Progress Report

Complementary Safety margin Assessment

August 15, 2011

EPZ Disclaimer

This Progress Report is solely meant to inform the regulator about the status of the Complementary Safety margin Assessment (CSA) being performed by EPZ upon request of the Netherlands Ministry of Economics, Agriculture and Innovation (EL&I).

Information in this document is preliminary and should be used for its purpose only.

No warranty, either express or implied, is made with respect to the information contained herein. EPZ retains its right to change the content upon its own discretion. Only when a final document of CSA at 31th of October 2011 is supplied for use in the public domain, EPZ will make sure that no changes are further needed.

1. Executive summary

On june 1st 2011 the Ministry of Economics, Agriculture and Innovation (EL&I) requested the Elektriciteits Produktiemaatschappij Zuid-Nederland EPZ (EPZ) to perform a targeted reassessment of the safety margins of NPP Borssele, the so-called "Stress test". After that request EPZ formally started a project named "Complementary Safety margin Assessment" (CSA) and established an experienced project team. To ensure the necessary expertise and resources to generate the CSA report, experienced external parties from the beginning take part in the project. In general they are involved in the execution of the analyses of the different issues, whereas EPZ employees are responsible for supervision and reviewing.

The CSA project is divided into 2 phases. Phase 1 includes the performance of the basic analyses of the main issues comprising the three elements: design base, evaluation of the margin in the design base and assessment of the margins "beyond design" and the delivery of the draft reports to EPZ. In Phase 2, starting mid August, the reports produced in Phase 1 will be reviewed, reported results discussed and, if necessary, additional analyses performed. The Licencee Final Report will be redacted, reviewed and approved. In the Final Report all results of the CSA will be reported.

EPZ is requested to report its progress on August 15, 2011 to the approved authorities.

With the release of this Licensee Progress report Phase 1 has been finished.

EPZ is confident that the Licensee Final Report can be completed before 31 October 2011 and that the report will meet the high demands on quality and integrity.

2. Introduction

Considering the accident at the Fukushima Nuclear Power Plant in Japan, the European Council of March 24th and 25th declared that "the safety of all EU nuclear plants should be reviewed, on the basis of a comprehensive and transparent risk assessment ("stress tests") (See Annex 1).

For now the "stress test" is defined as a targeted reassessment of the safety margins of all European nuclear power plants.

This reassessment will consist on the one side of an evaluation of the response of a nuclear power plant when facing a set of extreme situations and on the other hand of a verification of the preventive and mitigative measures that have to ensure the safety of the plant.

The licensee has the prime responsibility for safety. Hence, it is up to the licensee to perform the reassessments, and to the regulatory bodies to independently review them.

The "stress test" will focus on extreme natural events like earthquake and flooding but will also look for the consequences of loss of safety functions if the situation is provoked by indirect initiating events, for instance large disturbance from the electrical power grid impacting AC power distribution systems or forest fire, airplane crash. Furthermore disturbances that are caused wanton have to be taken into consideration.

The "stress test" must lead to insight into severe accident conditions and how NPP Borssele reacts, also if the emergency measures provided for that situation, will fail. This means that for the determination of the safety margins a deterministic approach is chosen. The intention is that an ever more serious threat (for example, an increasingly higher tidal wave or heavier earthquake) is assumed, and that will be determined how NPP Borssele and safety management system respond to that and to what level of threat the safety systems work adequately. For further evaluation and taking any measures it is of course important to know how likely it is that such an event occurs. This information will also be reported.

The "stress test" must lead to:

- how NPP Borssele and the safety management system react in ever more serious accidents and in which protective measures are supposed to be progressively defeated
- indication of the weak points of the installation and the safety management system
- any potential for modifications to improve the weak points.

The aim of this Progress Report is to give insight into the methodology used by the Nuclear Power Plant Borssele in performing the "stress test", to provide a table of contents of the Final Report and to report about the progress made in performing the risk assessment.

3. Nuclear safety Nuclear Power Station Borssele

3.1 General safety policy

Within EPZ's nuclear power plant, nuclear safety has overriding priority. It is for this reason that EPZ has a nuclear safety policy which is formalized through various policy statements. Generally speaking, this policy implies that all actions have the intentional purpose of minimizing exposure to, and maximizing protection against the dangers of radiation, both for individuals and the environment. All this is achieved by setting up and maintaining an effective defense mechanism against radiological hazards.

To ensure and improve its nuclear safety, EPZ uses two basic principles which complement each other and have a partial overlap. Both principles are applicable to the technical aspects of EPZ's nuclear power plant (NPP) as well as to the attitude of EPZ's organization, its management and individual employees.

The first principle is "defense in depth". Hereby several different levels of protection are applied in design, construction, operation and decommissioning. The principle results in the presence of multiple, often diversely accumulated, (physical) barriers and several provisions which are complementary and/or have a (partial) overlap. Technically this principle can be seen in, for example, the presence of several graded (safety) systems which independently fulfill the same (safety) function. In the work processes this principle is recognized by the way the NPP is operated. This way of operating is highly reliable. Error prevention has great attention. Sorting out the minor problems results in preventing larger problems. The principle of "defense in depth" can be seen in the fact that the prevention of nuclear accidents comes first and has the highest priority. By (changes in) design and ways of operating the NPP it is, with high reliability, assured that enforceable measures are taken to prevent nuclear accidents and significant releases of radioactivity. Because in principle such events cannot be totally excluded, mitigating facilities to minimize the consequences are also arranged. However, prevention is more important than mitigation.

The second principle is called "Safety management and Safety Culture". This principle means that management and individual employees are at all times aware of the aspects that are important for the nuclear safety. Apart from this awareness there is the expectation that individuals apply common sense on this subject.
Within EPZ both the awareness of nuclear safety and the related actions are present at all organizational levels. EPZ considers the attitude, the way of thinking and the alert and judicious acting of each employee as a major contributing factor to nuclear safety. For this reason continuously measures are taken to ensure and promote the proper behaviour of the management and individual employees.
With regard to nuclear safety, work practices and attitude and behaviour of employees, management frequently propagate what is considered important. Furthermore, ongoing specific training is provided to EPZ's staff to improve work practices like "self and peer checking", "independent verification" and "questioning attitude". A specific aspect in the EPZ's "safety culture / safety management" is that non conformaties are used as a basis for continous improvement. Reporting of non conformaties is promoted by a "no blame" culture.

EPZ is a member of the World Association of Nuclear Operators (WANO). As part of this membership, EPZ participates actively in a peer review program, which means that WANO peer members are invited every four to six years for a full scope peer review of the nuclear safety of EPZ's NPP. One to two years after the peer review, WANO is invited for a peer review follow-up to check the progress in the implementation of the recommendations as defined during the peer review. Significant Operating Experience Reports (SOER), written by WANO in reply to reported incidents in other NPP's, are used by EPZ as an important source of experience, and usefull lessons learned are implemented.

In conclusion can be stated that the principle of "defense in depth" is necessary because a unforeseen release of radioactivity can never be excluded with an absolute certainty. The presence of several different barriers results however in the situation that the chance that even simultaneous single failures can lead to such an event is as low as possible. By maintaining all barriers in an optimal condition the chance for a serious release of radioactivity is reduced to a minimum. Besides knowledge, an adequate and proactive attitude, way of thinking and acting is needed to maintain these barriers at all times (safety culture).

For a better insight in, and understanding of the basic principles of nuclear safety an internal policy statement on nuclear safety was put together. This policy is made accessible to any employee by means of the company intranet and all training for the employees is based on it. The three starting points of this policy are:

Safety is first priority

- Nuclear Safety has the highest priority and has an overriding priority over electricity production;
- Nuclear Safety is visibly present in all EPZ's activities;
- EPZ continuously develops and promotes its safety culture

Safety is a pursuit of excellence through continuous improvement

- EPZ does comply with the regulations and licensing requirements for its NPP;
- EPZ applies the most recent insights in its activites and and compares its safety level with best practices and the most recent international standards and guidelines of i.e. WANO and the International Atomic Energy Agency (IAEA);
- EPZ learns from previous experiences through the evaluation of internal and external events and continuously implements improvements;
- In EPZ's NPP there is a maximum system availability; deviations are brought back to a minimum;
- EPZ's attention is focused on solving deviations and problems; not on pointing out the culprits;
- EPZ strives for a high level of knowledge, skills and technique.

Safety is a proactive attitude

- EPZ's core values define corporate and individial behaviour and these values are adressed to each other throughout the organisation;

- In EPZ's work activities the risks are assessed and minimised, and work is organised in such a way that the chance that an error will lead to an incident is as small as possible;
- EPZ transparently gives account for the safety of its power plant systems and the ways of working;
- EPZ actively invites (international) institutions and peers for inspection and activily cooperates in these inspections.

3.2 Periodic safety review

3.2.1 Methodology

Every ten years an extensive safety evaluation is performed on nuclear safety and radiation protection. There are four main provisions to be evaluated:

- Technical
- Organizational
- Personnel
- Administrative.

The evaluation focuses on nuclear safety and radiation protection. The objective of a 10 yearly safety evaluation is to prove by a comprehensive assessment that the design basis and the safety documentation remains valid; that the arrangements in place to ensure the plants safety remain valid and effective until the next 10 yearly safety evaluation and to verify the extent to which the plant conforms to current national and international safety standards and practices. The goal of the evaluation is to improve the design of the plant and the operation of it, so that the nuclear safety and radiation protection performance will increase. This means that the plants design is as far as possible in accordance with the highest technical design levels for modern nuclear power plants and is operated in line with the latest safety guidelines and best practices. The requirement for performing every ten years an evaluation is according to the plants nuclear license.

A ten yearly safety evaluation starts with setting up a frame work to define the scope of the evaluation. This frame work consists of the latest developments on national and international nuclear regulation and standards but also on internal and external experiences on plant design, maintenance and operations. The licensing basis is compared against national and international developments in nuclear safety and radiation protection.

For the evaluation, experts of the plant are employed, and also external experts are involved. The evaluation period takes several years. The result of the evaluation assessment is a list of evaluation points that could lead to potential measures for improvement. This potential measures are reviewed using techniques like Probabilistic Safety Assessment (PSA) and As Low As Reasonably Achievable (ALARA) to define a final list of required improvements (cost-benefit analyses in terms of core damage frequency and radiation dose).

The next step in the process is the implementation of these measures for improvement. Usually an important part of these measures consists of technical (design) changes. But also changes in the way of operating and maintaining the plant can be the result. At the end documents like procedures, instructions, drawings and PSA are updated in accordance with the modifications.

During the complete evaluation the national authorities are extensively involved.

3.2.2 Results of previous safety reviews

The first safety evaluation was performed in the eighties of the last century. At that time focus was on post Three Miles Island (TMI) issues. This resulted in the following measures:

1. Adding of two independent reserve safety systems to the existing ones. These two extra systems (primary leakage control and secondary heat removal) are installed in a bunkered building, resistant to external hazards like flooding, explosions and earthquake. These reserve systems are supported by a new independent emergency power system (bunkered grid 2)
2. The safety related instrumentation was replaced by new instrumentation resistant to accident conditions (loss of coolant accident)
3. A meteo system was introduced
4. New emergency operating procedures were introduced.

The second safety evaluation was performed in the nineties and resulted in a major backfitting project which was carried out in 1997. The most important modifications are listed below.

1. Complete separation of the redundant parts of the emergency core cooling system
2. New emergency power system (main grid 1)
3. New primary safety valves (for feed and bleed operation)
4. Extensive earthquake measures
5. Fire safety modifications like improvement of the fire compartments and adding automatic extinguishing systems (inergen gas and water mist)
6. Complete exchange of the main control room
7. Building of a emergency control room
8. Installation of a filtered containment venting system
9. Installation of an emergency core cooling and spent fuel heat removal system (bunkered)
10. Passive hydrogen recombination equipment in the reactor building
11. Building a plant specific full scope simulator
12. Introduction of the severe accident management guidelines.

The third safety evaluation started in 2001. The most important modifications (2007) are listed below. Most of these modifications are for improvement of equipment for operating the plant in accident conditions (beyond design).

1. Diminishing of the dependency of the emergency power system (bunkered grid 2) on the external 10 kV supply
2. Installation of detectors and igniters for protection against external explosive gas clouds
3. Installation of an additional pump in the spent fuel pool cooling system resistant to external hazards
4. Increase of the autonomy (to 72 hours) during external events by the ability of connecting the bunkered water basins
5. Increase of the diesel fuel stock for the emergency power system (bunkered grid 2)

6. Improvement of the protection of the control room against toxic gases
7. Increase of the elevation of the air entrances for the emergency diesels of the emergency power system (bunkered grid 2)
8. Improvement of the protection against the drop of a spent fuel transport container
9. Improvement of the fire extinguishing means after an airplane crash (crash tender)
10. Installation of a second emergency decay heat removal pump, protected against external hazards
11. Installation of a connection for direct external water injection into the steam generators for accident management
12. Implementation of means for active opening of hatches inside the containment to improve mixing of the containment atmosphere in case of local high hydrogen concentrations.
13. Extension of the emergency procedures and severe accident management guidelines for non-power operation
14. Improvement of the control of the qualification of electrical and instrumentation equipment
15. Extension of the system of safety performance indicators
16. Improvement of configuration control and archiving of technical documents.

At this moment the fourth ten yearly safety evaluation (2003 – 2013) is in progress. An important issue of this evaluation is a comprehensive license renewal.

4. Complementary Safety margin Assessment NPP Borssele

The Complementary Safety margin Assessment (CSA) performed by the Nuclear Power Plant Borssele (KCB) is based both on the letter from the Ministry of Economics, Agriculture and Innovation to EPZ on this subject (see paragraph 2 and Annex 1) and on the specifications mentioned in Safety Annex I issued by the European Nuclear Safety Regulators Group (ENSREG) (see paragraph 4.1 and Annex 2).

4.1 ENSREG EU "Stress tests" specifications (See Annex 2)

The existing safety analysis for nuclear power plants in European countries covers a large variety of situations. The technical scope of the stress tests has been defined considering the issues that have been highlighted by the events that occurred at Fukushima, including combination of initiating events and failures. The focus will be placed on the following issues:

a) Initiating events
- Earthquake
- Flooding

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

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

c) Severe accident management issues

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

b) and c) are not limited to earthquake and tsunami as in Fukushima: flooding will be included regardless of its origin. Furthermore, bad weather conditions will be added. Furthermore, the assessment of consequences of loss of safety functions is relevant also if the situation is provoked by indirect initiating events, for instance large disturbance from the electrical power grid impacting AC power distribution systems or forest fire, airplane crash.

The review of the severe accident management issues focuses on the licensee's provisions but it may also comprise relevant planned off-site support for maintaining the safety functions of the plant. Although the experience feedback from the Fukushima accident may include the emergency preparedness measures managed by the relevant off-site services for public protection (fire-fighters, police, health services….), this topic is out of the scope of these stress tests.

The approach should be essentially deterministic: when analysing an extreme scenario, a progressive approach shall be followed, in which protective measures are sequentially assumed to be defeated.

The plant conditions should represent the most unfavourable operational states that are permitted under plant technical specifications (limited conditions for operations). All operational states should be considered. For severe accident scenarios, consideration of non-classified equipment as well as realistic assessment is possible.

All reactors and spent fuel storages shall be supposed to be affected at the same time.

Possibility of degraded conditions of the site surrounding area shall be taken into account. Consideration should be given to:

- automatic actions;
- operators actions specified in emergency operating procedures;
- any other planned measures of prevention, recovery and mitigation of accidents.

Three main aspects need to be reported:

- Provisions taken in the design basis of the plant and plant conformance to its design requirements;
- Robustness of the plant beyond its design basis. For this purpose, the robustness (available design margins, diversity, redundancy, structural protection, physical separation, etc) of the safety-relevant systems, structures and components and the effectiveness of the defence-in-depth concept have to be assessed. Regarding the robustness of the installations and measures, one focus of the review is on identification of a step change in the event sequence (cliff edge effect) and, if necessary, consideration of measures for its avoidance.
- any potential for modifications likely to improve the considered level of defence-in-depth, in terms of improving the resistance of components or of strengthening the independence with other levels of defence.

In addition, the licensee may wish to describe protective measures aimed at avoiding the extreme scenarios that are envisaged in the stress tests in order to provide context for the stress tests. The analysis should be complemented, where necessary, by results of dedicated plant walk down.

To this aim, the licensee shall identify:

- the means to maintain the three fundamental safety functions (control of reactivity, fuel cooling, confinement of radioactivity) and support functions (power supply, cooling through ultimate heat sink), taking into account the probable damage done by the initiating event and any means not credited in the safety demonstration for plant licensing;
- possibility of mobile external means and the conditions of their use;
- any existing procedure to use means from one reactor to help another reactor;
- dependence of one reactor on the functions of other reactors on the same site.

As for severe accident management, the licensee shall identify, where relevant:

- the time before damage to the fuel becomes unavoidable. For Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR), if the core is in the reactor vessel, indicate time before water level reaches the top of the core, and time before fuel degradation (fast cladding oxidation with hydrogen production);
- if the fuel is in the spent fuel pool, the time before pool boiling, time up to when adequate shielding against radiation is maintained, time before water level reaches the top of the fuel elements, time before fuel degradation starts.

4.2 The approach of the CSA project

4.2.1 Project execution

The basis for the project execution of the CSA project is the letter of EL&I of June 1st (see Annex 1) and especially the document of ENSREG (see Annex 2).

This implies at first that three elements will be comprised in the CSA:

- the design base
- evaluation of the margins in the design base
- assessment of the margins “beyond design”; how far the beyond design envelope can be stretched until accident management provisions (design and operation) cannot prevent a radioactive release to the environment that requires mitigative actions to protect the general public.

The latter element is the “stress element “in the CSA.

Secondly the ENSREG document stipulates that a number of issues should be evaluated (see pragraph 4.1). Based on these issues the work has been divided into modules and made up the basis for the initial Work Breakdown Structure of the project. Of selected modules a report will be generated. The information gained within the modules will be integrated in the Final Report.

To prevent that time consuming evaluations and analyses would appear too late in the project, EPZ had set itself the task that around mid August the basic analyses of the main issues should be finished, comprising all the three elements mentioned before: design base, evaluation of the margin in the design base and assessment of the margins “beyond design”. This is Phase 1 in the project, which is finished according to the planning.

From mid august Phase 2 of the CSA project will be executed, with the following main activities:

- Systematically reviewing of the reports of the separate modules that have been produced in the first phase
- Discussing and evaluating of possible (combination of) issues that have not been evaluated in the first phase
- Execution of complementary evaluations and analyses, especially on “beyond design” margin data which were generated in phase 1
- Systematically reviewing the total report to assure interrelated style and consistency of the modules
- If necessary execution of complementary “second opinion” on specific issues
- Final editing of the report.

EPZ is confident that the Licensee Final Report can be completed before 31 October 2011 and that the report will meet the high demands on quality and integrity.

4.2.2 Project organization

EPZ established an experienced project team, lead by a project manager and supervised by a Steering Committee. In the Steering Committee members from outside the nuclear environment and members from outside EPZ ensure the independency of the assessment. The EPZ Technology group is responsible for the analyses, the reviews, the results and in general for the technical quality of the report.
The Head of Nuclear Power Station Borssele (HKCB) will, in his responsibility for nuclear safety, execute an independent review on the report.

To ensure the necessary expertise and resources to generate the CSA report , experienced external parties from the beginning take part in the project. In general they are involved in the execution of the analyses of the different issues, whereas EPZ employees are responsible for supervision and reviewing.

Quality control on the project execution is assured by the Nuclear Safety and Quality Assurance Department , which is reporting directly to the CEO of EPZ.

4.3 Content Licensee Final Report

For drafting of the Licencee Final Report EPZ follows the guidance given by the ENSREG Report dated 17 07 2011.

1. General data about site/plant

1.1. Brief description of the site characteristics

- location (sea, river)
- number of units;
- license holder

1.2. Main characteristics of the unit

- reactor type;
- thermal power;
- date of first criticality;
- existing spent fuel storage (or shared storage).

1.3. Systems for providing or supporting main safety functions

In this chapter, all relevant systems should be identified and described, whether they are classified and accordingly qualified as safety systems, or designed for normal operation and classified to non-nuclear safety category. The systems description should include also fixed hook-up points for transportable external power or water supply systems that are planned to be used as last resort during emergencies.

1.3.1. Reactivity control

Systems that are planned to ensure sub-criticality of the reactor core in all shutdown conditions, and sub-criticality of spent fuel in all potential storage conditions. Report should give a thorough understanding of available means to ensure that there is adequate amount of boron or other respective neutron absorber in the coolant in all circumstances, also including the situations after a severe damage of the reactor or the spent fuel.

1.3.2. Heat transfer from reactor to the ultimate heat sink

1.3.2.1. All existing heat transfer means / chains from the reactor to the primary heat sink (e.g., sea water) and to the secondary heat sinks (e.g., atmosphere or district heating system) in different reactor shutdown conditions: hot shutdown, cooling from hot to cold shutdown, cold shutdown with closed primary circuit, and cold shutdown with open primary circuit.

1.3.2.2. Lay out information on the heat transfer chains: routing of redundant and diverse heat transfer piping and location of the main equipment. Physical protection of equipment from the internal and external threats.

1.3.2.3. Possible time constraints for availability of different heat transfer chains, and possibilities to extend the respective times by external measures (e.g., running out of a water storage and possibilities to refill this storage).

1.3.2.4. AC power sources and batteries that could provide the necessary power to each chain (e.g., for driving of pumps and valves, for controlling the systems operation).

1.3.2.5. Need and method of cooling equipment that belong to a certain heat transfer chain; special emphasis should be given to verifying true diversity of alternative heat transfer chains (e.g., air cooling, cooling with water from separate sources, potential constraints for providing respective coolant).

1.3.3. Heat transfer from spent fuel pools to the ultimate heat sink

1.3.3.1. All existing heat transfer means / chains from the spent fuel pools to the primary heat sink (e.g., sea water) and to the secondary heat sinks (e.g., atmosphere or district heating system).

1.3.3.2. Respective information on lay out, physical protection, time constraints of use, power sources, and cooling of equipment as explained under 1.3.2.

1.3.4. Heat transfer from the reactor containment to the ultimate heat sink

1.3.4.1. All existing heat transfer means / chains from the containment to the primary heat sink (e.g., sea water) and to the secondary heat sinks (e.g., atmosphere or district heating system).

1.3.4.2. Respective information on lay out, physical protection, time constraints of use, power sources, and cooling of equipment as explained under 1.3.2.

1.3.5. AC power supply

1.3.5.1. Off-site power supply

1.3.5.1.1. Information on reliability of off-site power supply: historical data at least from power cuts and their durations during the plant lifetime.

1.3.5.1.2. Connections of the plant with external power grids: transmission line and potential earth cable routings with their connection points, physical protection, and design against internal and external hazards.

1.3.5.2. Power distribution inside the plant

1.3.5.2.1. Main cable routings and power distribution switchboards.

1.3.5.2.2. Lay-out, location, and physical protection against internal and external hazards.

1.3.5.3. Main ordinary on-site source for back-up power supply

1.3.5.3.1. On-site sources that serve as first back-up if offsite power is lost.

1.3.5.3.2. Redundancy, separation of redundant sources by structures or distance, and their physical protection against internal and external hazards.

1.3.5.3.3. Time constraints for availability of these sources and external measures to extend the time of use (e.g., fuel tank capacity).

1.3.5.4. Diverse permanently installed on-site sources for back-up power supply

1.3.5.4.1. All diverse sources that can be used for the same tasks as the main back-up sources, or for more limited dedicated purposes (e.g., for decay heat removal from reactor when the primary system is intact, for operation of systems that protect containment integrity after core meltdown).

1.3.5.4.2. Respective information on location, physical protection and time constraints as explained under 1.3.5.3.

1.3.5.5. Other power sources that are planned and kept in preparedness for use as last resort means to prevent a serious accident damaging reactor or spent fuel

1.3.5.5.1. Potential dedicated connections to neighbouring units or to nearby other power plants.

1.3.5.5.2. Possibilities to hook-up transportable power sources to supply certain safety systems.

1.3.5.5.3. Information on each power source: power capacity, voltage level and other relevant constraints.

1.3.5.5.4. Preparedness to take the source in use: need for special personnel, procedures and training, connection time, contract arrangements if not in ownership of the Licensee, vulnerability of source and its connection to external hazards and weather conditions.

1.3.6. Batteries for DC power supply

1.3.6.1. Description of separate battery banks that could be used to supply safety relevant consumers: capacity and time to exhaust batteries in different operational situations.

1.3.6.2. Consumers served by each battery bank: driving of valve motors, control systems, measuring devices, etc.

1.3.6.3. Physical location and separation of battery banks and their protection from internal and external hazards.

1.3.6.4. Alternative possibilities for recharging each battery bank.

1.4. Significant differences between units

This chapter is relevant only for sites with multiple NPP units of similar type.

In case some site has units of completely different design (e.g., PWR’s and BWR’s or plants of different generation), design information of each unit is presented separately.

1.5. Scope and main results of Probabilistic Safety Assessments

Scope of the PSA is explained both for level 1 addressing core meltdown frequency and for level 2 addressing frequency of large radioactive release as consequence of containment failure.

At each level, and depending on the scope of the existing PSA, the results and respective risk contributions are presented for different initiating events such as random internal equipment failures, fires, internal and external floods, extreme weather conditions, seismic hazards.

Information is presented also on PSA's conducted for different initiating conditions: full power, small power, or shutdown.

2. Earthquakes

2.1. Design basis

2.1.1. Earthquake against which the plant is designed

2.1.1.1. Characteristics of the design basis earthquake (DBE)

Level of DBE expressed in terms of maximum horizontal peak ground acceleration (PGA). If no DBE was specified in the original design due to the very low seismicity of the site, PGA that was used to demonstrate the robustness of the as built design.

2.1.1.2. Methodology used to evaluate the design basis earthquake

Expected frequency of DBE, statistical analysis of historical data, geological information on site, safety margin.

2.1.1.3. Conclusion on the adequacy of the design basis for the earthquake

Reassessment of the validity of earlier information taking into account the current state-of-the-art knowledge.

2.1.2. Provisions to protect the plant against the design basis earthquake

2.1.2.1. Identification of systems, structures and components (SSC) that are required for achieving safe shutdown state and are most endangered during an earthquake. Evaluation of their robustness in connection with DBE and assessment of potential safety margin.

2.1.2.2. Main operating contingencies in case of damage that could be caused by an earthquake and could threaten achieving safe shutdown state.

2.1.2.3. Protection against indirect effects of the earthquake

2.1.2.3.1. Assessment of potential failures of heavy structures, pressure retaining devices, rotating equipment, or systems containing large amount of liquid that are not designed to withstand DBE and that might threaten heat transfer to ultimate heat sink by mechanical interaction or through internal flood.

2.1.2.3.2. Loss of external power supply that could impair the impact of seismically induced internal damage at the plant.

2.1.2.3.3. Situation outside the plant, including preventing or delaying access of personnel and equipment to the site.

2.1.2.3.4. Other indirect effects (e.g. fire or explosion).

2.1.3. Compliance of the plant with its current licensing basis

2.1.3.1. Licensee's processes to ensure that plant systems, structures, and components that are needed for achieving safe shutdown after earthquake, or that might cause indirect effects discussed under 2.1.2.3 remain in faultless condition.

2.1.3.2. Licensee's processes to ensure that mobile equipment and supplies that are planned to be available after an earthquake are in continuous preparedness to be used.

2.1.3.3. Potential deviations from licensing basis and actions to address those deviations.

2.2. Evaluation of safety margins

2.2.1. Range of earthquake leading to severe fuel damage

Weak points and cliff edge effects: estimation of PGA that would result in damage to the weakest part of heat transfer chain, and consequently cause a situation where the reactor integrity or spent fuel integrity would be seriously challenged.

2.2.2. Range of earthquake leading to loss of containment integrity

Estimation of PGA that would result in loss of integrity of the reactor containment.

2.2.3. Earthquake exceeding the design basis earthquake for the plant and consequent flooding exceeding design basis flood

Possibility of external floods caused by an earthquake and potential impacts on the safety of the plant. Evaluation of the geographical factors and the physical possibility of an earthquake to cause an external flood on site, e.g. a dam failure upstream of the river that flows past the site.

2.2.4. Potential need to increase robustness of the plant against earthquakes

Consideration of measures, which could be envisaged to increase plant robustness against seismic phenomena and would enhance plant safety.

3. Flooding

3.1. Design basis

3.1.1. Flooding against which the plant is designed

3.1.1.1. Characteristics of the design basis flood (DBF)

Maximum height of flood postulated in design of the plant and maximum postulated rate of water level rising. If no DBF was postulated, evaluation of flood height that would seriously challenge the function of electrical power systems or the heat transfer to the ultimate heat sink.

3.1.1.2. Methodology used to evaluate the design basis flood.

Reassessment of the maximum height of flood considered possible on site, in view of the historical data and the best available knowledge on the physical phenomena that have a potential to increase the height of flood. Expected frequency of the DBF and the information used as basis for reassessment.

3.1.1.3. Conclusion on the adequacy of protection against external flooding

3.1.2. Provisions to protect the plant against the design basis flood

3.1.2.1. Identification of systems, structures and components (SSC) that are required for achieving and maintaining safe shutdown state and are most endangered when flood is increasing.

3.1.2.2. Main design and construction provisions to prevent flood impact to the plant.

3.1.2.3. Main operating provisions to prevent flood impact to the plant.

3.1.2.4. Situation outside the plant, including preventing or delaying access of personnel and equipment to the site.

3.1.3. Plant compliance with its current licensing basis

3.1.3.1. Licensee's processes to ensure that plant systems, structures, and components that are needed for achieving and maintaining the safe shutdown state, as well as systems and structures designed for flood protection remain in faultless condition.

3.1.3.2. Licensee's processes to ensure that mobile equipment and supplies that are planned for use in connection with flooding are in continuous preparedness to be used.

3.1.3.3. Potential deviations from licensing basis and actions to address those deviations.

3.2. Evaluation of safety margins

3.2.1. Estimation of safety margin against flooding

Estimation of difference between maximum height of flood considered possible on site and the height of flood that would seriously challenge the safety systems, which are essential for heat transfer from the reactor and the spent fuel to ultimate heat sink.

3.2.2. Potential need to increase robustness of the plant against flooding.

Consideration of measures, which could be envisaged to increase plant robustness against flooding and would enhance plant safety.

4. Extreme weather conditions

4.1. Design basis

4.1.1. Reassessment of weather conditions used as design basis

4.1.1.1. Verification of weather conditions that were used as design basis for various plant systems, structures and components: maximum temperature, minimum temperature, various type of storms, heavy rainfall, high winds, etc.

4.1.1.2. Postulation of proper specifications for extreme weather conditions if not included in the original design basis.

4.1.1.3. Assessment of the expected frequency of the originally postulated or the redefined design basis conditions.

4.1.1.4. Consideration of potential combination of weather conditions.

4.2. Evaluation of safety margins

4.2.1. Estimation of safety margin against extreme weather conditions

Analysis of potential impact of different extreme weather conditions to the reliable operation of the safety systems, which are essential for heat transfer from the reactor and the spent fuel to ultimate heat sink.

Estimation of difference between the design basis conditions and the cliff edge type limits, i.e. limits that would seriously challenge the reliability of heat transfer.

4.2.2. Potential need to increase robustness of the plant against extreme weather conditions

Consideration of measures, which could be envisaged to increase plant robustness against extreme weather conditions and would enhance plant safety.

5. Loss of electrical power and loss of ultimate heat sink

For writing chapter 5, it is suggested that detailed systems information given in chapter 1.3. is used as reference and the emphasis is in consecutive measures that could be attempted to provide necessary power supply and decay heat removal from the reactor and from the spent fuel.

Chapter 5 should focus on prevention of severe damage of the reactor and of the spent fuel, including all last resort means and evaluation of time available to prevent severe damage in various circumstances. As opposite, the chapter 6 should focus on mitigation, i.e. the actions to be taken after severe reactor or spent fuel damage as needed to prevent large radioactive releases. Main focus in chapter 6 should thus be in protection of containment integrity.

5.1. Nuclear power reactors

5.1.1. Loss of electrical power

5.1.1.1. Loss of off-site power

5.1.1.1.1. Design provisions taking into account this situation: back-up power sources provided, capacity and preparedness to take them in operation.

5.1.1.1.2. Autonomy of the on-site power sources and provisions taken to prolong the time of on-site AC power supply

5.1.1.2. Loss of off-site power and loss of the ordinary back-up AC power source

5.1.1.2.1. Design provisions taking into account this situation: diverse permanently installed AC power sources and/or means to timely provide other diverse AC power sources, capacity and preparedness to take them in operation

5.1.1.2.2. Battery capacity, duration and possibilities to recharge batteries

5.1.1.3. Loss of off-site power and loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power sources

5.1.1.3.1. Battery capacity, duration and possibilities to recharge batteries in this situation

5.1.1.3.2. Actions foreseen to arrange exceptional AC power supply from transportable or dedicated off-site source

5.1.1.3.3. Competence of shift staff to make necessary electrical connections and time needed for those actions. Time needed by experts to make the necessary connections.

5.1.1.3.4. Time available to provide AC power and to restore core cooling before fuel damage: consideration of various examples of time delay from reactor shutdown and loss of normal reactor core cooling condition (e.g., start of water loss from the primary circuit).

5.1.2. Measures which can be envisaged to increase robustness of the plant in case of loss of electrical power

5.1.3. Loss of the ultimate heat sink

5.1.3.1. Design provisions to prevent the loss of the primary ultimate heat sink, such as alternative inlets for sea water or systems to protect main water inlet from blocking.

5.1.3.2. Loss of the primary ultimate heat sink (e.g., loss of access to cooling water from the river, lake or sea, or loss of the main cooling tower)

5.1.3.2.1. Availability of an alternate heat sink

5.1.3.2.2. Possible time constraints for availability of alternate heat sink and possibilities to increase the available time.

5.1.3.3. Loss of the primary ultimate heat sink and the alternate heat sink

5.1.3.3.1. External actions foreseen to prevent fuel degradation.

5.1.3.3.2. Time available to recover one of the lost heat sinks or to initiate external actions and to restore core cooling before fuel damage:

consideration of situations with various time delays from reactor shutdown to loss of normal reactor core cooling state (e.g., start of water loss from the primary circuit).

5.1.3.4. Loss of the primary ultimate heat sink, combined with station black out (i.e., loss of off-site power and ordinary on-site back-up power source).

5.1.3.4.1. Time of autonomy of the site before start of water loss from the primary circuit starts.

5.1.3.4.2. External actions foreseen to prevent fuel degradation.

5.1.4. Measures which can be envisaged to increase robustness of the plant in case of loss of ultimate heat sink

5.2. Spent fuel storage pools

Where relevant, equivalent information is provided for the spent fuel storage pools as explained in chapter 5.1 for nuclear power reactors.

5.2.1. Loss of electrical power

5.2.2. Measures which can be envisaged to increase robustness of the plant in case of loss of electrical power

5.2.3. Loss of the ultimate heat sink

5.2.4. Measures which can be envisaged to increase robustness of the plant in case of loss of ultimate heat sink

6. Severe accident management

6.1. Organisation and arrangements of the licensee to manage accidents

Chapter 6.1 should cover organization and management measures for all type of accidents, starting from design basis accidents where the plant can be brought to safe shutdown without any significant nuclear fuel damage and up to severe accidents involving core meltdown or damage of the spent nuclear fuel in the storage pool.

6.1.1. Organisation of the licensee to manage the accident

6.1.1.1. Staffing and shift management in normal operation
6.1.1.2. Plans for strengthening the site organisation for accident management
6.1.1.3. Measures taken to enable optimum intervention by personnel
6.1.1.4. Use of off-site technical support for accident management
6.1.1.5. Procedures, training and exercises.

6.1.2. Possibility to use existing equipment

6.1.2.1. Provisions to use mobile devices (availability of such devices, time to bring them on site and put them in operation)

6.1.2.2. Provisions for and management of supplies (fuel for diesel generators, water, etc.)

6.1.2.3. Management of radioactive releases, provisions to limit them

6.1.2.4. Communication and information systems (internal and external).

6.1.3. Evaluation of factors that may impede accident management and respective contingencies

6.1.3.1. Extensive destruction of infrastructure or flooding around the installation that hinders access to the site

6.1.3.2. Loss of communication facilities / systems

6.1.3.3. Impairment of work performance due to high local dose rates, radioactive contamination and destruction of some facilities on site

6.1.3.4. Impact on the accessibility and habitability of the main and secondary control rooms, measures to be taken to avoid or manage this situation

6.1.3.5. Impact on the different premises used by the crisis teams or for which access would be necessary for management of the accident

6.1.3.6. Feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods)

6.1.3.7. Unavailability of power supply

6.1.3.8. Potential failure of instrumentation

6.1.3.9. Potential effects from the other neighbouring installations at site.

6.1.4. Measures which can be envisaged to enhance accident management capabilities

6.2. Maintaining the containment integrity after occurrence of significant fuel damage (up to core meltdown) in the reactor core

6.2.1. Elimination of fuel damage / meltdown in high pressure

6.2.1.1. Design provisions

6.2.1.2. Operational provisions

6.2.2. Management of hydrogen risks inside the containment

6.2.2.1. Design provisions, including consideration of adequacy in view of hydrogen production rate and amount

6.2.2.2. Operational provisions

6.2.3. Prevention of overpressure of the containment

6.2.3.1. Design provisions, including means to restrict radioactive releases if prevention of overpressure requires steam / gas relief from containment

6.2.3.2. Operational and organisational provisions

6.2.4. Prevention of re-criticality

6.2.4.1. Design provisions
6.2.4.2. Operational provisions

6.2.5. Prevention of basemat melt through

6.2.5.1. Potential design arrangements for retention of the corium in the pressure vessel
6.2.5.2. Potential arrangements to cool the corium inside the containment after reactor pressure vessel rupture
6.2.5.3. Cliff edge effects related to time delay between reactor shutdown and core meltdown

6.2.6. Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity

6.2.6.1. Design provisions
6.2.6.2. Operational provisions

6.2.7. Measuring and control instrumentation needed for protecting containment integrity

6.2.8. Measures which can be envisaged to enhance capability to maintain containment integrity after occurrence of severe fuel damage

6.3. Accident management measures to restrict the radioactive releases

6.3.1. Radioactive releases after loss of containment integrity

6.3.1.1. Design provisions
6.3.1.2. Operational provisions

6.3.2. Accident management after uncovering of the top of fuel in the fuel pool

6.3.2.1. Hydrogen management
6.3.2.2. Providing adequate shielding against radiation
6.3.2.3. Restricting releases after severe damage of spent fuel in the fuel storage pools
6.3.2.4. Instrumentation needed to monitor the spent fuel state and to manage the accident
6.3.2.5. Availability and habitability of the control room

6.3.3. Measures which can be envisaged to enhance capability to restrict radioactive releases

5. General data about site/plant

In this chapter an example is given how EPZ intends to fulfil in the requested information in the Final Report.

Attention: At this moment this example is not yet fully in agreement with the guidance given by ENSREG (see paragraph 4.3).

5.1 Brief description of the site characteristics

The Borssele nuclear power plant (KCB) is situated on the northern shore of the river Westerschelde about 1.4 km northwest of the village of Borssele. The area belongs to the municipality of Borsele and is owned by the N.V. EPZ. EPZ has received its NPP operating license, based on the Nuclear Energy Law (KEW), from the former Ministry of VROM in The Hague.

A bird's view photograph of the Borssele site is given in figure 5.1

Figure 5.1 Borssele site

Several types of production units are located on the Borssele site. These units are:

Borssele Nuclear Power Plant (unit BS30)

- Construction started in 1969, first production in 1973.
- Gross capacity 512 Megawatt, net capacity 485 Megawatt.

Borssele Coal-fired Power Plant (unit BS12)

- Built as oil-fired station in 1972. Converted to coal-firing in 1987 and also able to use natural gas.
- Gross capacity 427 Megawatt, net 404 Megawatt.
- Modified to be fuelled also with Phosphorus Gas (by product from neighbouring industry) and biomass.

Wind powered turbines

At the site five wind turbines are in operation since early 2005. Installed capacity 11.75 Megawatts.

The total installed net capacity installed at the site is 898 Megawatts.

Figure 5.2 shows a copy of the plot plan of the units.

Surrounding area

The site is located directly behind the seawall of the Westerschelde. The area around the site is generally flat. On the north side the site is bounded by the industrial areas around the seaport Sloehaven. This port area comprises heavy industries like oil refinery, phosphor production, aluminum production, etc. The industries are located at a distance of 1-3 km from the EPZ-site.

East and south of EPZ is a mainly agricultural area, while from south-east to west one finds the water of the Westerschelde.

On the river Westerschelde, intensive ship traffic takes place. The number of sea going ships amounts over 40,000 per year. Origin or destination is in many cases the port of Antwerp (Belgium). Among these ships, there are also transports of dangerous materials, like LPG, flammable liquids and liquified ammonia.

The NPP Borssele is located approximately 7.2 km from the major A-58 highway (E312).The nearest point of the local road to the plant is 500 m (N254).

The NPP Borssele is located approximately 0.5 km from the nearest railway line. A local yard and tracks from the main line provide service to the local ports and industries.

The Airport Midden Zeeland is situated at about 10 km north of the site. This airport is intended for small civilian aircrafts with a maximum weight of less than 5.7 ton.

For large civil aircrafts with a maximum take-off weight of more than 5.7 ton, the so-called en-route flying must be carried out in prescribed airways. The airway A5 for flights from southern direction to Schiphol Airport and airway B29 for flights from Brussels to London are located 20 km east respectively and 20 km south of the KCB.

The closest military airbase is Woensdrecht in Noord Brabant, at a distance of 40 km in northeasterly direction. For nuclear power plants in The Netherlands a restricted area for military air traffic is applicable, with dimensions of 3.6 km * 3.6 km horizontally and 0.5 km vertically.

The village of Borssele is located at about 1.4 km northwest of the site. The cities of Vlissingen, Middelburg, Goes and Terneuzen lie at distances of respectively 10, 10, 15 and 16 km.

Figure 5.2 Map of the Borssele site buildings

5.2 Main characteristics of the unit

5.2.1 Technical description of Borssele NPP

Figure 5.3 Borssele Nuclear Power Plant

The Borssele Nuclear Power Plant was designed and built by Kraftwerk Union (KWU) and started commercial operation in October 1973. An overview of the plant is given in figure 5.3.

The nuclear reactor is a 1365 MW_{th} pressurized water reactor with two loops each with one primary pump and one steam generator. The thermal power has not been up rated. However, the turbines have been retrofitted in 2006 for better thermal efficiency. Presently, the gross capacity is 512 MW_e and the net capacity is 485 MW_e. The turbine project has added 35 MW_e. The steam generators are the original ones, tubed with Incoloy 800; only a small fraction of tubes have been plugged and the steam generators are in good condition.

The turbine generator system consists of one high pressure and three condensing dual-flow steam turbines, a generator and an exciter on a single shaft. The condensers have titanium-tubes and are cooled with salt water from the Westerschelde. As usual in the KWU/Siemens plant design, the condensate is collected and de-aerated in a large feed water accumulator. The original copper-nickel condenser tubes have been replaced with titanium. The hydrogen-cooled generator has 21 kV coils and a 150 kV main transformer.

The main control room was backfitted in the Modification Project (1997) and is based on ergonomically optimization of plant operation procedures, including emergency procedures. A redundant bunkered control room is available for controlled shutdown, core cooling and spent fuel pool cooling after external hazards and in beyond design conditions.

The reactor protection system has been replaced in 1997 and is based on the principle that no operator action is required in the first 30 minutes after start of the event, for design base accidents. Operating manuals for incidents and accidents are based on the event- and symptom-based Westinghouse Owners Group (WOG) Emergency Operating Procedures and Accident Management Guidelines.

The containment is a 46-meter spherical steel shell, which is in turn encapsulated by the concrete reactor building. The spherical shell not only contains the reactor and steam generators, but also the spent fuel pool. There is no separate fuel storage facility outside the containment. The water in the spent fuel pool contains boron at 2300 ppm. Boron is however not required to guarantee a sub criticality of 5%.

To cope with external hazards, important safety systems like emergency core cooling, spent fuel pool cooling, reactor protection system and emergency control room are installed in “bunkered” buildings. These buildings are qualified to withstand earthquake, flooding, gas cloud explosions, airplane crash and severe weather conditions.

There are two grids for emergency AC power system, for different levels of plant accident conditions. The main grid 1 has 300% capacity (3 diesel generators) and the bunkered grid 2 has 2 extra, smaller diesels (2 x 100%) in separate rooms. Likewise, other essential safety systems have been backed up in the bunkers. The 4-pump Emergency Core Cooling System is backed up by a 2-train bunkered system, and the 3-pump Auxiliary Feed water System is also backed up by a 2-train bunkered system.

For very improbable conditions that result in the failure of all trains of the emergency cooling water system the plant is equiped with a redundant ultimate heat sink.

This ultimate heat sink can remove decay heat from the reactor core and the spent fuel storage pool, and provide cooling water to the emergency diesels. Its cooling water is ground water, pumped from seven wells on the site. The system is operated from the emergency control room in the protected area.

A number of accident management systems are in place. There is a reactor vessel level indicator, accident-qualified primary pressure relief valves, a filtered containment venting line and hydrogen recombiners in the reactor building.

The plant specific full-scope control room simulator is used for operator training with the full range of operational events.

The reactor is fuelled with 121 fuel assemblies 15x15 grid, containing 38.8 ton uranium as UO_2. The enrichment level of the fuel has increased over the years from 3.3% ^{235}U to 4.4% ^{235}U.

The present reactor core exists of a mix of two enrichment levels 4.0% and 4.4%.

The reactor is run in a 12-month cycle with the annual refueling outage in April. Areva (formerly Framatome ANP) is the vendor of fuel elements and is contractor for specialized maintenance and inspection jobs.

Figure 5.4 shows a copy of the plot plan of the unit

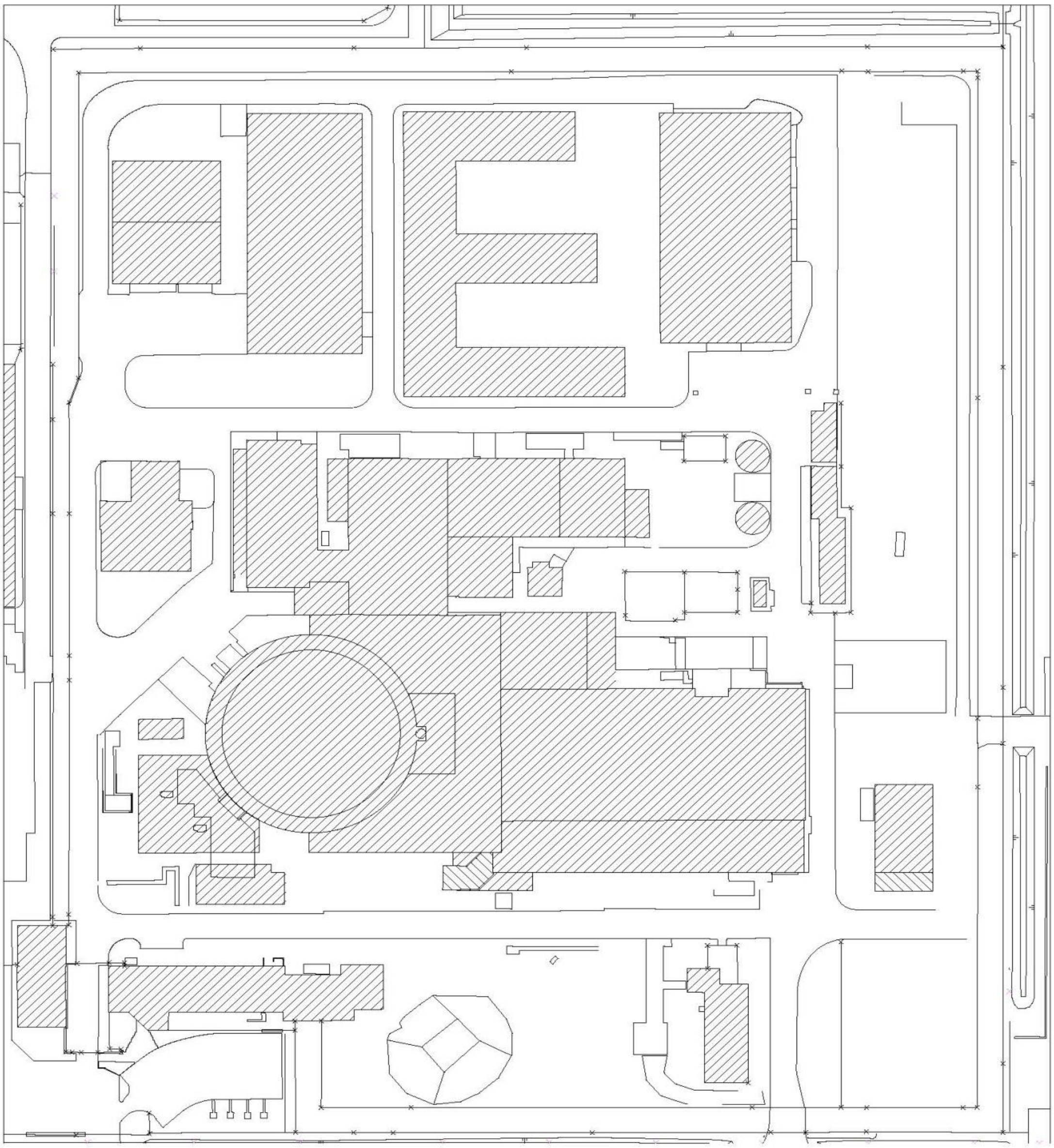

Figure 5.4 Map of NPP Borssele buildings

5.2.2 Key plant parameters and system characteristics

Tables 5.1 and 5.2 present the key plant parameters and the main safety systems characteristics for Borssele NPP.

Characteristics and measuring units	**Value**
Primary coolant pressure at the core outlet, bar	155 bar
Coolant temperature at the reactor inlet, C	292.5 °C
Coolant temperature at the reactor outlet, C	317.5 °C
Pressurizer temperature, C	362 °C
Coolant flow rate through the reactor, m^3/h	46,260 m^3/h
Steam flow rate at nominal parameters, kg/s	743 kg/s
Steam generator pressure at nominal load, bar	58 bar
Steam temperature at nominal load, C	272.2 °C
Feed water temperature in nominal mode, C	214.2 °C
Design primary pressure, bar	175 bar
Design primary temperature, C	350 °C
Design secondary pressure, bar	92.2 bar
Design secondary temperature, C	305 °C
Primary hydraulic test pressure: - for leak tightness, bar - for strength, bar	 171 bar 226 bar
Secondary hydraulic test pressure for tightness, MPa	114 bar
Pressurizer capacity (full volume), m^3	40.54 m^3
Spent fuel pool capacity, m^3	1,000 m^3
Reactor pool capacity, m^3	680 m^3
Boron concentration Spent fuel pool, ppm	2,300 ppm

Table 5.1 Key Plant Parameters of NPP Borssele

System	Features/Characteristics
Reactivity Control Systems	28 control rods 3 volume control pumps (3x4,4 kg/s) 2 boron injection pumps (2x2,2 kg/s @ 4,7 bar, 21000 ppm) connected to the volume control system 2 high head backup boron injection pumps (5,2 kg/s, 185 bar) with separate tanks (243 m^3 and 262 m^3 @ 2300 ppm)
Primary pressure protection system	3 tandem pressurizer relief valves with Power Operating Relief Valve function. Opening/closing pressures: 172/162 bar, 176/166 bar and 180/170 bar. Automatic pressure limiting by control rod drop if primary pressure exceeds 163 bar
System for emergency and scheduled cooling down of the primary circuit and fuel storage pool cooling	Primary system: 2 trains of Residual Heat Removal system with 2 pumps (2x167 kg/s @ 6.7 bar) each, seawater cooled (using the component cooling water system as interface). Separate heat removal system with 2 redundant pumps (2x61.1 kg/s), well water cooled Spent fuel pool: 2 cooling trains with 1 pump (64 kg/s @ 3.4 bar) and 1 cooler each, seawater cooled (using the component cooling water system as interface). Back up cooler, well water cooled and 1 back-up pump (64 kg/s @ 3.4 bar)
Coolant injection systems	2 trains of 2 high head safety injection pumps (max 110 bar, 55.6 kg/s @ 60 bar) each, 2 trains of 2 low head safety injection pumps (max 9 bar, 167 kg/s @ 6.7 bar) each, 2 trains of 2 accumulators (4x28 m^3, 25 bar) each 2 trains of 2 storage tanks (178 m^3 each) each

Table 5.2 Main Safety Systems Characteristics of NPP Borssele

Steam Generator Heat Removal Systems	3 main feed water pumps (3*380 kg/s @ 66 bar, 3*50%) 3 auxiliary feed water pumps (3x24.4 kg/s @ 100 bar, 3*100%) one of them turbine driven 2 back up feed water pumps (2*18 kg/s @ 80 bar, 2*100%) with 1 tank (450 m^3) each
Secondary side pressure protections and steam removal	2 trains of 10 safety relief valves, opening pressures 87 bar, 91.5 bar and 93 bar each 2 trains of 2 atmospheric steam dump valves, opening pressures 81.4 bar and 83.2 bar each 3 turbine bypass valves to the main condenser (3*50%), opening pressure 78.5 bar
Main steam lines isolation system	2 fast closing main steam isolation valves 2 self powered line break valves in the crossover line between the main steam lines
Containment Systems	Filtered containment venting system Passive hydrogen recombiners (PARs)
Key Safety Support Systems	Self testing reactor protection system Emergency control room Fire protection systems: Inergen, CO_2, fine water spray and Sprinkler systems, crash tender.
Diesel generators	2 grids for emergency AC power system, for different levels of plant accident conditions. Main emergency grid 1: 2 air-cooled 6 kV diesel generators (2 x 100%, 2 x 4.7 MW) and 1 separated water-cooled diesel generator (1 x 100%, 1 x 4.7 MW) Bunkered grid 2: 2 separately bunkered water-cooled 380 Volt, 1.8 MW diesel generator (2*100%). These diesels supply AC power to emergency grid 2, which is designed for essential safety functions in case of specific accident conditions (essentially, for the reactor protection system, feed water and primary injection, spent fuel pool and well cooling water systems). Batteries for the no-break power supply; capacity for at least 2 hrs

Table 5.2 Main Safety Systems Characteristics of NPP Borssele (con't)

5.3 Significant differences between units

Not relevant.

5.4 Scope and main results of Probabilistic Safety assessments

5.4.1 Overall scope of the Probabilistic Safety Assessments

The overall scope of the Probabilistic Safety Assessment (PSA) for NPP Borssele is established proceeding from the aim of providing a full, "living" PSA model, and therefore includes the following[1]:

1. A complete Level-1 PSA for power and non-power operations
2. A Level-2 analysis extending Level-1 results to identify the likelihood and mechanism of potential releases from the containment, and
3. A Level-3 assessment of a dose release and associated risk to the population.

5.4.1.1 Scope of the Level-1 PSA

The scope of Level-1 PSA includes internal initiating events, internal and external hazards for power and non-power operations.

The Level-1 PSA includes an analysis of more than 75 internal and spatial initiating events as well as an analysis of external initiating events. The degree of detail in the systems analysis is such that the effectiveness of potential hardware modifications can be demonstrated. The Borssele operating experience has been taken into account in the data analysis, which reflects the plant specific maintenance policy and its effect on plant specific component test and maintenance unavailabilities.Consideration is given to dependencies and human failures; dependency matrices are developed for all systems (front line and support systems). The human reliability analysis embedded in the Borssele systems analysis considers the evaluation of pre- and post-accident human actions, as well as actions potentially inducing initiating events.

The Level-1 analysis evaluates the core damage frequency and plant damage state frequencies, and the identification of the main weak points in the plant safety features.The external events PSA has been conducted based on a successive screening process. First, the external events scenarios were identified, the initiating event frequency quantified, and the impact on the plant determined. If the frequency was low, then the scenario has been screened out. If the frequency was above the truncation frequency, then the plant response and other factors were considered.

5.4.1.2 Scope of the Level-2 PSA

The total core damage frequency obtained in Level-1 PSA is further developed into a release frequency in the Level-2 analysis. The severe accident progression analysis has been performed using MAAP4, with ex-vessel phenomena having been analysed with more

[1] Probabilistic Safety Assessment for the Borssele Nuclear Power Plant. Living PSA 2009-1. September 2009

detailed mechanistic codes to obtain details concerning core/concrete interaction, hydrogen distribution and containment loads.

The Level-2 analysis is based on twelve accident sequences representing the major physical processes during accident progression. The results of the Level-2 analysis are used as the input into the Level-3 analysis.

5.4.1.3 Scope of the Level-3 PSA

The results of the Level-2 analysis were used as an input into the Level-3 analysis. The COSYMA computer program was used to determine the dose release; then the latter have been used to derive consequences to human life, wild life, and vegetation.

5.4.2 Results of the Probabilistic Safety Assessment

5.4.2.1 Results of the Level-1 PSA

The results of full-power Level-1 PSA are presented in Table 5.3.

Total Core Damage Frequency-P			
Initiating Event Group	**Subgroup**	**Core Damage Frequency (per year)**	**Percent of Total**
Internal	LOCA (Loss of cooling accidents)	1.95E-07	13.3%
	ISLOCA (Interfacing systems LOCA)	5.78E-09	0.4%
	SGTR (Steam generators tube rupture)	2.85E-08	1.9%
	SBO (Station blackout)	2.05E-09	0.1%
	ATWS (Anticipated transients without scram)	7.99E-09	0.5%
	TRANS (Transients)	4.77E-09	0.3%
	SSIE (Special system initiators)	2.02E-07	13.8%
INTERNAL HAZARDS		9.38E-07	64.1%
EXTERNAL HAZARDS		7.92E-08	5.4%
TOTAL		1.46E-06	100.0%

Table 5.3 Full power Level-1 PSA results

This power plant operational state (POS) is the largest contributor to the core damage frequency due to the fact that the plant is usually in this POS for about 95% of the year and therefore it contributes dominantly to the total core damage frequency. It could be seen that internal hazards provide the highest contribution to the core damage frequency for power operation conditions.

Low power and shutdown Level-1 PSA results are presented in Table 5.4.

Total CDF contributions of all POSs			
	Fraction of year	**Core Damage Frequency (per year)**	**Percent of Total**
POS-HE/HL[2]	5.92E-3	9.44E-09	1.28%
POS-RE/RL	5.80E-3	1.34E-07	18.17%
POS-ME/ML	8.93E-3	5.56E-07	75.40%
POS-CU/CL	9.78E-3	6.20E-09	0.84%
POS-FE	2.28E-2	3.18E-08	4.31%
TOTAL		7.37E-07	100%

Table 5.4 Low power and shutdown Level-1 PSA results

The contribution of low power and shutdown POS to the core damage frequency is about 35%.

The highest contributor is the POS-ME/ML (Midloop operation) with the core damage frequency fraction of about 75% for all non-power operational states.

Although there is only a short time interval in this POS, the POS is important because of the reduced inventory and the fact that there is no redundancy for the low pressure Emergency Core Cooling System. The contribution of the initiating event groups within this highest contributing POS is shown in Table 5.5.

[2] POS-HE/HL: hot standby, early, late; POS-RE/RL: cold standby, early, late; POS-ME/ML: midloop operation, early, late; POS-CU/CL: unloading and loading of the core; POS-FE: core in spent fuel pool.

TCDF-ME/ML			
Initiating Event Group	**Subgroup**	**Core Damage Frequency (per year)**	**Percent of Total**
Internal	LOCA	1,11E-06	1,8%
	ISLOCA	4,34E-09	0,0%
	SGTR	0,00E+00	0,0%
	SBO	4,80E-08	0,1%
	ATWS	3.06E-08	0.0%
	TRANS	5.74E-07	0.9%
	SSIE	1.65E-06	2.7%
INTERNAL HAZARDS		5.81E-05	93.4%
EXTERNAL HAZARDS		7.04E-07	1.1%
TOTAL		6.23E-05	100.0%

Table 5.5 Contribution of the initiating event groups within POS-ME/ML

Similar to power operation, internal hazards provides the highest contribution into the core damage frequency for low power and shutdown states.

5.4.2.2 Results of the Level-2 PSA

The total core damage frequency is divided into four time frames associated with the releases:

Early Release	Release 0 - 12 hours following reactor trip or shutdown
Late Release	Release 12 - 72 hours following reactor trip or shutdown
Very Late Release	Release greater than 72 hours after reactor trip or shutdown
No Release	Core degradation arrested prior to containment failure

The summary of the source terms frequencies and fission products release fractions are presented in Table 5.6.

Time Phase	STC	Frequency	Percent of total	Percent of Time Phase Total	Containment Release Mode
Early Releases	1	4,41E-09	0,2%	18,8%	Dry SGTR without isolation
	2	1,66E-10	0,0%	0,7%	Dry SGTR with isolation
	3	1,65E-08	0,8%	70,4%	Induced SGTR with secondary water
	4	2.18E-09	0,1%	9.3%	Containment Rupture
	5	1.81E-10	0,0%	0,8%	Containment Leak
Total Early Releases		2.34E-08	**1,1%**		
Late Releases	6	6.68E-09	0,3%	21.2%	Interfacing System LOCA
	7	1,15E-10	0,0%	0,4%	SGTR without secondary water
	8	1,14E-08	0,5%	36.2%	SGTR with secondary water
	9	4.99E-09	0,2%	15.9%	Containment Rupture
	10	8.28E-09	0,4%	26,3%	Containment Leak + Isolation Failure
Total Late Releases		3.14E-08	**1.5%**		
Very Late Releases	11	1,34E-12	0,0%	0,0%	ISLOCA + Isolation Failure
	12	1,55E-10	0,0%	0,0%	SGTR with and without secondary water
	13	2,18E-09	0,0%	0,1%	Containment Rupture and Leak
	14	2,32E-09	0,1%	0,1%	Basemat Penetration
	15	1.59E-06	74,9%	99,8%	Filter Vented Release
Total Very Late Releases		1.59E-06	**75.0%**		
No Release	16	4.75E-07	22.4%	100	No Containment Failure
Total No Release		4.75E-07	22.4%		
All		**2.21E-06**			

Table 5.6 Summary of Source Term Grouping

5.4.2.3 Results of the Level-3 PSA

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According to the Dutch risk policy two criteria are to be met:

1. The maximum allowable individual risk to die as a consequence of operation of a certain installation is 10^{-6} per year. According to the Dutch risk approach, the individual risk shall be calculated for one year old children, since this is in general the most vulnerable group of the population.

2. The societal risk is defined as the risk of 10 or more casualties, which are directly attributable to the accident, and this risk shall be lower than 10^{-5} per year for 10 deaths, 10^{-7} per year for 100 deaths, 10^{-9} per year for 1000 deaths, etc.

Ad 1. Total lifetime individual risk

The total lifetime individual risk for all source terms is shown in Figure 5.5. Furthermore, the maximum individual risk limit (10^{-6} per year) is shown on this figure.

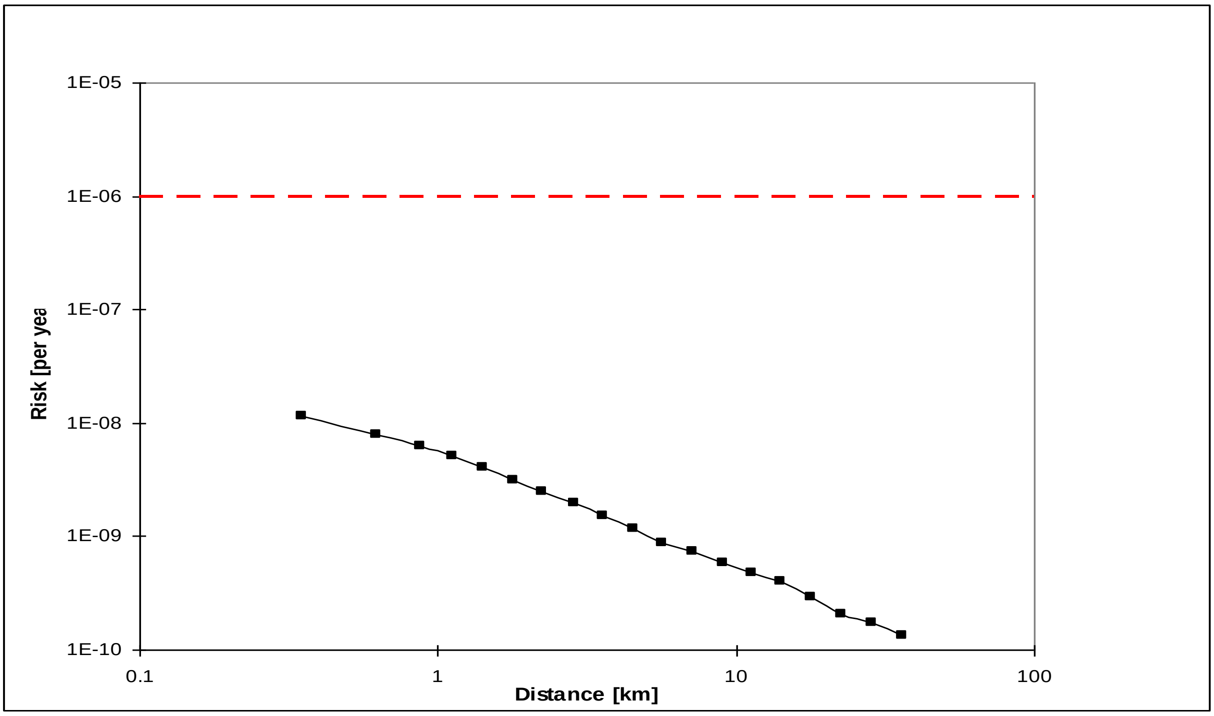

Figure 5.5 Total lifetime individual risk

It can be seen that individual risk is well below the criterion for Borssele NPP.

Ad 2. Societal risk

The probability of exceeding certain early deaths as quantified in Level-3 PSA for Borssele NPP is shown in Figure 5.6. In this figure, the societal risk criterion (acceptable level) is shown.

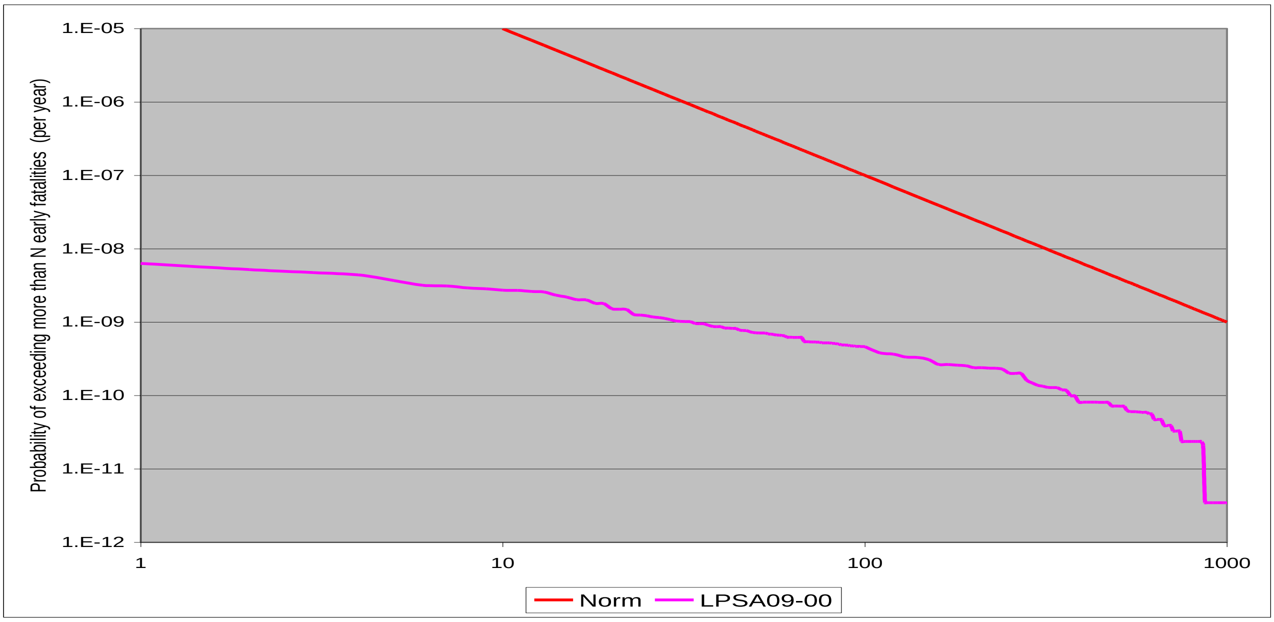

Figure 5.6 CCDF of early fatalities

It can be seen that the assessed societal risk is also well below the criterion for Borssele NPP.

ANNEX 1 Letter to EPZ from the Ministry of Economics, Agriculture and Innovation, Identification ETM/ED/11074538, June 1, 2011.

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Ons kenmerk
ETM/ED / 11074538

Uw kenmerk

Bijlage(n)
1

Datum 1 juni 2011

Betreft Uitvoering stresstest

Geachte heer Bongers,

Het ongeval in Fukushima heeft op Europees niveau geleid tot de beslissing om de bestaande kerncentrales in Europa te onderwerpen aan een stresstest.
Over de inhoud ervan en de procedure is inmiddels overeenstemming bereikt. Bijgesloten treft u de documenten aan die in Brussel door ENSREG (European Nuclear Safety Regulators) zijn besproken en geaccordeerd.

Zoals u daarbij kunt lezen, houdt de stresstest een doelgerichte vaststelling in van de veiligheidsmarges van iedere kerncentrale in Europa. De stresstest bestaat uit enerzijds een evaluatie van de manier waarop een kerncentrale reageert als deze blootgesteld wordt aan extreme omstandigheden en anderzijds een verificatie van de preventieve en mitigerende maatregelen die de veiligheid van de centrale moeten borgen.

Ik acht het van belang dat op korte termijn niet alleen de kerncentrale Borssele aan een stresstest wordt onderworpen, maar ook de onderzoeksreactoren in Nederland (Hoge Flux Reactor en Lage Flux reactor in Petten en Hoger Onderwijs Reactor in Delft). Daarmee kunnen snel op duidelijke en transparante wijze de veiligheidsmarges van deze kernreactoren worden beoordeeld.

Uitvoering van de stresstest
De vergunninghouder is als eerstverantwoordelijk voor de nucleaire veiligheid is belast met het uitvoeren van de stresstest. Het bevoegde gezag zal de stresstest, op onafhankelijke wijze, beoordelen. Doel van deze brief is een eerste stap te zetten om te komen tot de invulling van de stresstest voor de kerncentrale Borssele. Om die reden zal op korte termijn een afspraak met u worden gemaakt, om de uit te voeren stresstest nader met u te bespreken en specificeren. Daarbij zal het hieronder genoemde tijdschema als randvoorwaarde gelden.

Bij de stresstest wordt specifiek gekeken naar buitengewone gebeurtenissen zoals een aardbeving en overstromingen, maar er wordt ook gekeken naar de consequenties van andere gebeurtenissen die kunnen leiden tot het verlies van meerdere veiligheidsfuncties en tot een ernstig ongeval. Denk daarbij bijvoorbeeld

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Duurzaamheid

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ETM/ED / 11074538

aan een ernstige storing de elektriciteitnet, een bosbrand of een neerstortend vliegtuig. Tevens zal rekening moeten worden gehouden met moedwillige verstoringen.

De stresstest moet leiden tot inzicht in ernstige ongevalcondities en hoe de installatie daarop reageert, ook in het geval dat de voorziene noodmaatregelen in die situatie wegvallen. Dit betekent dat voor het vaststellen van de veiligheidsmarges een deterministische benadering wordt gekozen. De bedoeling is dat hierbij uitgegaan wordt van een steeds ernstigere bedreiging (bijvoorbeeld een steeds hogere vloedgolf of zwaardere aardbeving) en dat bezien wordt hoe de installatie en het veiligheidsmanagementsysteem hierop reageren en tot welk niveau van bedreiging de veiligheidssystemen afdoende werken. Voor de verdere evaluatie en het nemen van eventuele maatregelen is uiteraard wel van belang te weten hoe groot de kans is dat een dergelijke gebeurtenis zich voordoet, en deze informatie zal dan ook gerapporteerd moeten worden.

Als resultaat zal de stresstest het volgende moeten opleveren:
1. inzicht in hoe de installatie en het veiligheidsmanagementsysteem reageren bij in ernst steeds toenemende ongevallen en bij veiligheidvoorzieningen die geleidelijk aan in onbruik raken;
2. wat eventueel zwakke plekken van de installatie en het veiligheidsmanagementsysteem zijn;
3. hoe deze zwakke plekken verbeterd kunnen worden.

Verdere informatie over de afbakening van de stresstest kunt u lezen in bijgevoegde documenten.

Peer review en transparantie
Ik hecht eraan een paar aspecten van het proces dat tot de Europese stresstest zal leiden, te benadrukken: de peer review, de transparantie en het zonodig nemen van maatregelen.

Om de betrouwbaarheid en de verantwoording van het hele proces te verhogen, zullen de nationale rapporten, zoals gevraagd door de Europese Raad, worden onderworpen aan een internationale peer review. Belangrijkste doel van het nationaal rapport is het trekken van conclusies uit de resultaten van de stresstest uitgevoerd door de vergunninghouders volgens de afgesproken systematiek.

Bij het hele EU stresstest proces zal ik de eerder dit jaar door ENSREG vastgestelde principes voor openheid en transparantie volgen: de rapporten zullen openbaar zijn, rekening houdend met nationale wetgeving en internationale verplichtingen, ook op het gebied van beveiliging.
Bij de peer reviews zullen de conclusies van de nationale rapporten worden besproken en de mate waarin zij in overstemming zijn met de afgesproken methodologie. De resultaten van de peer review zullen openbaar zijn.

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Het is de bedoeling de resultaten van de reviews te presenteren en bediscussiëren in openbare nationale en Europese bijeenkomsten. Transparantie en de mogelijkheid voor de maatschappij om daarbij betrokken te zijn, zal bijdragen aan de acceptatie van deze Europese stresstest.

Planning
Conform het EU voorstel, zal ik voor de stresstest voor de kerncentrales Borssele de volgende planning hanteren:

- 1 juni 2011: het bevoegde gezag verzoekt de vergunninghouders de stresstest uit te voeren;
- 15 augustus 2011: de vergunninghouders leveren een voortgangsrapport aan het bevoegde gezag;
- 15 september 2011: de lidstaten leveren een voortgangsrapport aan de Europese Commissie;
- 31 oktober 2011: de vergunninghouders leveren de definitieve rapporten met de resultaten van de stresstests aan bij het bevoegde gezag;
- 9 december 2011: behandeling van de nationale voortgangsrapporten, met conclusies daarover van de Europese Commissie, in de Europese Raad;
- 31 december 2011: de lidstaten leveren landenrapporten over de definitieve rapporten van de vergunninghouders aan bij de Europese Commissie; deze worden aan een peer review onderworpen
- 30 april 2012: de peer reviews van de landenrapporten zijn afgerond
- Medio 2012: de landenrapporten, inclusief peer reviews en de conclusies erover van de Europese Commissie, worden in de Europese Raad besproken.

Maatregelen
Tenslotte wil ik benadrukken dat, indien de resultaten van de stresstest daartoe aanleiding geven, maatregelen ter verhoging van de veiligheidsmarges genomen moeten worden.

Ik verzoek u in lijn met de aangegeven planning de stresstest uit te voeren. Voor nadere vragen kunt u zich wenden tot de met de mw. Delfini, van het ministerie van ELenI, of de heer Verweij van de Kernfysische Dienst.

Hoogachtend,

mr. Anneke van Limborgh
MT-lid directie Energie en Duurzaamheid

Pagina 3 van 3

ANNEX 2 ENSREG Safety Annex I EU "Stress tests" specifications

Introduction

Considering the accident at the Fukushima nuclear power plant in Japan, the European Council of March 24th and 25th declared that "the safety of all EU nuclear plants should be *reviewed, on the basis of a comprehensive and transparent risk assessment ("stress tests"); the European Nuclear Safety Regulatory Group (ENSREG) and the Commission are invited to develop as soon as possible the scope and modalities of these tests in a coordinated framework in the light of the lessons learned from the accident in Japan and with the full involvement of Member States, making full use of available expertise (notably from the Western European Nuclear Regulators Association); the assessments will be conducted by independent national authorities and through peer review; their outcome and any necessary subsequent measures that will be taken should be shared with the Commission and within ENSREG and should be made public; the European Council will assess initial findings by the end of 2011, on the basis of a report from the Commission*".

On the basis of the proposals made by WENRA at their plenary meeting on the 12-13 of May, the European Commission and ENSREG members decided to agree upon "an initial independent regulatory technical definition of a "stress test" and how it should be applied to nuclear facilities across Europe". This is the purpose of this document.

Definition of the "stress tests"

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.

This reassessment will consist:

- in an evaluation of the response of a nuclear power plant when facing a set of extreme situations envisaged under the following section "technical scope" and
- in a verification of the preventive and mitigative measures chosen following a defence-in-depth logic: initiating events, consequential loss of safety functions, severe accident management.

In these extreme situations, sequential loss of the lines of defence is assumed, in a deterministic approach, irrespective of the probability of this loss. In particular, it has to be kept in mind that loss of safety functions and severe accident situations can occur only when several design provisions have failed. In addition, measures to manage these situations will be supposed to be progressively defeated.

For a given plant, the reassessment will report on the response of the plant and on the effectiveness of the preventive measures, noting any potential weak point and cliff-edge effect, for each of the considered extreme situations. A cliff-edge effect could be, for instance, exceeding a point where significant flooding of plant area starts after water overtopping a protection dike or exhaustion of the capacity of the batteries in the event of a station blackout. This is to evaluate the robustness of the defence-in-depth approach, the adequacy of current accident management measures and to identify the potential for safety improvements, both technical and organisational (such as procedures, human resources, emergency response organisation or use of external resources).

By their nature, the stress tests will tend to focus on measures that could be taken after a postulated loss of the safety systems that are installed to provide protection against in the design. Adequate performance of those systems has been assessed in connection with plant licensing. Assumptions concerning their performance are re-assessed in the stress tests and they should be shown as provisions in place. It is recognised that all measures taken to protect reactor core or spent fuel integrity or to protect the reactor containment integrity constitute an essential part of the defence-indepth, as it is always better to prevent accidents from happening than to deal with the consequences of an occurred accident.

Process to perform the "stress tests" and their dissemination

The licensees have the prime responsibility for safety. Hence, it is up to the icensees to perform the reassessments, and to the regulatory bodies to independently review them.

The timeframe is as follows:

The national regulator will initiate the process at the latest on June 1 by sending requirements to the licensees.

	Progress report	**Final report**
Licensee report	August 15	October 31
National report	September 15	December 31

- The final national reports will be subjected to the peer review process described below.
- The European Commission, with the support of ENSREG, will present a progress report to the EU Council for the meeting scheduled on 9th December 2011 and a consolidated report to the to the EU Council for the meeting scheduled for June 2012.

Due to the timeframe of the stress test process, some of the engineering studies supporting the licensees' assessment may not be available for scenarios not included in the current design. In such cases engineering judgment is used.

During the regulatory reviews, interactions between European regulators will be necessary and could be managed through ENSREG. Regulatory reviews should be peer reviewed by other regulators. ENSREG will put at the disposal of all peer reviews the expertise necessary to ensure consistency of peer reviews across the EU and its neighbours.

Peer review process

In order to enhance credibility and accountability of the process the EU Council asked that the national reports should be subjected to a peer review process. The main purpose of the national reports will be to draw conclusions from the licensees' assessment using the agreed methodology. The peer teams will review the fourteen national reports of Member States that presently operate nuclear power plants and of those neighbouring countries that accept to be part of the process.

- **Team composition.** ENSREG and the Commission shall agree on team composition. The team should be kept to a working size of seven people, one of whom should act as a chairperson and a second one as rapporteur. Two members of each team will be permanent members with the task to ensure overall consistency. The Commission will be part of the team. Members of the team whose national facilities are under review will not be part of that specific review. The country subject to review has to agree on the team composition. The team may be extended to experts from third countries.
- **Methodology.** In order to guarantee the rigor and the objectivity of any peer review, the national regulator under review should give the peer review team access to all necessary information, subject to the required security clearance procedures, staff and facilities to enable the team, within the limited time available.
- **Timing.** Reviews should start immediately when final national reports become available. The peer reviews shall be completed by the end of April 2012.

Transparency

National regulatory authorities shall be guided by the "principles for openness and transparency" as adopted by ENSREG in February 2011. These principles shall also apply to the EU "stress tests".

The reports should be made available to the public in accordance with national legislation and international obligations, provided that this does not jeopardize other interests such as, inter alia, security, recognized in national legislation or international obligations.

The peer will review the conclusions of each national report and its compliance with the methodology agreed. Results of peer reviews will be made public.

Results of the reviews should be discussed both in national and European public seminars, to which other stakeholders (from non nuclear field, from non governmental organizations, etc) would be invited.

Full transparency but also an opportunity for public involvement will contribute to the

EU "stress tests" being acknowledged by European citizens.

Technical scope of the "stress tests"

The existing safety analysis for nuclear power plants in European countries covers a large variety of situations. The technical scope of the stress tests has been defined considering the issues that have been highlighted by the events that occurred at Fukushima, including combination of initiating events and failures. The focus will be placed on the following issues:

a) Initiating events

- Earthquake
- Flooding

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

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

c) Severe accident management issues

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

b) and c) are not limited to earthquake and tsunami as in Fukushima: flooding will be included regardless of its origin. Furthermore, bad weather conditions will be added.

Furthermore, the assessment of consequences of loss of safety functions is relevant also if the situation is provoked by indirect initiating events, for instance large disturbance from the electrical power grid impacting AC power distribution systems or forest fire, airplane crash.

The review of the severe accident management issues focuses on the licensee's provisions but it may also comprise relevant planned off-site support for maintaining the safety functions of the plant. Although the experience feedback from the Fukushima accident may include the emergency preparedness measures managed by the relevant off-site services for public protection (fire-fighters, police, health services….), this topic is out of the scope of these stress tests.

The next sections of this document set out:

- general information required from the licensees;
- issues to be considered by the licensees for each considered extreme situation.

General aspects

Format of the report

The licensee shall provide one document for each site, even if there are several units on the same site. Sites where all NPPs are definitively shutdown but where spent fuel storages are still in operation shall also be considered.

In a first part, the site characteristics shall be briefly described:
- location (sea, river);
- number of units;
- license holder

The main characteristics of each unit shall be reflected, in particular:
- reactor type;
- thermal power;
- date of first criticality;
- presence of spent fuel storage (or shared storage).

Safety significant differences between units shall be highlighted.
The scope and main results of Probabilistic Safety Assessments shall be provided.

In a second part, each extreme situation shall be assessed following the indications given below.

Hypothesis

For existing plants, the reassessments shall refer to the plant as it is currently built and operated on June 30, 2011. For plants under construction, the reassessments shall refer to the licensed design.

The approach should be essentially deterministic: when analysing an extreme scenario, a progressive approach shall be followed, in which protective measures are sequentially assumed to be defeated.

The plant conditions should represent the most unfavourable operational states that are permitted under plant technical specifications (limited conditions for operations). All operational states should be considered. For severe accident scenarios, consideration of non-classified equipment as well as realistic assessment is possible.

All reactors and spent fuel storages shall be supposed to be affected at the same time.

Possibility of degraded conditions of the site surrounding area shall be taken into account.

Consideration should be given to:
- automatic actions;
- operators actions specified in emergency operating procedures;
- any other planned measures of prevention, recovery and mitigation of accidents;

Information to be included

Three main aspects need to be reported:

- Provisions taken in the design basis of the plant and plant conformance to its design requirements;
- Robustness of the plant beyond its design basis. For this purpose, the robustness (available design margins, diversity, redundancy, structural protection, physical separation, etc) of the safety-relevant systems, structures and components and the effectiveness of the defence-in-depth concept have to be assessed. Regarding the robustness of the installations and measures, one focus of the review is on identification of a step change in the event sequence (cliff edge effect[1]) and, if necessary, consideration of measures for its avoidance.
- any potential for modifications likely to improve the considered level of defence-in-depth, in terms of improving the resistance of components or of strengthening the independence with other levels of defence.

In addition, the licensee may wish to describe protective measures aimed at avoiding the extreme scenarios that are envisaged in the stress tests in order to provide context for the stress tests. The analysis should be complemented, where necessary, by results of dedicated plant walk down.

To this aim, the licensee shall identify:

- the means to maintain the three fundamental safety functions (control of reactivity, fuel cooling, confinement of radioactivity) and support functions (power supply, cooling through ultimate heat sink), taking into account the probable damage done by the initiating event and any means not credited in the safety demonstration for plant licensing;
- possibility of mobile external means and the conditions of their use;
- any existing procedure to use means from one reactor to help another reactor;
- dependence of one reactor on the functions of other reactors on the same site.

As for severe accident management, the licensee shall identify, where relevant:

- the time before damage to the fuel becomes unavoidable. For PWR and BWR, if the core is in the reactor vessel, indicate time before water level reaches the top of the core, and time before fuel degradation (fast cladding oxidation with hydrogen production);
- if the fuel is in the spent fuel pool, the time before pool boiling, time up to when adequate shielding against radiation is maintained, time before water level reaches the top of the fuel elements, time before fuel degradation starts;

Supporting documentation

Documents referenced by the licensee shall be characterised either as:

- validated in the licensing process;
- not validated in the licensing process but gone through licensee's quality assurance program;
- not one of the above.

[1] Example: exhaustion of the capacity of the batteries in the event of a station blackout

Earthquake

I. Design basis

a) Earthquake against which the plant is designed :

- Level of the design basis earthquake (DBE) expressed in terms of peak ground acceleration (PGA) and reasons for the choice. Also indicate the DBE taken into account in the original licensing basis if different;
- Methodology to evaluate the DBE (return period, past events considered and reasons for choice, margins added…), validity of data in time;
- Conclusion on the adequacy of the design basis.

b) Provisions to protect the plant against the DBE

- Identification of the key structures, systems and components (SSCs) which are needed for achieving safe shutdown state and are supposed to remain available after the earthquake;
- Main operating provisions (including emergency operating procedure, mobile equipment…) to prevent reactor core or spent fuel damage after the earthquake;
- Were indirect effects of the earthquake taken into account, including:
    1. Failure of SSCs that are not designed to withstand the DBE and that, in loosing their integrity could cause a consequential damage of SSCs that need to remain available (e.g. leaks or ruptures of non seismic pipework on the site or in the buildings as sources of flooding and their potential consequences);
    2. Loss of external power supply;
    3. Situation outside the plant, including preventing or delaying access of personnel and equipment to the site.

c) Plant compliance with its current licensing basis:

- Licensee’s general process to ensure compliance (e.g. , periodic maintenance, inspections, testing);
- Licensee’ process to ensure that off-site mobile equipment/supplies considered in emergency procedures are available and remain fit for duty;
- Any known deviation, and consequences of these deviations in terms of safety; planning of remediation actions;
- Specific compliance check already initiated by the licensee following Fukushima NPP accident.

II. Evaluation of the margins

d) Based on available information (which could include seismic PSA, seismic margin assessment or other seismic engineering studies to support engineering judgement), give an evaluation of the range of earthquake severity above which loss of fundamental safety functions or severe damage to the fuel (in vessel or in fuel storage) becomes unavoidable.

- Indicate which are the weak points and specify any cliff edge effects according to earthquake severity.
- Indicate if any provisions can be envisaged to prevent these cliff edge effects or to increase robustness of the plant (modifications of hardware, modification of procedures, organisational provisions…).

e) Based on available information (which could include seismic PSA, seismic margin assessment or other seismic engineering studies to support engineering judgement), what is the range of earthquake severity the plant can withstand without losing confinement integrity.

f) Earthquake exceeding DBE and consequent flooding exceeding DBF

- Indicate whether, taking into account plant location and plant design, such situation can be physically possible. To this aim, identify in particular if severe
  damages to structures that are outside or inside the plant (such as dams, dikes, plant buildings and structures) could have an impact of plant safety.
- Indicate which are the weak points and failure modes leading to unsafe plant conditions and specify any cliff edge effects. Identify which buildings and equipment will be impacted.
- Indicate if any provisions can be envisaged to prevent these cliff edge effects or to increase robustness of the plant (modifications of hardware, modification of procedures, organisational provisions…).

Flooding

I. Design basis

a) Flooding against which the plant is designed :

- Level of the design basis flood (DBF) and reasons for choice. Also indicate the DBF taken into account in the original licensing basis if different;
- Methodology to evaluate the DBF (return period, past events considered and reasons for choice, margins added…). Sources of flooding (tsunami, tidal, storm surge, breaking of dam…), validity of data in time;
- Conclusion on the adequacy of the design basis.

b) Provisions to protect the plant against the DBF

- Identification of the key SSCs which are needed for achieving safe shutdown state and are supposed to remain available after the flooding, including:
  - o Provisions to maintain the water intake function;
  - o Provisions to maintain emergency electrical power supply;
- Identification of the main design provisions to protect the site against flooding (platform level, dike…) and the associated surveillance programme if any;
- Main operating provisions (including emergency operating procedure, mobile equipment, flood monitoring, alerting systems…) to warn of, then to mitigate the effects of the flooding, and the associated surveillance programme if any;
- Were other effects linked to the flooding itself or to the phenomena that originated the flooding (such as very bad weather conditions) taken into account, including:
  - o Loss of external power supply;
  - o Situation outside the plant, including preventing or delaying access of personnel and equipment to the site.

c) Plant compliance with its current licensing basis:

- Licensee's general process to ensure compliance (e.g., periodic maintenance, inspections, testing);
- Licensee's process to ensure that off-site mobile equipment/supplies considered in emergency procedures are available and remain fit for duty;
- Any known deviation and consequences of these deviations in terms of safety; planning of remediation actions;
- Specific compliance check already initiated by the licensee following Fukushima NPP accident.

II. Evaluation of the margins

d) Based on available information (including engineering studies to support engineering judgement), what is the level of flooding that the plant can withstand without severe damage to the fuel (core or fuel storage)?

- Depending on the time between warning and flooding, indicate whether additional protective measures can be envisaged/implemented.
- Indicate which are the weak points and specify any cliff edge effects. Identify which buildings and which equipment will be flooded first.
- Indicate if any provisions can be envisaged to prevent these cliff edge effects or to increase robustness of the plant (modifications of hardware,

modification of procedures, organisational provisions…).

Loss of electrical power and loss of the ultimate heat sink

Electrical AC power sources are:
- o off-site power sources (electrical grid);
- o plant generator;
- o ordinary back-up generators (diesel generator, gas turbine…);
- o in some cases other diverse back-up sources.

Sequential loss of these sources has to be considered (see a) and b) below).

The ultimate heat sink (UHS) is a medium to which the residual heat from the reactor is transferred. In some cases, the plant has the primary UHS, such as the sea or a river, which is supplemented by an alternate UHS, for example a lake, a water table or the atmosphere. Sequential loss of these sinks has to be considered (see c) below).

a) Loss of off-site power (LOOP[2])
- Describe how this situation is taken into account in the design and describe which internal backup power sources are designed to cope with this situation.
- Indicate for how long the on-site power sources can operate without any external support.
- Specify which provisions are needed to prolong the time of on-site power supply (refueling of diesel generators…).
- Indicate any envisaged provisions to increase robustness of the plant (modifications of hardware, modification of procedures, organisational provisions…).

For clarity, systems such as steam driven pumps, systems with stored energy in gas tanks etc. are considered to function as long as they are not dependent of the electric power sources assumed to be lost and if they are designed to withstand the initiating event (e.g. earthquake)

b) Loss of off-site power and of on-site backup power sources (SBO) Two situations have to considered:
- LOOP + Loss of the ordinary back-up source;
- LOOP + Loss of the ordinary back-up sources + loss of any other diverse back- up sources.

For each of these situations:
- Provide information on the battery capacity and duration.
- Provide information on design provisions for these situations.
- Indicate for how long the site can withstand a SBO without any external support before severe damage to the fuel becomes unavoidable.
- Specify which (external) actions are foreseen to prevent fuel degradation:
  - o equipment already present on site, e.g. equipment from another reactor;

[2] All offsite electric power supply to the site is lost. The offsite power should be assumed to be lost for several days. The site is isolated from delivery of heavy material for 72 hours by road, rail or waterways. Portable
light equipment can arrive to the site from other locations after the first 24 hours.
11/

- assuming that all reactors on the same site are equally damaged, equipment
- available off-site;
- near-by power stations (e.g. hydropower, gas turbine) that can be aligned to provide power via a dedicated direct connection;
- time necessary to have each of the above systems operating;
- availability of competent human resources to make the exceptional connections;
- identification of cliff edge effects and when they occur.

- Indicate if any provisions can be envisaged to prevent these cliff edge effects or to increase robustness of the plant (modifications of hardware, modification of procedures, organisational provisions…)

c) Loss of primary ultimate heat sink (UHS[3])

- Provide a description of design provisions to prevent the loss of the UHS (e.g. various water intakes for primary UHS at different locations, use of alternative UHS, …)

Two situations have to be considered:

- Loss of primary ultimate heat sink (UHS), i.e. access to water from the river or the sea;
- Loss of primary ultimate heat sink (UHS) and the alternate UHS.

For each of these situations:

- Indicate for how long the site can withstand the situation without any external support before damage to the fuel becomes unavoidable:
  Provide information on design provisions for these situations.
- Specify which external actions are foreseen to prevent fuel degradation:
  - equipment already present on site, e.g. equipment from another reactor;
  - assuming that all reactors on the same site are equally damaged, equipment available off-site;
  - time necessary to have these systems operating;
  - availability of competent human resources;
  - identification of cliff edge effects and when they occur.
- Indicate if any provisions can be envisaged to prevent these cliff edge effects or to increase robustness of the plant (modifications of hardware, modification of procedures, organisational provisions…).

d) Loss of the primary UHS with SBO

- Indicate for how long the site can withstand a loss of “main” UHS + SBO without any external support before severe damage to the fuel becomes unavoidable
- Specify which external actions are foreseen to prevent fuel degradation:
  - equipment already present on site, e.g. equipment from another reactor;
  - assuming that all reactors on the same site are equally damaged,

[3] The connection with the primary ultimate heat sink for all safety and non safety functions is lost. The site is isolated from delivery of heavy material for 72 hours by road, rail or waterways. Portable light equipment can arrive to the site from other locations after the first 24 hours.

equipment available off site;
  - availability of human resources;
  - time necessary to have these systems operating;
  - identification of when the main cliff edge effects occur.
- Indicate if any provisions can be envisaged to prevent these cliff edge effects or to increase robustness of the plant (modifications of hardware, modification of procedures, organisational provisions…)

Severe accident management

This chapter deals mostly with mitigation issues. Even if the probability of the event is very low, the means to protect containment from loads that could threaten its integrity should be assessed. Severe accident management, as forming the last line of defense-in-depth for the operator, should be consistent with the measures used for preventing the core damage and with the overall safety approach of the plant.

a) Describe the accident management measures currently in place at the various stages of a scenario of loss of the core cooling function:

- before occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes;
  - last resorts to prevent fuel damage
  - elimination of possibility for fuel damage in high pressure
- after occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes;
- after failure of the reactor pressure vessel/a number of pressure tubes;

b) Describe the accident management measures and plant design features for protecting integrity of the containment function after occurrence of fuel damage

- prevention of H2 deflagration or H2 detonation (inerting, recombiners, or igniters), also taking into account venting processes;
  - prevention of over-pressurization of the containment; if for the protection of the containment a release to the environment is needed, it should be assessed, whether this release needs to be filtered. In this case, availability of the means for estimation of the amount of radioactive material released into the environment should also be described;
- prevention of re-criticality
- prevention of basemat melt through
- need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity

c) Describe the accident management measures currently in place to mitigate the consequences of loss of containment integrity.

d) Describe the accident management measures currently in place at the various stages of a scenario of loss of cooling function in the fuel storage (the following indications relate to a fuel pool):

- before/after losing adequate shielding against radiation;
- before/after occurrence of uncover of the top of fuel in the fuel pool
- before/after occurrence of fuel degradation (fast cladding oxidation with hydrogen production) in the fuel pool.

For a) b) c) and d), at each stage:

- identify any cliff edge effect and evaluate the time before it;
- assess the adequacy of the existing management measures, including the procedural guidance to cope with a severe accident, and evaluate the potential for additional measures. In particular, the licensee is asked to consider:
  - o the suitability and availability of the required instrumentation;

o the habitability and accessibility of the vital areas ot the plant (the control room, emergency response facilities, local control and sampling points, repair possibilities);
o potential H2 accumulations in other buildings than containment;

The following aspects have to be addressed:

- Organisation of the licensee to manage the situation, including:
  - o staffing, resources and shift management;
  - o use of off-site technical support for accident and protection management (and contingencies if this becomes unavailable);
  - o procedures, training and exercises;
- Possibility to use existing equipment;
- Provisions to use mobile devices (availability of such devices, time to bring them on site and put them in operation, accessibility to site);
- Provisions for and management of supplies (fuel for diesel generators, water…);
- Management of radioactive releases, provisions to limit them;
  Management of workers' doses, provisions to limit them;
- Communication and information systems (internal, external).
  Long-term post-accident activities.

The envisaged accident management measures shall be evaluated considering what the situation could be on a site:

- Extensive destruction of infrastructure around the plant including the communication
  - facilities (making technical and personnel support from outside more difficult);
- Impairment of work performance (including impact on the accessibility and habitability of the main and secondary control rooms, and the plant emergency/crisis centre) due to high local dose rates, radioactive contamination and destruction of some facilities on site;
- Feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods);
- Unavailability of power supply;
- Potential failure of instrumentation;
- Potential effects from the other neighbouring plants at site.

The licensee shall identify which conditions would prevent staff from working in the main or secondary control room as well as in the plant emergency/crisis centre and what measures could avoid such conditions to occur.

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