czech-Austrian projects (“Interfacing Projects“, CIP and AIP), which are currently being conducted and will deliver a data base for the seismological assessment of the site. [UMWELTBUNDESAMT 2011]

The National Report states: “With 95% likelihood, an earthquake of a magnitude exceeding 6.5° MSK-64 (PGA_{hor} = 0.08 g) cannot occur within the location of the Temelín NPP.” The low probability of 5 % is used as the argument to ignore an earthquake exceeding the PGA of 0.08g.This is in direct contradiction with the very idea of the Stress Tests, which is to analyze those low frequency events, which could have severe impacts on the NPP. The SUJB Report stated that PGA values exceeding 0.15 g could threaten the fulfilment of the safety functions. The operator report discusses an accident scenario where a mechanical defect caused by a beyond design basis earthquake could prevent control rods from falling into the core. The reactor would be shut down through the negative reactivity effects, or by the injection of boron into reactor cooling system (RCS) using at least one of the three emergency boron injection systems, provided that the mechanical damage does not disturb the boron injection.

The seismicity issue is just one example for how CEZ and SUJB do not take the lessons from Fukushima and the very idea of stress tests seriously: events and sequences beyond the Design Basis need to be considered. „Evaluation of the historical data and long-term monitoring have revealed that the site of the Temelín NPP is seismically very quiet.“ [SUJB, p.210/211]

Another example is Temelín NPP on Station Black out which is assessed only under ideal preconditions where all other safety systems work and no other event occurs: „No design accident or failure was registered immediately before or after the SBO; the following in particular are excluded: Seismicity, fire, floods. All systems in the power plant, besides those systems that caused the loss of power supply for own consumption, continue to function or are able to function [SUJB, p.199].

In case of a severe accident with core melt, the retention of the molten core inside the vessel is not possible; therefore the operator proposes to implement severe accident management guides, which plan the flooding of the containment to postpone the release of a large amount of radioactive nuclides by washout. The containment baseplate could fail after 24 hour. The later the release the more radio-nuclides have decayed. However, even if the release is late the environmental contamination could be significant.

Key Results of the Peer review

Earthquake

According to SUJB, the seismic resilience of Temelín NPP buildings and selected parts of the NPP proved that relevant safety systems and structures significantly exceed the value of MDE (I=7°MSK / PGAH = 0.10g). Data are provided for a selected list of safety-relevant SSC showing that calculated HCLPF exceed the MDE (SL2) value of 0.10g. Fulfilling the safety functions could be at threat at acceleration values exceeding PGAH = 0.15g.

The original DBE assessment stems from the 1980ies. Improvements are identified in the area of the ongoing Seismic Hazard Assessment (SHA) to confirm or re-assess current DBE levels at the background of new geological, paleoseismologic and geodetic data obtained from the Bohemian Massif. This SHA is still to be validated. The peer review recommends that SUJB considers the implication during the PSR.

Flood

Convincing information was provided that the flooding from external water courses is “inherently ruled out” and the possible maximum flooding due to extreme rainfall is limited due to the morphological characteristics of the sites. At the same time it was suggested for the Temelín site to increase the resilience of the emergency diesel generators with a reference to the latest PSR.

Extreme weather

The peer review concluded on this topic, that the procedures for special handling of weather related threats need to be elaborated and some specific additions might be necessary to the emergency management procedures. The organizational arrangements to ensure the necessary staff in case of longer lasting extreme weather conditions have to be elaborated. The considerations for extreme low temperatures may be too simple, not taking into account the realistic related effects, e.g. station black-out. Some refined further analyses and verification of current analyses was judged to be necessary. The elaboration of diverse connection to the ultimate heat sink and the load analyses of specific civil structures are already in progress and it is recommended that the SUJB should ensure that the question of diverse ultimate heat sink is resolved effectively.

Loss of Power and Loss of UHS

As alternative heat sink of the plant it is proposed to pump water from fire trucks into SG through the so called super emergency feed-water system. This water will evaporate in the secondary side of the steam generator and the steam will be released into the atmosphere (secondary feed&bleed). Connections to inject water from fire brigade equipment are already available at Dukovany, and are planned to be installed in Temelín.

During the review meeting the peer review team has been informed that the time available to recover the lost heat sink before fuel damage in the worst case for Temelín is 2.5 hours for SBO; UHS being dependent on power supply, loss of UHS is an inevitable consequence of SBO.

There are different hardware modifications foreseen to enlarge coping times. Specific possible safety improvements for Temelín NPP related to the loss of UHS according to the peer review report are:

- Install new hook up points for fire trucks
- Develop a procedure for the loss of UHS and ESW systems in both units
- Extensive damage mitigation guidelines for the use of alternative means
- Alternative replenishment of water to steam generator/SFP/primary circuit (with unsealed

primary circuit)

- Analysis of heat removal from I&C systems following a loss of ESW

SEVERE ACCIDENT MANAGEMENT

According to the peer review report the following measures have to be implemented at NPP Temelín:

- development of SAMGs for shutdown modes including open reactor and SFP accidents;
- alternative containment sump water make up (Temelín)
- selection and implementation of appropriate solution for protecting containment from the overpressure loads
- providing mobile (portable) equipment for ensuring feasibility of the SAM actions
- further analyses of the impacts from the infrastructure damages, multiple Unit accidents on the SAM and emergency response provisions
- increase robustness of storage building structures for mobile devices including fire trucks, or relocation of equipment
- cooling of molten core is still an open issue
- installation of additional re-combiners sufficient for severe accident conditions

A filtered venting system designed for severe accident conditions to protect the containment and mitigate radioactive releases, should be analyzed for Temelín. Accidents during shutdown states and in SFP are not addressed in the existing SAMGs, but should be available in 2014; the implementation shall be monitored by SUJB.

CRITICAL COMMENT

The fact that the national Report of SUJB has considered only the minimum requirements of ENSREG for the stress test lead to many recommendations formulated by the peer review. The impression is that SUJB and CEZ did not take the stress test very seriously, if they were not able assess the time available to recover the lost UHS before fuel damage in the worst case. The discussion about the seismicity is still open. A lot of improvements have to be analyzed and implemented in Temelín. To rely too heavily on the fire brigades for cooling the reactor in case of an accident is not an acceptable solution.

4.10 NPP Wylfa, UK

NPP Wylfa is located on the North coast of the island of Anglesey in North Wales, United Kingdom. The UK stress test report is written by the Office of Nuclear Regulation (ONR), based on reports from the operators.

Magnox reactors are classified as Generation I. The Wylfa twin reactors are in operation since 1971. Wylfa is the last and largest of its type built in the UK. There are four turbo/generators (two per reactor) giving the station a total electrical output of around 840 MW. The adjacent Irish Sea provides the ultimate heat sink. The reactors are due for permanent shutdown in 2012 (Reactor 2) and 2014 (Reactor 1).

Magnox reactors are cooled by pressurised carbon dioxide (CO_2) gas (primary cooling system). The moderator is graphite. The fuel is natural uranium clad in a Magnox (magnesium non-oxidising) alloy. Power density of the Magnox reactor is low compared to PWRs and the core is much larger.

The heated CO_2 gas transfers the heat from the core via a boiler (=heat exchanger) to the secondary cooling water system which creates the steam for the turbo-generator.

The Magnox reactor core consists of interlocking graphite blocks assembled in a series of layers set on and within a core support and restraint structure. The graphite blocks have circular holes through them, thereby creating continuous fuel channels. The Wylfa NPP has over 6000 fuel channels per reactor core with 8 fuel elements per channel. Fuel exchanges are made during reactor operation. In addition there are several channels dispersed across the core, which are used for control rods and instrumentation.

Wylfa has 185 control rods which drop into the core by gravity in case of a reactor trip. The Office for Nuclear Regulation (ONR) states that this provides a shutdown massive redundancy: insertion of a small number of control rods is enough to stop the fission process. The primary shutdown system (control rods) was supplemented (with limited diversity) by the installation of the articulated control rods, which are to be inserted into the channels if the graphite blocks are disarranged. The reactors at the Wylfa site are also equipped with a tertiary shutdown system based on the injection of boron dust into the core, although this action would result in the permanent shutdown of the reactor.

The two pressure vessels at Wylfa are spherical internally with a steel liner of an internal diameter of 29 m. Magnox reactors have no secondary containment. ONR states that for Magnox reactors, containment can be considered to be provided by the prestressed concrete reactor pressure vessel and steel liner, associated penetration closures and connected plant and equipment that collectively form the reactor pressure boundary. The low power density and high thermal inertia means that there are significant timescales available in the event of loss of post-trip cooling.

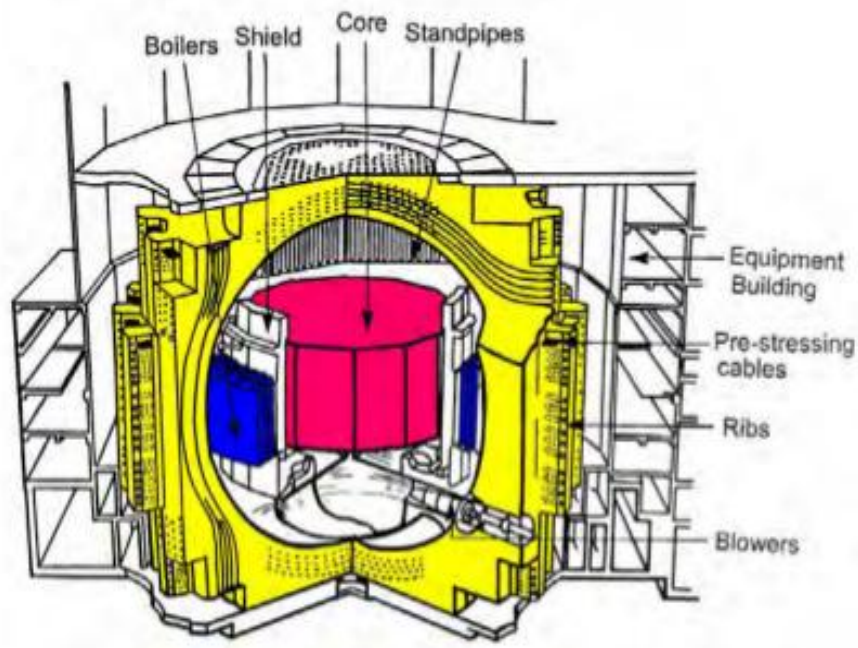


Figure 4: *Wylfa's Magnox concrete pressure vessel (ONR 2011)*

Magnox reactors have diverse and redundant systems for post-trip cooling. Providing the pressure vessel is intact the fuel is cooled by the gas circulators pumping the CO_2 coolant through the reactor core and boilers, with heat being removed from the boilers by the post-trip feed water systems (FWS). At Wylfa back-up feed systems are standalone with fuel and water for a minimum of 24 hours operation supplying both reactors. Wylfa also has a third standalone feed water system.

At Wylfa spent fuel is discharged from the reactor into the refuelling machine which transfers the fuel in one of three dry stores, which are supplied with pressurized CO_2 gas. The fuel remains in this storage until it has cooled down sufficiently to be moved into transport containers and brought two other on-site facilities that store the fuel in dry air.

Weaknesses the UK stress test Described [ONR 2011]

- Wylfa Design Basis Earthquake (DBE): PGA = 0.18g. Magnox reactors do not have automatic seismic shutdown systems. Therefore the operator has to initiate the reactor trip manually in response to a signal from the seismic monitoring system.
- The dry fuel stores at Wylfa are demonstrated to be robust against the DBE, although the cooling system is not. A summary of operator actions required in the safety case has been provided.
- Loss of external power (national grid connection) is assumed to occur at the instant that a seismic event occurs. There are no claims made in terms of a period to reinstate the grid connections.
- Essential stocks of CO_2, diesel and cooling water are provided for a period of at least 24 hours. The report examines the effects of a larger seismic event which may impact access to the site. The ONS report assumes that that access to the site could be re-established within 24 hours following an event consistent with the DBE.
- The third feed water supply system which should be available for Severe Accident Management relies on the availability of fire engines or other pumps to cool the core via the boilers; it is not clear how many times it would be necessary to organize a connection to the

pumps.

- Magnox reactors do not have containment buildings. The final barrier is the prestressed concrete pressure vessel. Thus prevention of over-pressurization of the RPV is essential. For that safety relief valves and blow-down routes are installed for lowering the CO_2 gas pressure. The discharge routes have filters to limit the discharge of particulates in design basis incidents. During severe accidents the filters may be clogged and then the filters are bypassed by bursting disks, which results in radioactive emissions into the environment.
- The design of gas cooled reactors did not consider severe accident hydrogen generation and associated risk. Hydrogen build-up in a gas cooled reactor building is possible in the event of gas release from pressure vessel penetrations, but this will be mixed with carbon dioxide and to some extent carbon monoxide. This could mitigate the risk of a fast deflagration to some degree; however the issue of combustible gas is still a cause of concern.
- Magnox fuel rods are solid uranium metal clad in a magnesium alloy. Magnox Ltd believes that even in a completely drained pond the irradiated fuel element temperature rise will not generate significant quantities of hydrogen. One Magnox Ltd site, NPP Wylfa, is unique in that it stores its discharged irradiated fuel in dry stores in a CO_2 atmosphere. The licensee states that hydrogen production there is not an issue.
- In comparison to Light Water Reactors, severe accidents at Magnox reactors show a slower development. Therefore the primary focus of accident management lies with restoration and maintenance of core cooling capability.
  ONR agrees that High Pressure Melt Ejection (HPME) from a Magnox pressure vessel is highly improbable. The core can melt and slump onto the vessel floor in severe accident conditions and, therefore the generation of gases could be possible. The potential explosive hazard arising from the production of CO during a severe accident is currently not fully understood. Pressure in the reactor building, due to CO or hydrogen build-up, is unlikely as the structures are not designed to retain pressure and many vent paths are provided for potential hot gas and steam release within the building, thus radioactive substances escape into the environment.

Weaknesses the UK Stress Test Ignored

Ageing effects which cause material degradation are not considered in the stress test report. Large defines three specific ageing effects concerning Wylfa [LARGE 2007]:

- Cracking of the RPV steel liner closure plates near the vessel wall penetrations carrying the superheated steam pipes from the boilers
- Corrosion of the internal steelwork of the reactor (core restraint garter)
- Corrosion (radiolytic oxidation) of the graphite core.
- The simultaneous failure of a group of steam pipes results in high pressure differentials within the reactor which subjects the graphite core and its restrain system to excessive loadings. The result could be the misalignment of the graphite blocks in the core. Disturbing the geometry of the graphite fuel channels could lead to local overheating of the fuel.
- For the case where the RPV failed air intrusion increased (chemical reactivity) burning of graphite and carbon dust contribute to overheating the core, resulting in temperatures sufficiently high in prompt magnesium and fuel ignition.
- The combination of external impacts (for example: earthquake) and material degradation can have significant impacts on the development of accidents. Even the three shutdown systems are not diverse enough to guarantee a safe shutdown of the reactor. “Distortion of the core may be a result of the fault condition underway at the time that emergency shut-down is

> required. Although the articulated control rods provide a degree of diversity, other than being able to cope with a limited degree of core distortion these rods share that same features and prerequisites required for operation for the other control rods (clutch release and standpipe route integrity)." The third shutdown system (injection of boron dust) also relies upon the remaining of the fuel channels accessible and on continuing coolant gas flow to fully disperse the boron dust all over the core" .[LARGE 2007]

CONCLUSION

The UK stress test report is written by the Office of Nuclear Regulation, based on reports from the operators.

Magnox reactors are classified as Generation I. These reactors have no secondary containment. The massive concrete reactor pressure vessel is the last barrier to retain emissions from the reactor core. Containment function relies on the stability and leak-tightness of pipings and welds penetrating the reactor vessel wall.

The Wylfa twin reactors are in operation since 1971. Wylfa is the last and largest of its type built in the UK. Despite Wylfa's 40 years of operation, the UK stress test report does not recognize material degradation as the main contribution to safety problems.

Corrosion of the internal steelwork of the reactor (core restraint garter) could in combination with an earthquake or in case of the break of a group of steam pipes (which results in high pressure differentials) result in distortion of the graphite blocks geometry. Thus could result in cooling and shutdown problems.

Air intrusion in the reactor core could result in graphite burning and thus contributes to overheating of the core.

In the stress test report the operators and ONR seems to have not much interest in improving the safety of the old and outdated Wylfa reactors.

The operator seems to consider investigating the safety problems as not worth the effort because this plant practically reached the end of its operational lifetime. Only one unit will be operating until 2014 and use up the last Magnox fuel elements. Recommendations for improvements of Severe Accident Measures are focused on mobile equipment, such as mobile diesel generators and fire engines to cool-down the plant in case of an accident. The ONR report assumes that the computerization of the I&C is not necessary for Wylfa's last years of operation.

Instead of installing a ventilation system which would maintain the habitability of the control room, staff at the Wylfa site will be provided with respiratory equipment.

Considering that the NPP Wylfa Severe Accident Management mainly relies on staff and on mobile emergency equipment, we recommend to shut down both Wylfa units immediately.

KEY RESULTS OF THE PEER REVIEW

According to the UK peer review report, 3 out of 18 operating reactors are Magnox reactors, these still operating Magnox plants may be defueled in about 6 years.

That is rather confusing information, because Oldbury 1 and 2 were shut-down in 2011 and 2012, respectively. Wylfa 2 was shutdown in April 2012. Wylfa 1 is now the only Magnox reactor still running.

As the Magnox reactor program winds down, limited supplies of fuel prompted the operator Magnox Ltd to close Wylfa 2 in to optimize operations at Wylfa 1, which is expected to continue in service until 2014. The plant received the last delivery of fresh Magnox fuel in December 2011 and Magnox

Ltd has requested regulatory approval to transfer partially used fuel from unit 2 to unit 1, which is relatively simple because of the shared service floor and refuelling machine.[30]

The ONR stress test report does not care much about the oldest still operating NPP in UK and this lack of interest appears also in the peer review report.

EARTHQUAKE

A beyond design basis capability is inferred but not quantified and no specific evidence is provided that margins to cliff edges and potential specific improvements have been considered systematically for all the NPPs. ONR has identified this as an area for improvement. This work will be monitored by ONR as part of the post stress test program of work.

FLOOD

The reviewers conclude that the currently available DBF assessments did not take into account recent tsunami research work. However ONR believes that these studies are unlikely to significantly affect previous understanding of maximum credible tsunami heights. **There is no satisfactory evidence of capability of the plants beyond the design basis.** The UK regulator was advised to consider introducing a specific program for additional review regarding the design basis, adequate margin assessment and identifies specific potential plant improvements. **ONR requires the operators to update their design basis assessment with consistent data and state of the art methods.**

ONR were requested to consider clarifying its technical requirement in the implementation of the defence in depth principle regarding flooding, and consider requirements for warning and prevention of flooding of the site, protection against flooding of rooms and mitigation, for the whole site. ONR accepted this and already raised findings on beyond design basis/margins. The EU peer review team has confirmed with ONR that this will include the defence in depth principle.

EXTREME WEATHER

For some specific external hazards, beyond design basis capability are inferred but not quantified and no specific evidence is provided that margins to cliff edge effects and potential specific improvements have been considered systematically for all NPP. In some cases there is no satisfactory evidence of a plant´s capability the beyond design basis (e.g. tsunami, lightning).

Cliff edge effects beyond design basis have not been established in a consistent manner for external hazards. The review team noted that the UK regulator has claimed for additional review regarding the design basis approach and an adequate response regarding margins assessment beyond the design basis and identifies specific potential plant improvements. **The review team encourages the ONR to establish a strong regulatory oversight program on this matter.**

LOSS OF POWER AND LOSS OF UHS

The peer review concluded that the plants comply with the license/safety case, but not all plants fully comply with all of the WENRA Reference Levels yet. They suggested the following improvements:

- Inject water into the reactor core as an ultimate means to provide residual heat removal from the core without use of the boilers and identify the means/equipment that would be used, including filtering (AGR/Magnox);
- Stocks of fuel etc, for at least 72h,

[30] World Nuclear News: http://www.world-nuclear-news.org/RS-Wylfa_2_bows_out-2604127.html.

- Battery capacity is low compared to other countries and therefore should be increased or recharged by additional generators for most of the plants.

For AGRs/Magnox the review team believes in the light of Fukushima, the longer grace times should not be used as an argument for postponing the implementation of fixed hardware provisions.

Specific to Magnox Ltd ONR has formulated two findings. One finding from the assessment of the progressive loss of electrical systems was not reported. The second finding concerns the demonstration of fuel integrity when the natural draft air ducting of the dry fuel stores becomes filled with water.

ONR strategy is to force the licensees to implement additional resilience measures in 2012 (connections and mobile EDGs). First step might consist of quick solutions, followed by engineered equipment.

SEVERE ACCIDENT MANAGEMENT

Importance of the SAM has been recognized in the UK as an important component of the defence in depth concept. Analyses of severe accidents have been performed for all operating reactors in the UK.

Emergency arrangements are in place to address severe accidents.

Accident management for gas cooled reactors represent a special case due to their unique design features, in particular absence of a separate containment building and very large thermal inertia. The majority of accident management measures focuses either on prevention of severe accident e.g. protection of reactor pressure vessel integrity, or on measures for mitigation of releases in the case of loss of vessel integrity.

In spite of the robustness of the plants (in particular gas cooled reactors) against progression of an event into a severe accident, hardware provisions are being implemented to further strengthen plant capabilities to prevent unacceptable consequences. Important enhancements should address long term behaviour of concrete under severe accident conditions.

In accordance with the existing plans, the on-site emergency facilities should be strengthened to be resistant against external hazards and provide for working conditions in case of severe accident.

The need for a backup control room providing for shutdown and cool-down to safe condition of the plant should be considered.

SAM guides should cover all spectrums of accident scenarios, including shutdown conditions. Radiation conditions which may potentially develop on site in case of severe accident, possibly at several units, should be more comprehensively analysed and appropriate measures to address them implemented.

The review team supports the existing plans of strengthening hardware provisions for SAM in all reactors.

The need for operability of newly installed equipment under conditions of extreme external hazards and prolonged SBO should be considered. Provisions for ensuring sufficient coolant inventory in the SFP should be further strengthened by providing e.g. additional delivery of coolant from external sources.

CRITICAL COMMENT

At meetings with operators and ONR it was stated that modifications should be implemented by the end of 2014. Magnox Ltd is planning to implement its enhancements in an even shorter timescale due to the stage in the lifecycle of the Magnox sites.

According to the ONR final report SAM for Wylfa relies mainly on mobile emergency equipment and staff. Wylfa was 40 years in operation. Material fatigue and other ageing effects can lead to breaks of pipes, damage of the reactor core geometry in case of an earthquake, or other external impacts. It is irresponsible to assume that in a NPP of this age all safety relevant components of the plant will stay intact under severe accident conditions.

It is a dangerous and desperate intention to fight a severe accident with mobile equipment only. Because hardware for severe accident management for Wylfa will probably not be invested, it is recommended to shutdown the plant immediately.

4.11 NPPs Fessenheim, Gravelines and Cattenom, France

All 58 French nuclear power plants (NPPs) are owned and operated by Electricité de France (EDF) and equipped with two, four or even six pressurised water reactors (PWR). The oldest reactors (34) belong to the 900 MW class divided in the CP0, CP1 and CP2 series; the 1300 MW reactors (20) consisting of P4 and P´4 series. The 1450 MW reactors, or N4 series, comprise four reactors.

Fessenheim NPP has two PWR of the 900 MW class, model CP0. The units, which started commercial operation in 1978, are the oldest operating reactors in France. The NPP is situated in a seismic area. The distance to Germany is about one kilometre, to the city of Freiburg about 30 km.

Gravelines NPP is the biggest nuclear power plant in France and comprises six reactors. All units belong to the 900 MW class, model CP1. Units 1 – 4 started commercial operation in 1980/81, units 5 and 6 followed in 1985. The NPP is situated on the French coast of the British channel between Calais and Dunkirk (both about 20 km). The distance to Belgium is around 35 km, to Bruges about 90 km.

Cattenom NPP comprises four reactors that belong to the 1300 MW class, model P4´. Commercial operation of the four units started successively in 1987, 1988, 1991 and 1992. The NPP is situated at the river Mosel about 9 km south of the boarder and about 50 km south of the city of Luxemburg.

The French Nuclear Safety Authority, the Autorité de Sûreté Nucléaire (ASN) published the French stress test report. After the accident in Fukushima, the French government put in place a process for complementary safety assessments (CSAs) for the nuclear facilities. These complementary safety assessments are part of a two-fold approach: on the one hand, performance of a nuclear safety audit on the French civil nuclear facilities in the light of the Fukushima event, which was requested from ASN by the Prime Minister and on the other, the organisation of "stress tests" requested by the European Council [ASN 2011].

Weaknesses the French Stress Test described [ASN 2011]

Regarding all operating reactors of the 900 MW and 1300 MW class:

- The Probabilistic Safety Analysis (PSA) does not include earthquake or any accident associated with the spent fuel pool.
- Only one emergency diesel generator is available per site, but it is not designed to withstand an earthquake. In the event of a common mode failure affecting all site backup diesels, only one unit could be backed up. In the event of an earthquake, even the availability of this emergency generator cannot be guaranteed.
- Fire detection and fixed extinguishing systems are not electrically backed-up by seismically qualified equipment; ASN will require EDF to reinforce the fire sectoring.
- Earthquake-induced explosion risk was identified, but EDF intends to do application implement the necessary modifications to hydrogen systems during the next ten-yearly outages, in 2023 at the latest. ASN will require EDF to speed this up.
- The seismic margins of safety relevant equipment (e.g. electrical equipment, seals between buildings and tanks) are small.
- The conformity work on the component cooling system section that is not seismic qualified is pending. EDF envisages speeding up this work.
- According to ASN, the seismic margin values evaluated on the basis of an analysis performed within a very short period of time are not sufficiently justified.
- The flooding of the Blayais NPP in 1999 illustrated the need for reinforcement of the French NPPs against flooding. However, not all modifications and tasks defined by the experience

feedback approach were implemented yet. These modifications primarily consist of elevating and strengthening the wave protection, electrical back-up for the plant sewer system pumps. ASN requires the work to be carried out latest until 2014.

- The management of volumetric protection[31] (VP) needs to be improved on several of the inspected sites. VP plays a key role in protecting the plants against the off-site flooding risk. ASN requires EDF to implement rapid conformity remediation work.
- Regarding flood protection, EDF did not take account the ageing of the “waterstop seals[32]”.
- ASN requires strengthening the protection against the risk of flooding in excess of the current baseline safety requirements, because the studies highlighted the existence of cliff-edge effects (loss of electrical power supplies) for levels close to those used in the baseline safety requirements.
- Studies taking extreme natural phenomena (snow, hail, lightning and wind gust) into account have not been prepared. ASN requires EDF to present such studies.
- The time before damage to the fuel becomes inevitable in the event total station blackout (SBO) for the entire site:
  -when the primary system is closed, after about one day;
  -when the primary system is just open, after ten hours;
  -when the primary system is sufficiently open: for the 900 MW class (Fessenheim and Gravelines), after a few hours.
- None of the operating nuclear power reactor has an alternate heat sink (lake, water table or atmosphere).
- However, the vulnerability of the ultimate heat sink was highlighted by the recent events of clogging and partial loss of the heat sink at Cruas and at Fessenheim in December 2009, which shows that reinforcement of all heat sinks is necessary. ASN requires completing the heat sink design review, in particular with regard to prevention of the risk of clogging.
- Analyses are needed to prove the robustness of the civil engineering structures needed to prevent loss of the heat sink (in particular the pumping station and networks) or electrical power supplies (in particular the electrical and diesel buildings)
- Certain key equipment (e.g. electrical equipment) situated in areas no longer cooled could be lost due to loss of heat sink situation.
- The availability of water from the fire-fighting network to make up the spent fuel pools is not guaranteed in the event of an earthquake.
- If all pumps of the fire-fighting water production or distribution system are unavailable, the fuel will be uncovered within one and a half day.
- EDF has been assuming that the integrity of the pools equipment remains intact and/or is functional. However, this is not justified (for example see below, INES 2 event in Cattenom NPP). A potential leakage can result in a cliff-edge effect, in particular because of the significantly faster dewatering of the fuel pool and the particular constraints of operational management of these accidents. Shortly after dewatering large non-filtered releases to the atmosphere occur.
- Current organisation and studies do not sufficiently address the management of a "multi-facility" emergency, possibly resulting from an external hazard, affecting all or part of the

31 Protective devices are implemented, whenever necessary, to offer protection against the flooding.
32 Tightness of the expansion joints in the concrete walls (water stop strip).

installations simultaneously.

- ASN requires number of means enabling the NPPs to deal with long-duration loss of electrical power sources or heat sink situations, capable of affecting all the facilities on a site. Pending the progressive deployment of these measures, which will take several years, ASN will require the implementation of interim measures as of 2012, such as mobile electricity generating sets.
- The equipment necessary for emergency management, and in particular mobile safety equipment, was not managed satisfactorily; the storage conditions did not guarantee permanent availability, particularly in the event of external hazards.
- Habitability and accessibility of the emergency management rooms and control rooms in the case of filtered venting is not guaranteed.
- Instrumentation dedicated to severe accident management, able to detect reactor vessel melt-through and the presence of hydrogen in the containment is missing. The installation is currently planned for the next ten-yearly inspections.
- Filtered venting systems are not seismically qualified. Also, the filters are designed to retain caesium, but they do not retain iodine, which is responsible for short-term exposure of people living in the NPP vicinity.
- There is no bunkered emergency control room.

Fessenheim:

- The seismic robustness of the dykes and other structures designed to protect the facilities against flooding and the possible consequences of a failure of these structures are not analysed. ANS requires such a study to be conducted.
- A failure of the Grand Canal d'Alsace embankments would result in a water layer on the site, likely to lead to a scenario involving total loss of the off-site and onsite power supply, as well as the potential loss of safety relevant equipment. ASN requires EDF to conduct studies to assess the precise level of water if the embankments fail and to evaluate the resulting consequences.
- The heat sink being at a higher elevation than the site platform means a major leak in the event of rupture of the cooling systems for the facilities connected to them.
- The basemats are only 1.5 meters thick, which doesn't guarantee corium retention for 24 hours. EDF is preparing to add about 70 centimetres to the basemats by the end of 2013.

Gravelines:

- The implementation of reinforcements regarding design basic earthquake (DBE) has been completed only at Gravelines 1. The implementation at the units 2 – 6 will be carried out during the ten-yearly outages at the other reactors. The work is scheduled to be completed in roughly six years (end of 2017).
- The retaining walls along the sides of the intake channel need to remain stable to guarantee the heat sink flow. ASN believes that EDF needs to carry out additional studies going beyond the safe shut down earthquake (SSE).
- A leak of toxic gas at nearby industrial facilities (port of Dunkirk) could make it impossible for the operators to remain at the reactor, because it is not protected against such accidents.

Fessenheim and Gravelines:

- High vulnerability against external events because the reactors are only protected with singlewalled containment structure; it is sealed with an inner metallic liner.

Cattenom:

- Cattenom NPP has a large water reservoir (Mirgenbach lake), which constitutes the backup heat sink. But on the other hand, the ultimate heat sink (river Mosel) has shortcomings regarding design basis earthquake and potential flooding events. Thus, this lake has actually acted as the ultimate heat sink. But there are doubts about the seismic margins of parts of this heat sink [MAJER 2012].
- The double-walled containment consisting of one reinforced concrete and one pre-stressed concrete structure was designed to provide better resistance to external initiating events. But the absence of an inner metallic liner has made those reactors more vulnerable to disruption from internal threats such as hydrogen explosions [MAKHIJANI 2012].

Hard core and FARN

To compensate the shortcomings a “hard core” of material and organizational measures should be created to ensure that the core and spent fuel pool can be cooled for at least 24 hours. External help would be available via the Nuclear Rapid Action Force (FARN) with mobile equipment such as emergency diesel generators and additional water sources. The FARN would make emergency response teams and equipment available within 24 hours of the start of a severe accident at a nuclear installation.

In the short term (2012-2015), the FARN will be in place. In the medium term (2016-2020), EDF will begin installing diesel generators as well as permanent protection for water sources that belongs to the “hard core”. The content and specification of this “hard core” has to be defined. Provisional measures will be put in place pending deployment of more major modifications [NW 2011b].

Weaknesses the French Stress Test Ignored

Regarding all operating reactors of the 900 MW and 1300 MW class:

Ageing has the potential to aggravate or to trigger an accident. An example for a safety relevant ageing issue is the occurrence of micro cracks in a bottom-mounted instrumentation penetration nozzle on the reactor vessel of **Gravelines-1**. The cracks were detected in non-destructive examination conducted during the reactor’s 30th-year outage in summer 2011. The issue is a potential safety concern because a nozzle with sufficient cracking could break off during operation. There are 50 bottom-mounted instrumentation nozzles on each of the French 900-MW-class PWRs.[33] ASN had now asked EDF to inspect all the bottom-mounted instrumentation nozzles on all its 900-MW-class and 1300-MW-class PWRs. But these inspections will be conducted during the units’ decennial outages scheduled for the coming years [NW 2011c].

The **fuel cladding** is made from an alloy called zircaloy, which consists mainly of the element zirconium. The uncovering of the fuel due to a loss of coolant from the reactor vessel triggers a series of events that can lead to core meltdown. Zircaloy plays a central role in this series of events as well as in the production of hydrogen and in the risk of an explosion. Experts recommend a systematic research and development program to find a substitute for zircaloy, with the goal of reducing the probability of a serious accident with core meltdown [MAKHIJANI 2012].

In December 2011, Greenpeace activists climbed into two EDF nuclear reactor sites to highlight the

33 These nozzles are vulnerable to stress corrosion cracking due to their metallic composition; alloy 600 for the penetrations and alloy 182 for the welds between penetrations and vessel base metal and stainless steel lining.

security issue. More stringent measures of passive protection (alarm systems, fences, and video surveillance) at nuclear sites are currently under consideration. In addition the government will revise the national security guidelines for all “vitally important sectors of activity including nuclear” and improve coordination among operators, police, armed forces, and safety regulator ASN. Design weaknesses make terror attacks easier and at the same time increase the “success” of a terror attack [NW 2011c].

Conditions concerning the use of **outside contractors** have been assessed during the complementary safety assessment (CSA) excluded from the scope of the European "stress tests". As a result ASN stated that EDF has not adequately demonstrated that the scope of the activities subcontracted, both in terms of the types of activities concerned and the internal skills preserved, is compatible with the licensee's prime responsibility for safety and radiation protection. Improvements have to be evaluated. Regarding the envisaged back-fitting measures is this fact a very important issue [ASN 2011].

<u>MOX fuel at Gravelines NPP:</u>

All six reactors are authorized to use **MOX fuel** and five of them are currently using it. MOX fuel presents a set of safety problems in case of an accident and its storage in spent fuel pools is more complicated due to its greater thermal source term. The consequences of the meltdown of a core that includes MOX fuel or of a fire in a spent fuel pool in which MOX is stored could also be much more serious than those involving only uranium dioxide fuel [MAKHIJANI 2012].

<u>INES 2 event at Cattenom NPP:</u>

On 18 January 2012, EDF notified ASN that the absence of a "siphon breaker" orifice on the **fuel storage pools** of reactors 2 and 3 had been detected. During an inspection carried out on 21 December 2011 as part of the measures taken following the post-Fukushima complementary safety assessments (CSA), the operator found out that these siphon-breakers were only present on the reactors 1 and 4. In the event of an incident, for example, inadvertent operation of certain valves, the injection pipe could extract the water from the pool through a siphon effect, instead of injecting it, which would lead to a drop in the water level. An orifice, the "siphon-breaker", is drilled in this pipe near the surface of the pool, in order to stop any siphon effect. A significant drop in the water level would lead to an exposure of the fuel assemblies and to their damage. The anomalies observed were mended between 1 and 3 February 2012. Owing to its potential consequences, this event was rated level 2 on the INES scale [NEWS 2012].

The absence of siphon breaker is not at all the first non-conformance the inspections in the frame of the CSA have been revealed. During the conformance checks conducted in August 2011, ASN observed 35 non-conformances during spot test. This large amount of and their safety relevance indicates an insufficient safety culture of the operator [MAJER 2012]. One example is the insufficient flooding protection: Cattenom site declared a significant safety-related event regarding flooding of the fuel oil tank room, partly owing to a loss of tightness of the waterstop seals. It was notified, that it was impossible to test the waterstop seals, which are a key part of the flooding protection [ASN 2011]. The problem of the seals has already been dealt with by a remediation action, but it shows shortcomings in the **safety culture**.

CONCLUSION

The French nuclear power plants we assessed show considerable deficiencies. Safety important equipment, for example the filtered venting system of the containment, is not seismically qualified. Flood protection shows a lot of shortcomings. The evaluations of robustness were not sufficiently reliable, and additional site-specific studies are needed. However under most unfavourable conditions the loss of heat sink and loss of electrical power scenarios could lead to core melt within a few hours.

EDF and the responsible authorities tried to outline protection measures in the future (about 2020), after the “Hard Core” and the FARN will have been implemented. However, to assess the hazard, the current situation is to be considered: an earthquake or flooding can occur any day.

The required improvements or a considerable part of them will probably be done during the ten-yearly outages as was practice until now. Even if the authorities try to speed this up, who can carry out this enormous amount of work reliably? The CSAs revealed that there are problems with outside contractors.

It is common understanding, that a second-generation reactor cannot be backfitted to the same safety level as a reactor that was designed to withstand severe accidents (see chapter 3). However, some of the backfitting measures required now as a result of the stress tests, EDF was already planning in the framework of the utility’s plan to receive the permit for 60-year reactor operation. Life time extension seems to be the next step foreseen for the old dangerous plants.

The Fessenheim NPP is the most vulnerable plant, and at the same time the plant with the highest risk. According to the IRSN, Fessenheim NPP is the only French NPP that should immediately improve seismic protection and to upgrade flooding protection. This situation calls for the only reasonable reaction: stop power operation immediately.

The same option applies for Gravelines NPP. There are six reactors at the coastal site, with insufficient flood protection, completely unprepared for a multi unit event, suffering ageing related problems, using MOX fuel resulting in an irresponsible risk.

The recent INES 2 event at Cattenom NPP should be understood as the last warning to stop power operation and checking the complete plant properly.

KEY RESULTS OF THE PEER REVIEW

EARTHQUAKE

The design basis earthquake (DBE) has been developed according to the French regulation, i.e. solely based on a deterministic approach for seismic hazard assessment. The review team recommends ASN to continue considering the implementation of probabilistic methods (PSHA).[34]

Many safety improvement measures have been identified by EDF and ASN. The most ambitious improvement measure is the deployment of a "hardened safety core" of the material and organizational measures. ASN will require integrating an electricity generating set and an emergency cool down water supply for each reactor in the "hardened safety core", which will be subject to more stringent requirements, particularly with respect to the earthquake and flooding risks. The hardened safety core will be based mainly on new equipment diversified form the existing one to prevent common cause failure. The definition of the “hardened safety core” is scheduled for June 2012.

According to the peer review team, the seismic instrumentation appeared to offer the potential of improvement to a state of the art concept.

ASN requested EDF to deepen the Seismic Margin Assessment (SMA), which was performed in a simplified way. The combination with earthquake induced flooding beyond design (dam failure, embankments failure) will also be considered. The peer review team appreciated that ASN will require a more systematic evaluation.

FLOOD

34 PSHA studies were not performed e.g. for Fessenheim, Gravelines and Cattenom.

The design basis flooding (DBF) has been adequately developed – but only according to the French regulation. ASN and IRSN stated that the current flood level calculation does not allow calculating of 10,000 year levels according international requirements. The peer review team recommends performing a comparative evaluation between the levels of DBF defined according to ASN requirements with the methodologies used in other European countries.

ASN holds the opinion that the requirements resulting from the complete reassessment of this risk, completed in 2007, could give (theoretically) a high level of protection against the risk of flooding. However, ASN observes that the steps to meet these requirements have not yet all been taken. ASN will require EDF to complete these protective measures until 2014.

The analysis performed by IRSN during the stress test review revealed cliff-edge effects close to review flood levels (DBF). The “hardened safety core concept” shall improve the flooding safety.

ASN will require that EDF conducts a study on the seismic resistance of the Canal d'Alsace embankment beyond design basis and to evaluate the consequences at Fessenheim NPP in case of embankment failure.

EXTREME WEATHER

The peer reviewers confirm the conclusion drawn by ASN that further studies need to be conducted to provide a complete and systematic design basis and safety margin assessment with respect to extreme weather conditions. Safety margins are estimated in a simplified way by expert judgment.

ASN requires EDF to conduct the analyses for those climatic phenomena which are related to flooding. The peer reviewers recommend including also tornadoes, heavy rainfall, extreme temperatures and the relevant combinations of extreme weather conditions in these complementary studies. The peer review team recommends furthermore considering extreme meteorological conditions in the required definition of the “hardened safety core”.

LOSS OF POWER AND LOSS OF UHS

Several issues were identified in particular for the 900MWe series plants (including Fessenheim and Gravelines) as possible cliff edge effects, but also for the other operating NPP designs.

The peer reviewers noted that measures were proposed to remedy these findings. Specific safety improvements have not been implemented yet, but ASN presented during the country visit the draft requirement containing requirements for implementation of the necessary measures. In addition to the hardened safety core, ASN identified about 40 different requirements that contain short-term and mid-term safety measures. The draft schedule for implementation of these measures is 2012 – 2018.

The vulnerability of the UHS has been highlighted by recent events of clogging and partial loss of the heat sink (e.g. at Fessenheim). No French reactor has an alternative ultimate heat sink. A situation with loss of UHS can currently be induced by a DBE or by flooding slightly beyond the DBF and will affect all units at a site. In those cases, the core could become uncovered in just a few hours. EDF started to reinforce the robustness of the UHS.

The peer review team noted that ASN already requested EDF to increase seismic resistance of equipment used to manage LOOP, SBO and LUHS situations. The seismic qualification will continue at several SSCs that are considered as measures to prevent or mitigate the core damage risks (SBO, LUHS).

The battery discharge time has been identified as the cliff edge effect for all reactors (loss of information in the control room and of the instrumentation and control). The reviewers observed that the battery discharge time by design is in the range of 1 hour. ASN requires EDF to significantly increase the autonomy of the batteries used in the event of total station blackout. The reviewers recommend ASN to also consider the recharging the batteries before their complete depletion.

SEVERE ACCIDENT MANAGEMENT

The SAMGs do not cover accidents in the spent fuel pool (SFP), and do not include multi-unit events. Furthermore, it was not assumed that an extreme external event can cause a severe accident. For this reason, many SAM related provisions were not seismically qualified. These include the venting filters, but also mobile equipment. ASN required general improvement of such equipment.

Furthermore the filtration efficiency if used on two reactors simultaneously (at Fessenheim and Gravelines) as well as the improvement in the filtration of iodine isotopes will be studied.

The fuel building is not designed to ensure containment in the event of a pressure rise following a release of steam owing to boiling of the SFP. It consists of a metal cladding roof and a thin concrete wall (about 30 cm).

To cope with severe beyond design basis accidents (BDBA), the above mentioned “hardened safety core” of material and organizational measures shall include also the emergency management rooms and all equipment and devices needed for emergency management (e.g. instruments for radiation protection). For accidents lasting longer than 24 hours, the creation of a FARN is foreseen. It will be composed by specialized crews and equipments. These crews consist of the licensee's employees based on 4 NPPs distributed in France.

According to the peer review team, the main improvements to be made to cope with severe accidents possibly affecting multiple units and caused by natural hazards have been pointed out by ASN. One recommendation of the peer review process is to guarantee their implementation. The reviewers consider the identified actions to be adequate. They have stated, that the consideration and implementation of these issues is important to be realised as soon as possible, apart from the PSRs, which are usually the reference for introducing new safety standards in France.

CRITICAL COMMENT

The peer review team has not assessed the current safety level of the French NPPs, but only the potential increased safety level which should have been achieved in 2018. Currently, there are several known shortcomings regarding the protection against earthquake and flooding as well as the known impossibility to cope with a severe accident especially in event of earthquake or flooding affecting all units of the site. The reviewers only described the weaknesses, but they do not present an overall assessment of all facts. This is necessary for politicians to be able to assess the risk.

Gravelines NPP for example is sited at the sea and houses six units (!). The Fessenheim NPP has considerable shortcomings regarding both seismic and flood protection. The Cattenom NPP also shows shortcomings regarding seismic and flooding events. The peer review team did not consider all safety issues that could trigger or aggravate an accident situation (e.g. ageing, use of MOX fuel, safety culture).

5 POTENTIAL IMPACTS OF SEVERE NUCLEAR ACCIDENTS IN EUROPE

This study discussed the results of the EU stress test for nuclear power plants. To give a more complete picture we present the potential impacts of severe accidents of the 12 selected reactors. For this we use results of the Austrian project flexRISK (Flexible tools for assessment of nuclear risk in Europe).

The project flexRISK studies the geographical distribution of the risk due to severe accidents in nuclear facilities, especially nuclear power plants in Europe. Starting with source terms and accident frequencies, the large-scale dispersion of radionuclides in the atmosphere is simulated for about 1,000 meteorological situations.

All severe nuclear accidents have one common feature: they took a course the safety considerations did not assess. This is due to the complexity of nuclear installations which makes a complete coverage of all accident situations impossible.

For each reactor an accident scenario with a large release of nuclear material – usually rather unlikely – was selected. To determine the possible radioactive release for the chosen accident scenarios the specific characteristics of each nuclear installation (e.g. retrofitting) were taken in consideration to the extent known.

Even though the possibility of a severe nuclear accident is estimated to be small – 1 in 10 million operation years (that is comparable to winning the lottery!) the damage caused is very large. Chernobyl disaster resulted in tens of thousands of people still suffering from the effects of the disaster (illnesses, relocation, contaminated farmland, meadows and forests, ...).

The frequency of occurrence of severe accidents assumed in flexRISK is derived from the calculation of the failure rates in all the imaginable event sequences. The figures provided by the operating companies come from probabilistic safety analysis (PSA), which, however, are not always based on comparable assumptions: some consider only accidents caused by failure of nuclear power plant components, the ageing of materials is difficult to include, others take accidents caused by external triggers into consideration (flooding, earthquakes, plane crash, ...). Human error is especially difficult to quantify. The estimated frequencies of severe accidents are therefore afflicted with high uncertainties (factor of 10 and more).

The characteristic of a severe accident is core damage and the release of radioactive material: Noble gases, iodine caesium are the most volatile substances which escape from the reactor core into the environment.

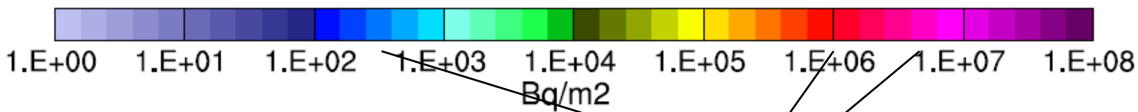
FlexRISK simulates the consequences of each accident for different meteorological conditions.

The dispersion of radioactive clouds as a consequence of serious accidents in nuclear facilities in Europe and neighbouring countries is calculated for selected accidents with varying weather conditions. Wind and rain determine which region will be affected to what extent by a release of radioactive material. In the flexRISK project from the infinite number of possible weather situations more than 2700 examples were selected from a 10-year period (2000-2009) according to a system that warrants an equal frequency of all times of day and all times of the year.

In addition 88 cases of the year 1995 were calculated, because 1995 was rather characteristic for the climate in Europe and this year had been analysed in an earlier project (Riskmap).

The meteorological data are taken from the European Centre for Medium Range Weather Forecasting.

For this study we used the results for the 88 cases of the year 1995 for the 12 selected reactors, for which the stress test reports are discussed in this study. Using the Lagrangian particle model FLEXPART concentrations of radionuclides in the air as well as their deposition on the ground were calculated and visualised in graphs. Some of these graphs (88 for each reactor) we show as examples. The total cesium-137 deposition per square-meter is used as an indicator of the contamination.



650 Bq/m^2 putting livestock into stables (Germany)

1000 kBq/m² removal of top soil layer (Germany)

7000 kBq/m² relocation considered (Germany)

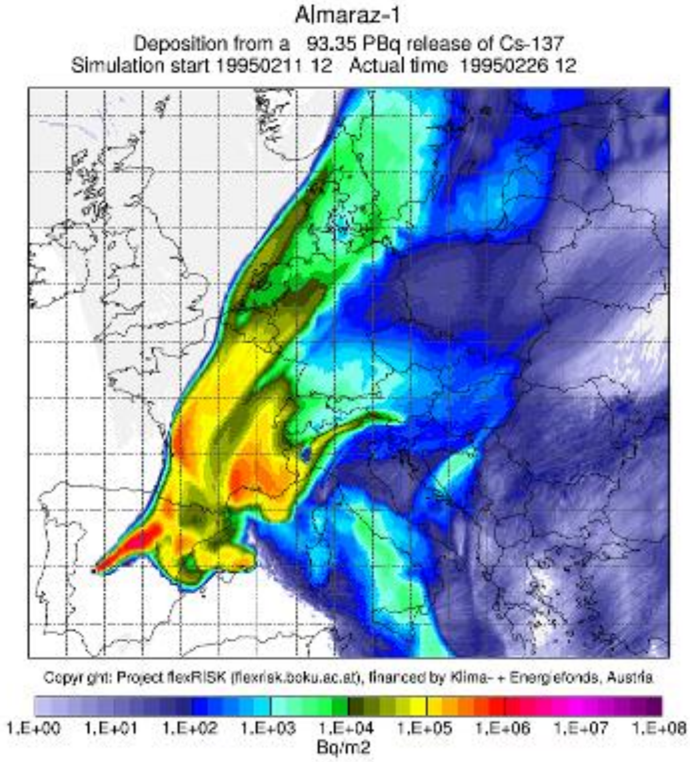
After the Chernobyl accident in the Soviet Union the following contamination limits were used:

- 37 – 185 kBq/m² was defined as a contaminated area; radiation monitoring was carried out in this area (restrictions in use of local food and stay outside were recommended) (estimated dose<1Sv/a)
- 185 – 555 kBq/m² people were allowed to leave the region (estimated dose 1- 5 mSv/a)
- 555 – 1480 kBq/m² relocation at a later time (estimated dose > 5 mSv/a)
- > 1480 kBq/m² immediate evacuation (estimated dose > 5 mSv/a)

The German radiation protection commission (SSK) has published catalogue of protection measures for severe accidents (SSK 2008), most indicators for measures are dose limits, but some are defined by soil contamination. These are used as illustration of the colour scale on the contamination maps.

The accident scenarios for the dispersion calculation are severe accidents with core melt and containment bypass or containment failure; the release rates for caesium are in the range of 20 to 65 % of the core inventory of caesium.

The project flexRISK is funded by the Austrian Climate and Energy Fund through the program “Neue Energien 2020”. Details on the project and the consortium can be found at http://flexrisk.boku.ac.at/.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Copyright: Project flexRISK (flexrisk.boku.ac.at), financed by Klima- + Energiefonds, Austria
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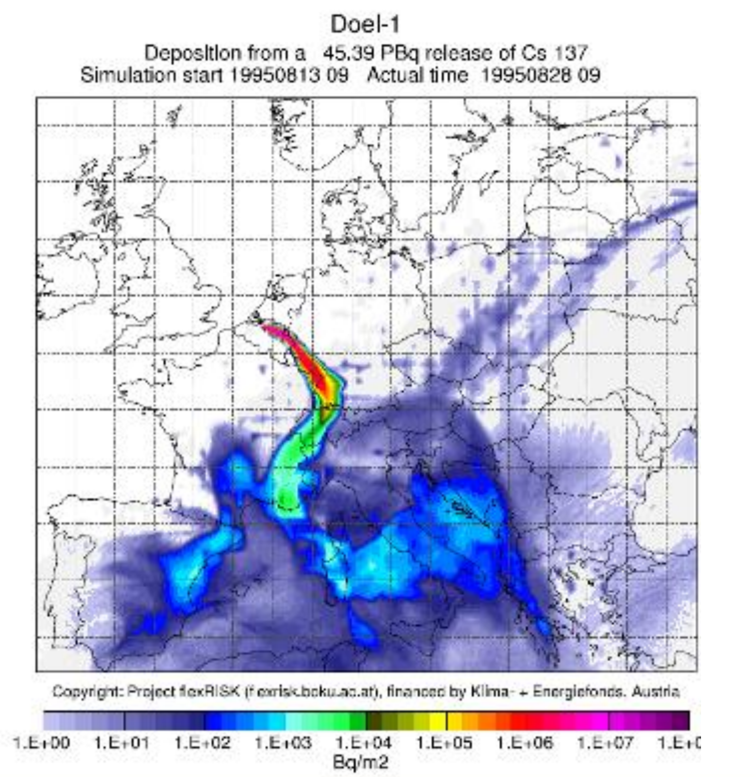
Bq/m2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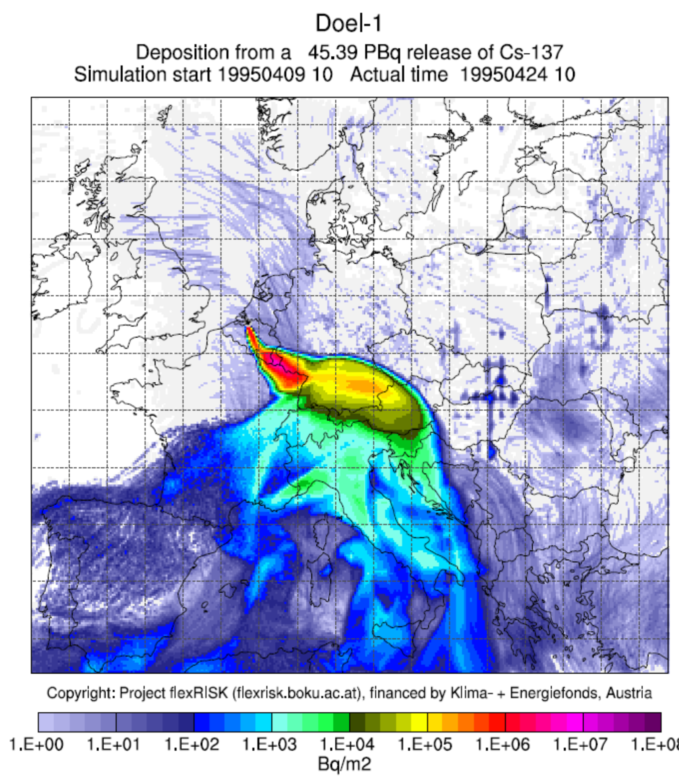
Bq/m2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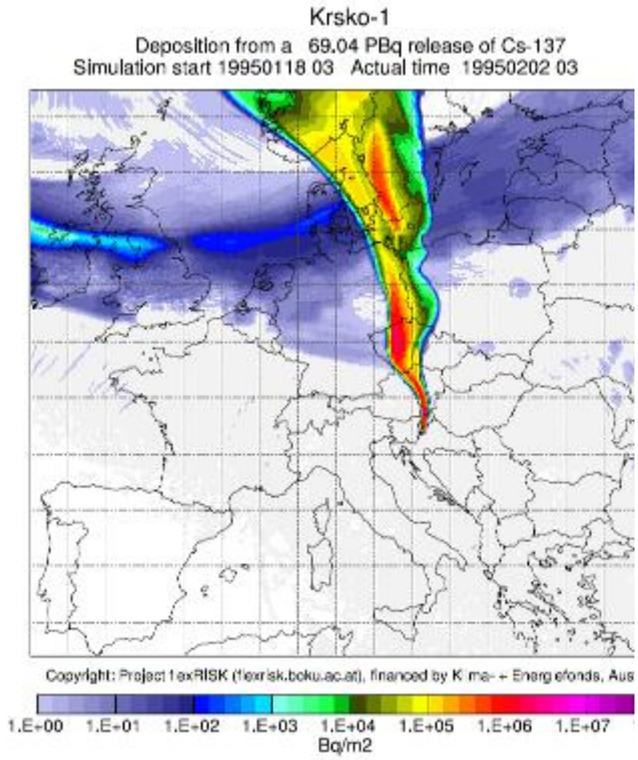
Bq/m2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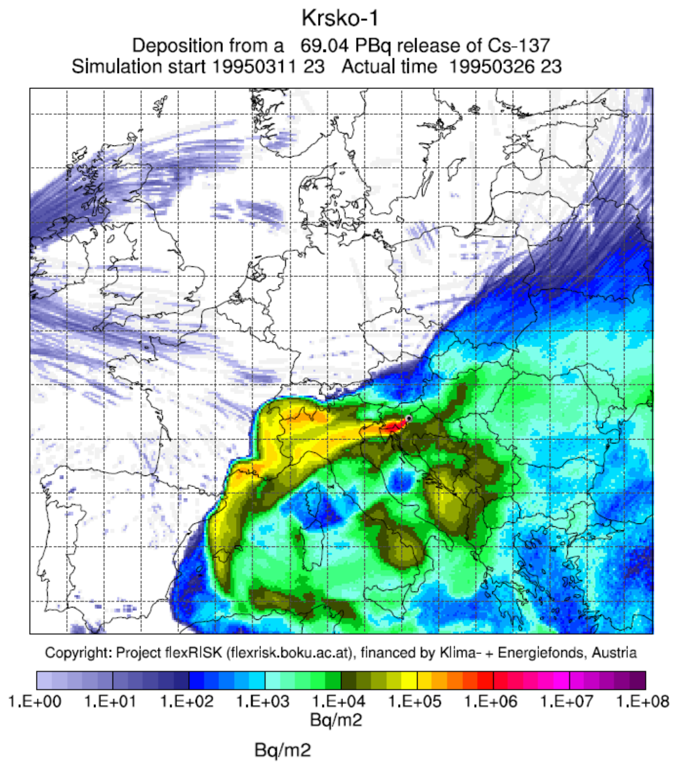
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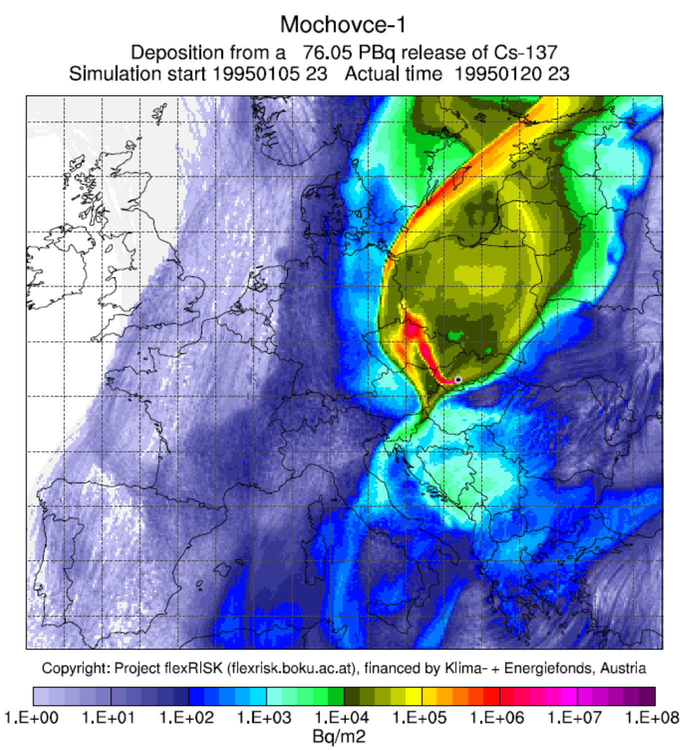
Bq/m2
Bq/m2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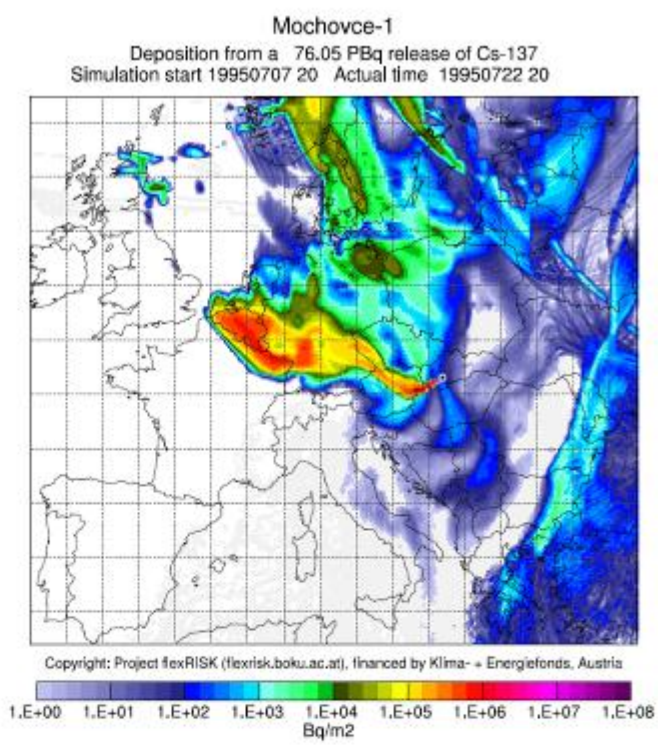
Bq/m2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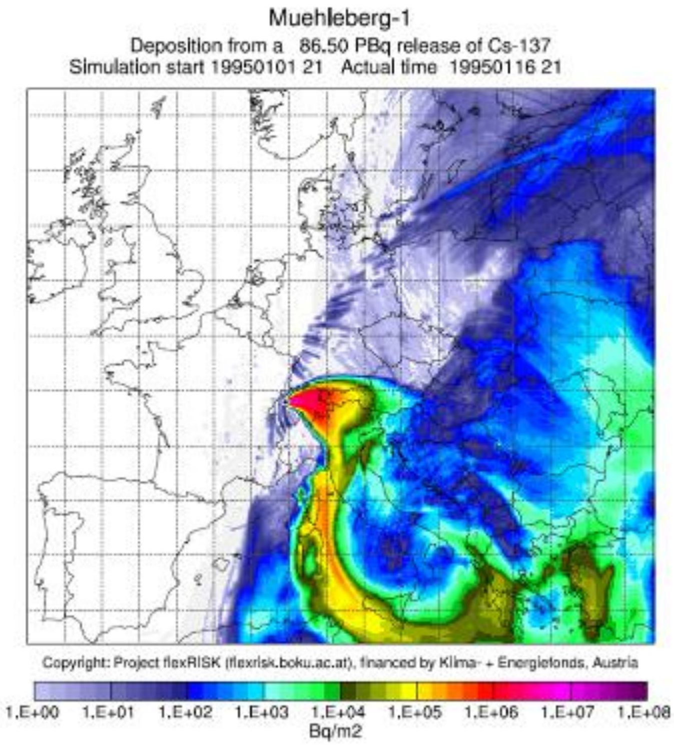
Bq/m2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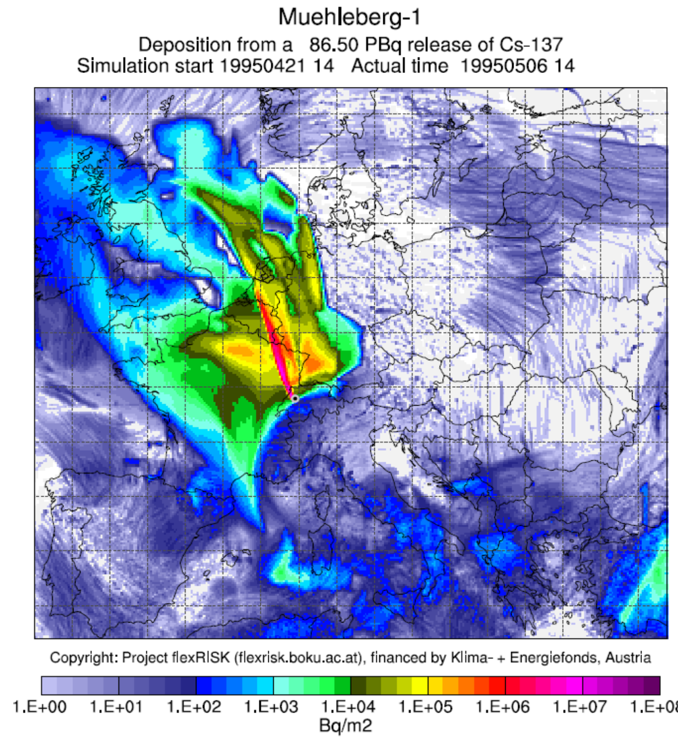
Bq/m2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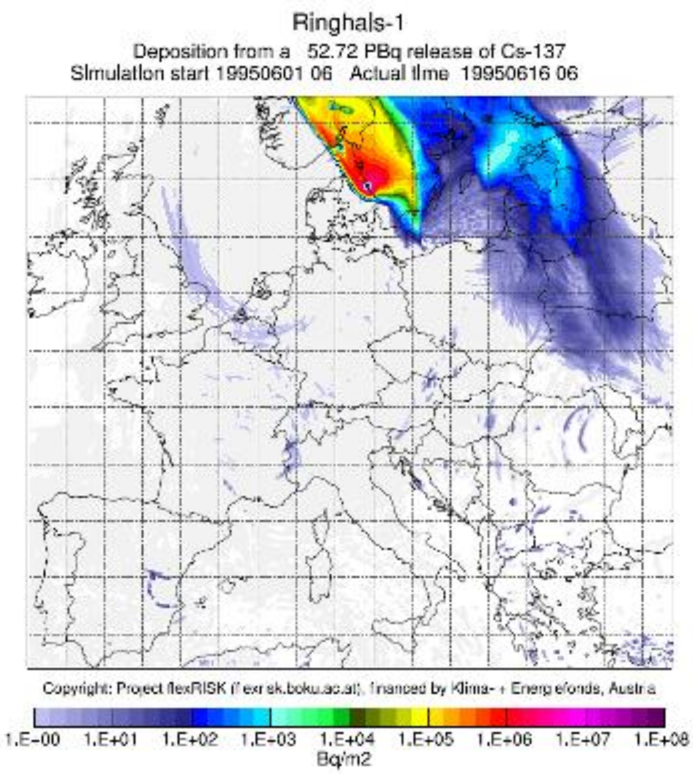
Bq/m2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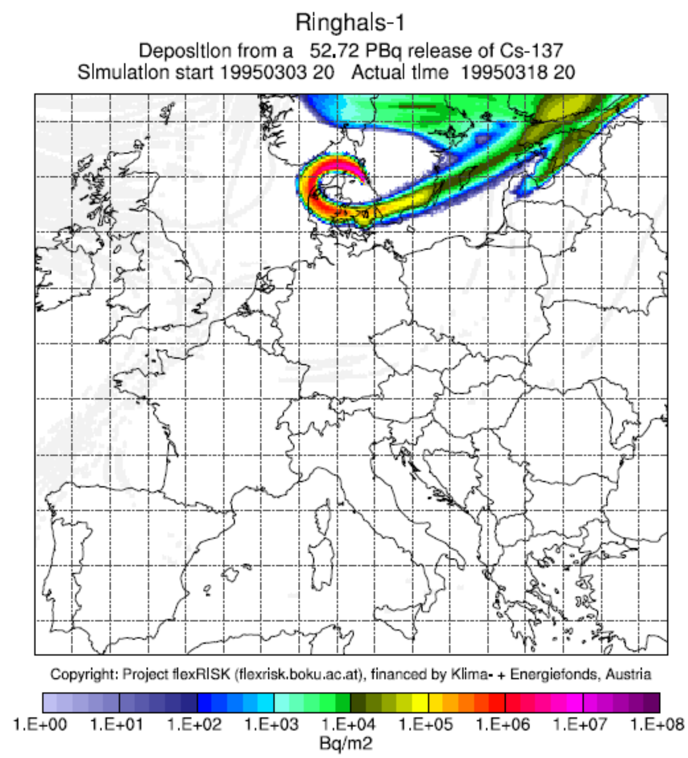
Bq/m2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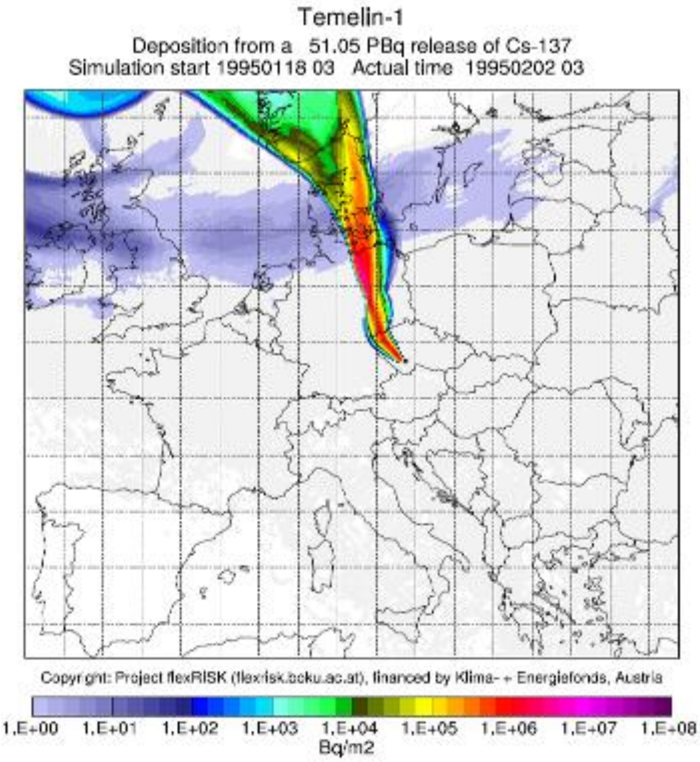
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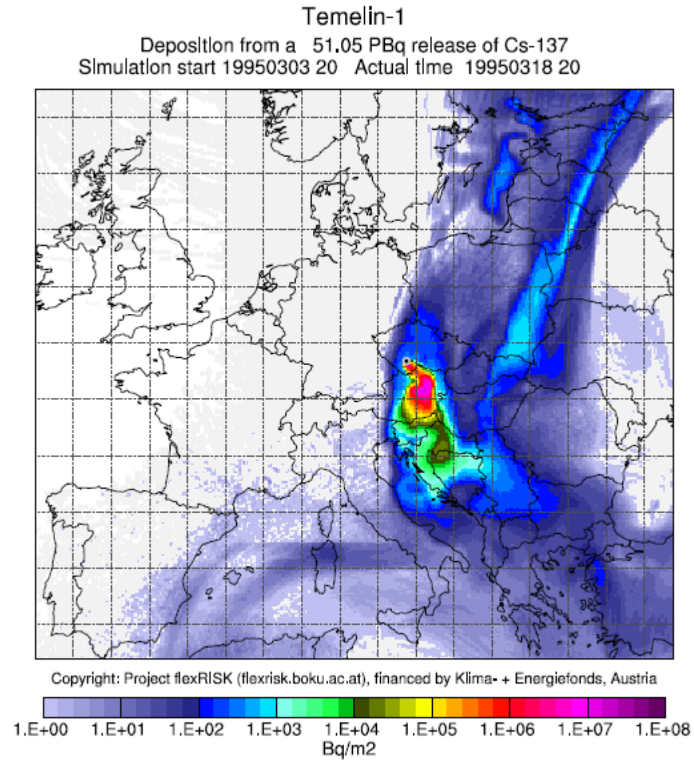
Bq/m2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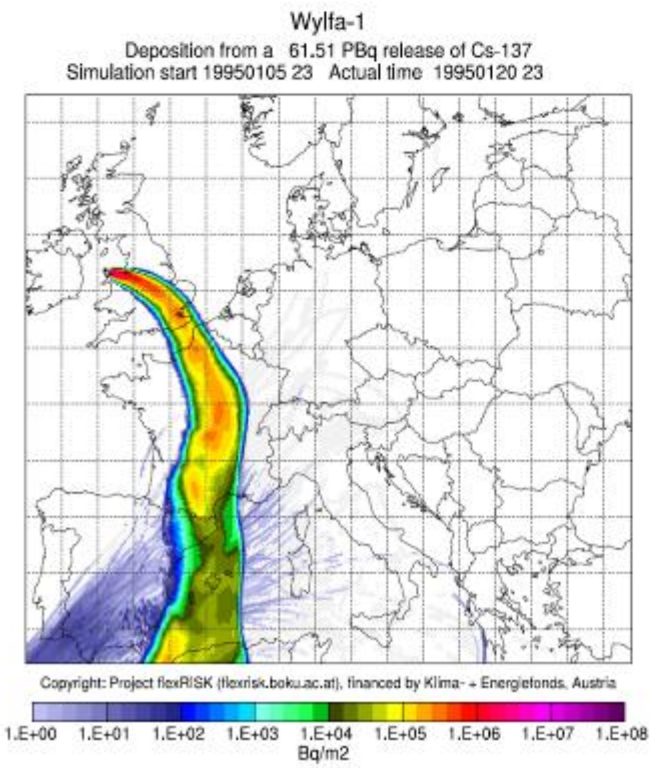
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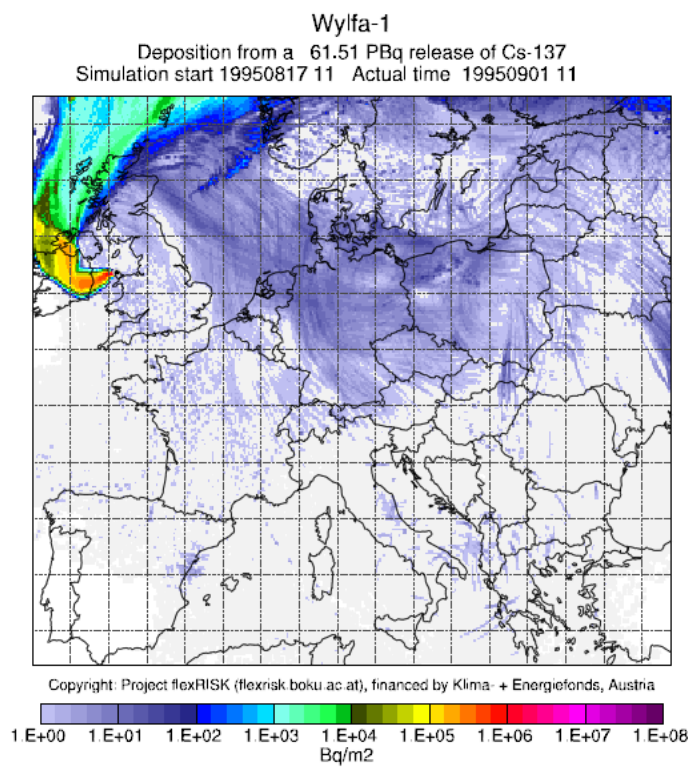
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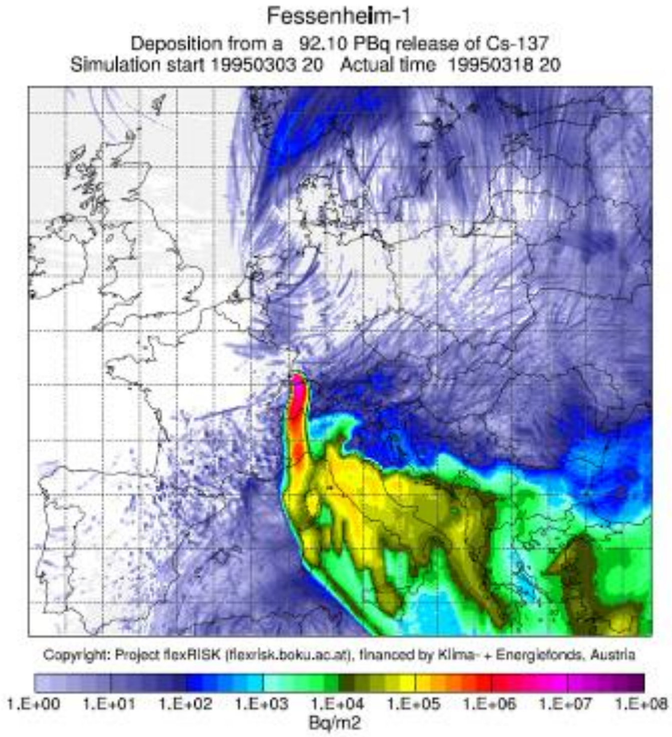
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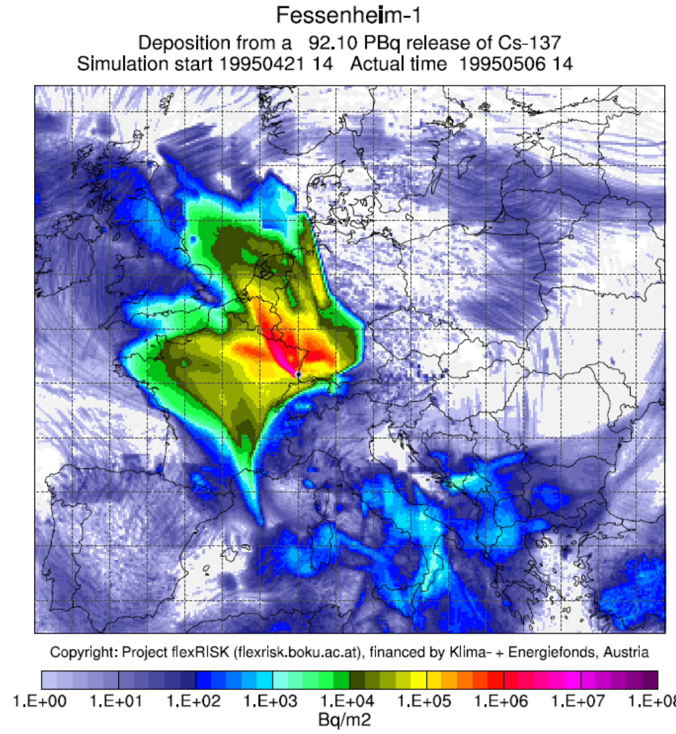
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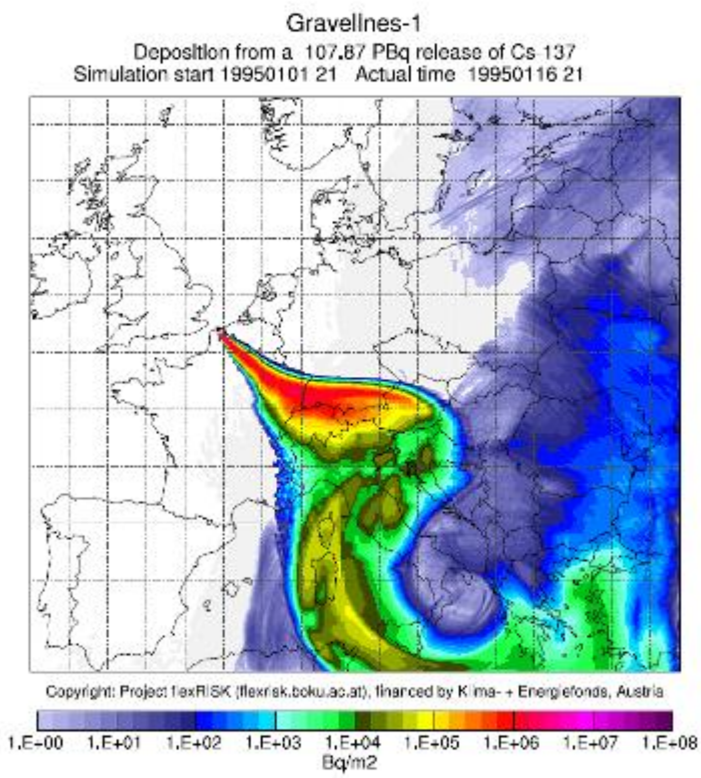
Bq/m2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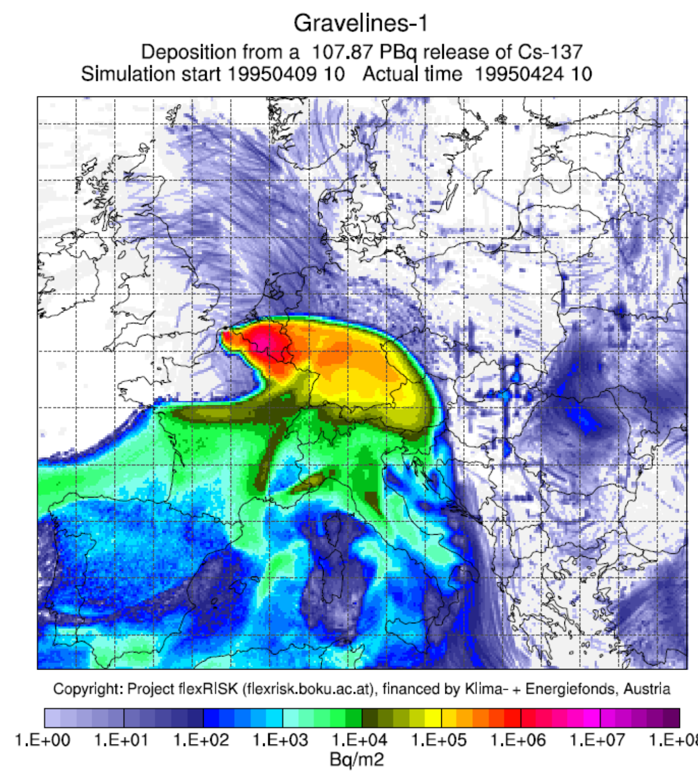
Bq/m2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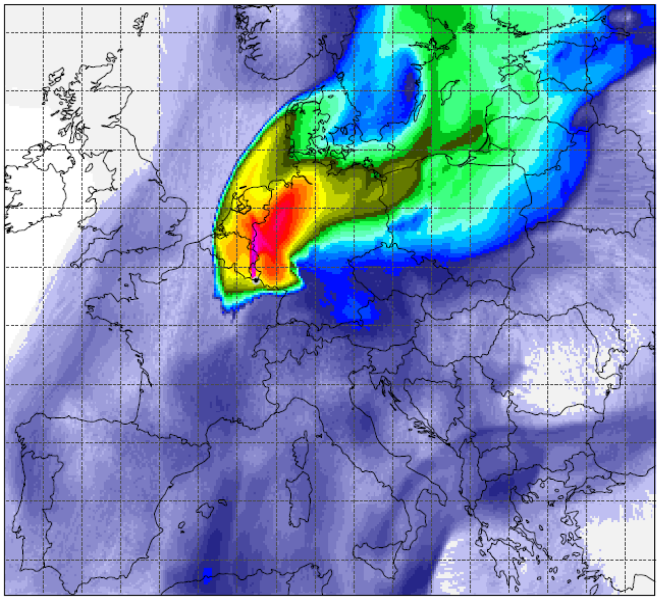

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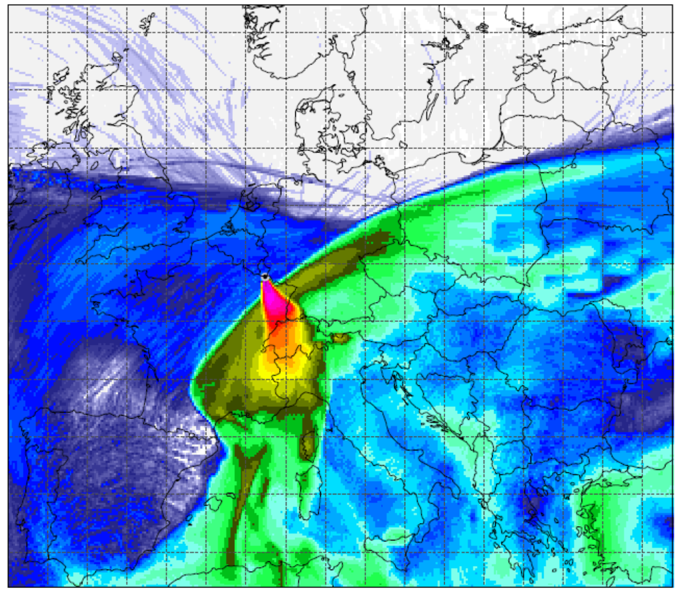

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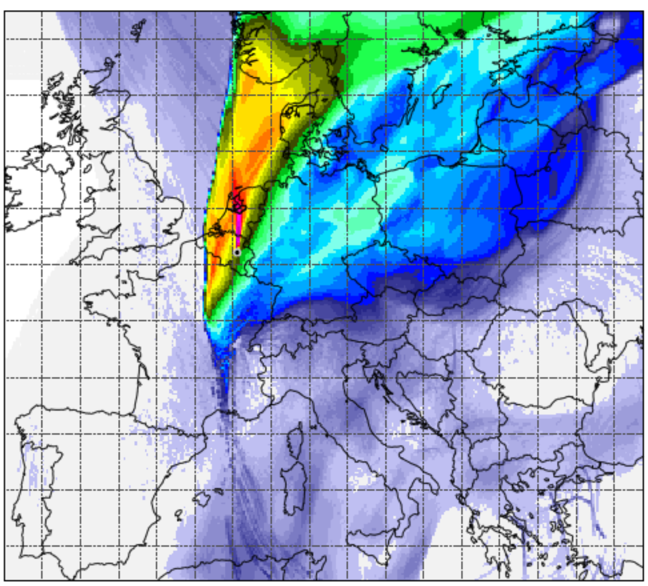

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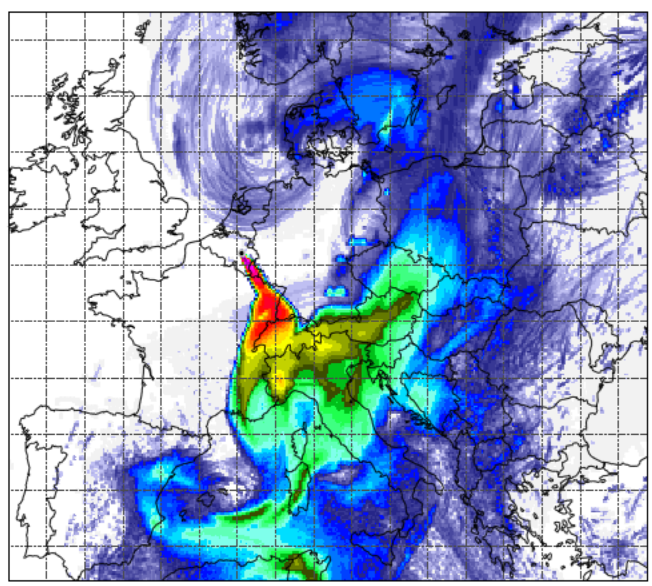

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Bq/m2

GENERAL DESCRIPTION OF REACTOR TYPES

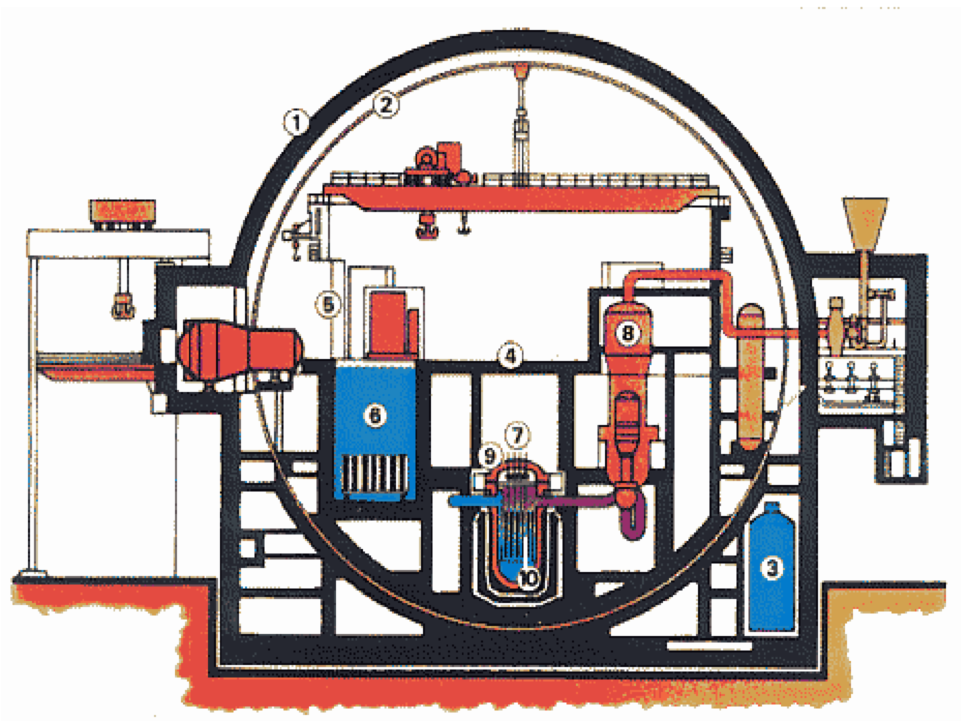
Boiling Water Reactor (BWR)

BWRs are cooled and moderated by water. The main difference to the PWR is that the water in the Reactor Cooling System (RCS) is boiling and the steam produced in the core is directly conducted to the turbine. The core is similar to a PWR core and is also confined in a vessel. In the upper part of Reactor Pressure Vessel (RPV), steam separator and drier are located. Only a small part of the coolant is converted to steam. The remainder is circulated by pumps in the bottom part of RPV.

Because the upper part of reactor pressure vessel is filled with the steam separator, the control rods have to be inserted from the bottom. Whereas PWR containments are generally designed to withstand the full pressure generated by a large LOCA, BWR containments depend on pressure suppression systems: The steam escaping e.g. after a pipe rupture is conducted into a heat sink (a water pool).

Pressurized Water Reactor (PWR)

Of all NPPs PWRs are the most frequently used type world-wide. In PWRs water is used as coolant and moderator simultaneously. The core, where heat is produced by nuclear fission is located inside a pressure vessel. The coolant through the core is circulating through a separate coolant circuit, which in most PWRs is completely confined in the containment. Pressure in the primary coolant system is high enough to prevent coolant from boiling. Integrity of primary system is a crucial point for safety of a PWR. Neutron flux and stress to the components is high and leads to embrittlement and fatigue. The primary circuit is divided into several loops with the corresponding number of SGs and pumps. The number of loops is different and reaches from 2 e.g. some Westinghouse PWRs to 6 in WWER-440 reactors.



1 primary containment prestressed concrete (2m)

2 secondary containment (steel)

3 accumulator tank

4 concrete shield

5 protection against missiles

6 water-cooled fuel pool

7 control rod drives

8 steam generator

9 reactor pressure vessel

10 reactor core:

The steam, which is necessary for driving the turbine, is generated in a special heat exchanger, the Steam Generator (SG), where the heat from the closed primary coolant circuit is transferred to a second coolant circuit (secondary or feedwater). The containment is supposed to isolate the primary system in case of accidents. Because of several penetrations into the containment the isolating function can be degraded e.g. by valve failures. Hydrogen explosions and pressure build-up can cause destruction of the containment under accident conditions.

WWER-TYPE REACTOR PLANTS

Water-cooled water-moderated power reactor (WWER) belongs to the most prevailing type of light water reactors PWR. OKB"GIDROPRESS" is the Russian developer of VVER-type reactor plants with a power range (from 300 to 1000 MWe)

Between WWER and "Western" PWR are differences both in design and materials used.

-WWER distinguishing features:

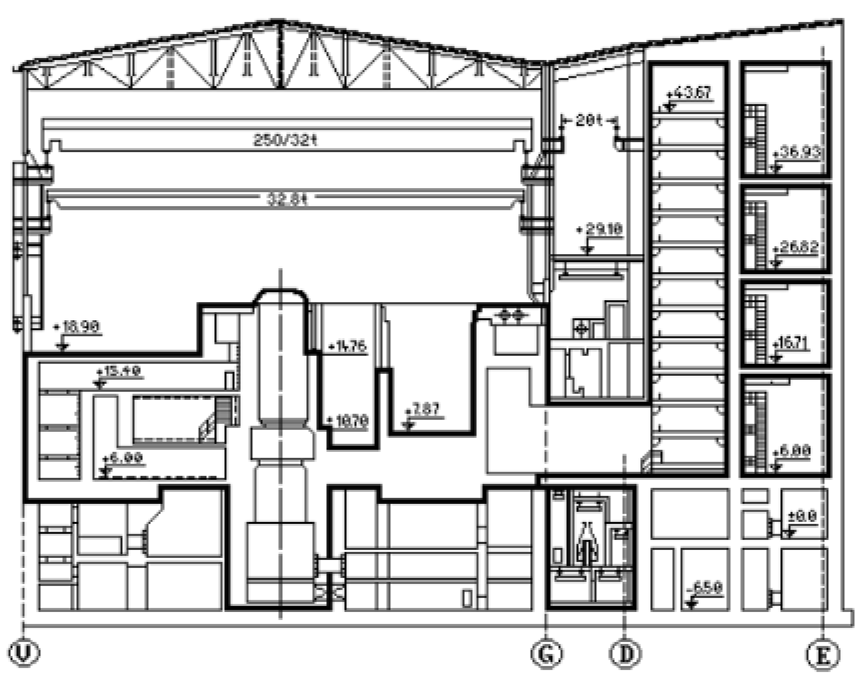
-Horizontal steam generators

-Hexahedral fuel assemblies

-No bottom penetrations in the VVER vessel

-High-capacity pressurisers

VVER 440/V213



Note: Thick line – external containment boundary

Figure 6: Vertical cross-section of reactor building

The VVER 440/V213 is a PWR of Russian design with 6 primary cooling loops. VVER 440 reactor plants are twin units, located in a common reactor building. The VVER 440/V213 is not equipped with a full pressure containment, which is a common feature of most pressurized water reactors. The confinement at VVER 440/V213 consists of compartments, which enclose the essential primary circuit components: steam generator, pipelines, pumps, shut off valves and reactor pressure vessel. The VVER 440/V213 confinement alone does not guarantee to hold back the radioactive steam from big leaks, but needs to condense the steam in the special pressure relief system (Bubbler Condenser). A failure of the relief system can cause the confinement to burst at its weakest point and radioactive material is released into the environment. In recent years studies on the behaviour during severe accidents were commenced. The results are to be used for the development and improvement of the SAMG - severe accident management guidelines. Safety analyses showed that in particular the Bubbler Condenser has very low or even no safety margins under certain conditions and for certain components . In case of EMO 3/4 improvements are planned (e.g. installing recombiners for hydrogen removal). [WENISCH et al 2009] The spent fuel pool is outside of the containment barrier in the reactor hall. If the reactor is in operation the SFP is covered.

Significant time margins are available for core cooling even in case of loss of electric power and loss of ultimate heat sink, because of the large thermal inertia due to low power and comparably large amount of water both in primary and secondary system, as well as a large volume of water inside in the pressure suppression system.

All VVER 440/V213 reactors are implementing severe accident management measures in connection to lifetime extensions. These measures concern hydrogen removal from the confinement and from spent fuel pool, as well as cooling of the reactor pressure vessel by flooding the reactor shaft with water from the bubbler condenser. Thus melting of the overheated core is cooled and shall be retained in the vessel.

These measures are a try to achieve safety objectives required by WENRA. **Safety objective 3** is an ambitious requirement, to control core melt accidents and reduce radioactive emissions to the environment. However it must be mentioned, that if the vessel fails the reactor cavity door is not designed to retain the molten core. Escape of a large release of radioactive substances is to be assumed.

VVER 1000

The VVER 1000 series is a 4- loop pressurized water reactor similar to the 4- loop Westinghouse rector. New Models achieve electric outputs > 1000 MWe. They are equipped with modern passive and active accident prevention features.

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ABBREVIATIONS

AC	Alternating Current
AFW	Auxiliary Feedwater
AHGNS	Ad Hoc Group on Nuclear Security
AHRS	Additional Residual Heat Removal System
BDBA	Beyond Design Basis Accident
BMU	Bundesministerium für Umwelt
BWR	Boiling Water Reactor
CDF	Core damage Frequency
CSN	Consejo de Seguridad Nuclear
DC	Direct Current
DBE	Design Basis Earthquake
DBF	Design Basis Flood
DG	Diesel Generator
ECCS	Emergency Core Cooling System
ECR	Emergency Control Room
EDG	Emergency Diesel Generator
ENSREG	European Nuclear Safety Regulators Group
ESW	Essential Service Water
FANC	Federal Agency for Nuclear Control
IAEA	International Atomic Energy Agency
MCP	Main Cooling Pump
MCR	Main Control Room
MOX	Mixed Oxide
MWe	Megawatt electric
MWt	Megawatt thermal

NCM	Non-Conventional Means
NEK	Nuklearna Elektrarna Krško
NPP	Nuclear Power Plant
ONR	Office of Nuclear Regulation
PGA	Peak ground acceleration
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RHWG	Reactor Harmonization Working Group
RPV	Reactor Pressure Vessel
RSK	Reactor Safety Commission
SAMG	Severe Accident Management Guide
SBO	Station Black Out (total grid & EDG failure)
SFP	Spent Fuel Pool
SG	Steam Generators
SIS	Safety Injection System
SL	Seismic Level
SNSA	Slovenian Nuclear Safety Administration
SO	Safety Objectives
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SUJB	State office for nuclear safety CR
UHS	Ultimate Heat Sink
UJD	Nuclear regulatory authority of the Slovak Republic
WENRA	Western European Regulators Association