Ministry of Economic Affairs,
Agriculture & Innovation

Netherlands' National Report on the Post-Fukushima Stress Test

for the Borssele Nuclear Power Plant

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Netherlands’ National Report

On the post-Fukushima stress test for the Borssele Nuclear Power Plant

Ministry of Economic affairs, Agriculture & Innovation
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Abstract

This is the National Report of the Kingdom of the Netherlands on the post-Fukushima 'stress test' of the Borssele (one unit) Nuclear Power Plant, the KCB. This report complies with the guidelines published by ENSREG in May (objectives & scope) and October 2011 (structure of report) for National Reports.

The operator of the KCB has submitted a Licensee Report to the regulatory body, that addresses all topics prescribed in the ENSREG guidelines for the 'stress test' and meets the prescribed format.

The National Report presents conclusions about licensee's compliance with its design basis. The conclusions are based on the Licensee Report as well as on several decades of regulatory oversight, including regulatory inspections, evaluations of various applications for modification of the licences, regulatory control of the special Long-Term Operation programme and the various extensive Periodic Safety Reviews.

The National Report presents conclusions on the safety margins identified in the Licensee Report.

The National report notes the measures proposed and considered in the Licensee Report. In principle, the regulatory body can endorse various of these measures, but further assessment is needed to establish the effectiveness of these. The regulatory body proposes additional topics suitable for (more detailed) assessment.

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List of Symbols and Abbreviations

AC	Alternating Current
ACC	Alarm Coördinatie Centrum (Alarm Coordination Centre)
ADBE	Amplitude (design basis earthquake)
AM	Accident Management
APC	Air Plane Crash
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
b&f	bleed and feed
CCB	Conventionele Centrale Borssele (Borssele Coal-fired Power Plant)
CSA	Complementary Safety margin Assessment
CVCS	Chemical and Volume Control System
DBE	Design Basis Earthquake
DBF	Design Basis Flood
DC	Direct Current
DG	Diesel Generator
DHR	Decay Heat Removal
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EDMG	Extensive Damage Mitigation Guidelines
EL&I	'Ministerie van Economische zaken, Landbouw & Innovatie'; ministry of economic affairs, agriculture & innovation
EMS	European Macroseismic Scale
ENSREG	European Nuclear Safety Regulator Group
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EPZ N.V.	Elektriciteits-Produktiemaatschappij Zuid-Nederland EPZ
ERO	Emergency Response Organisation
EU	Europian Union
10EVA13	Current 10 yearly safety evaluation; periodic safety review (PSR)
EY	Diesel Generators
I&M	'Ministerie van Infrastructuur & Milieu'; ministry of infrastructure & the environment
IAEA	International Atomic Energy Agency
FHP	Functie Herstel Procedure (Function Restoration Procedure)
FRG	Functional Restoration Guidelines
HCLPF	High Confidence Low Probability of Failure
I & C	Instrumentation and Control
KCB	Kerncentrale Borssele (NPP Borssele)
KFD	Kernfysische Dienst (Nuclear Safety Department)
KNMI	Koninklijk Nederlands Meteorologisch Instituut
KTA	Kerntechnische Ausschuss
KWU	Kraftwerk Union
LOCA	Loss-Of-Coolant Accident
LOOP	Loss Of Offsite Power
LPG	Liquefied Petroleum Gas
LPSI	Low pressure Safety Injection
LPAUS	Loss of Primary and Alternate Ultimate Heat Sink
LPUHS	Loss of Primary Ultimate Heat Sink
LUHS	Loss of Ultimate Heat Sink
MCCI	Molten Core-Concrete Interaction

MCR	Main Control Room
MMI	Modified Mercally Intensity
MOX	Mixed Oxides
MSK	Medvedev, Sponheuer en Karnik
mSv	milliSievert
MWe	Megawatts Electrical
MWth	Megawatts Thermal
NAP	Normaal Amsterdams Peil
NBP	Nood Bedienings Procedure (Emergency Operating Procedure)
N.B.P.	Nucleair Basis Peil
NEN	Nederlandse Norm
N.O.P.	Nucleair Ontwerp Peil (Nuclear Design Level)
NPK	Nationaal Plan Kernongevallenbestrijding
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NS 1	Nood Stroom net 1 (Emergency Grid 1)
NS 2	Nood Stroom net 2 (Emergency Grid 2)
NUREG Reports	Nuclear Regulatory reports – A Series of the United States Nuclear Regulatory Commission
PAR	Passive Autocatalytic Recombiner
PGA	Peak Ground Acceleration
PORV	Power-Operated Relief Valve
POS	Plant Operational State
Ppm	parts per million
PRA	Probabilistic Risk Analysis
PSA	Probabilistic safety Analysis
PWR	Pressurised Water Reactor
RCS	Reactor Coolant System
RESA	Reaktor Schnell Abschaltung (Reactor Scram)
RHR	Residual Heat Removal
ROT	Regional Operational Team
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWS	Rijkswaterstaat
SAG	Severe Accident Guidelines
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout
SCG	Severe Challenge Guideline
SCRAM	Security Control Rod Axe Man
SED	Site Emergency Director
SITRAP	SITuation REPort (in Dutch SITRAP)
SFP	Spent Fuel Pool
SG	Steam Generator
SMA	Seismic Margin Assessment
SOB	Splijtstof Opslag Bassin (Spent Fuel Pool, SFP)
SOER	Significant Operating Experience Report
SSCs	Structures, Systems and Components
TIP	Technisch Informatie Pakket (Technical Information Package)
TS	Technische Specificaties (Technical Specifications)
UK	United Kingdom
UPS	Uninterrupted Power Supply
US	United States
WANO	World Association of Nuclear Operators
WOG	Westinghouse Owner's Group

System code description of safety relevant systems

EY	Emergency Grid 1 NS 1
EY	Emergency Grid 2 NS 2
RA	Main steam system
RL	Main and auxiliary feedwater system (denoted as RL-M(ain) and RL-E(mergency) respectively)
RM	Main condensate system
RS	Backup feed water system
RY	Steam generator letdown system
RZ	Demin water supply system
SF	Turbine bypass system
TA	Volume control system
TB	Chemical control system
TC	Coolant cleaning and degassing system
TD	Coolant storage and regeneration system
TE	Backup residual heat removal system
TF	Component cooling water system
TG	Spent fuel pool cooling system including TG080
TJ	Safety injection system & residual heat removal system
TL	Nuclear ventilation system (inclusive the filter for containment venting)
TM	Biological barrier cooling system
TN	Demineralised water distribution system
TP	Gas and compressed air supply system
TR	Radioactive waste water system
TS	Radioactive gas treatment system (including TS100 the H2-recombiners)
TT	Radioactive solid waste system
TV	Nuclear sampling system
TW	Backup coolant makeup system
TY	Plant drainage and plant degassing system
TZ	Nuclear building water drainage system
UA	Demineralised water plant
UF	High pressure fire extinguishing system including sprinkler system
UG	Transformer fire extinguishing system
UJ	Low pressure fire extinguishing system including fine water spray system
UV	Chilled water system
UW	Heating ventilation and airconditioning
UX	Halon and CO2 fire extinguishing system
VA	Cooling water filtering system
VC	Main cooling water system
VE	Backup cooling water system
VF	Conventional emergency cooling water system
VG	Conventional component cooling water system
XA	Containment
XQ	Atmosphere radiation measurement system
YA	Reactor coolant system
YB	Steam generators
YC	Reactor vessel
YD	Reactor coolant pump
YP	Pressure control system
YQ	Nuclear instrumentation
YS	Control rods
YX	Neutron flux measurement outside the core system
YZ	Reactor protection system

Introduction

This section starts with the background of the "stress test" and then sets out the purpose and scope of this document: '*Netherlands' National Report on the Post-Fukushima Stress Test for the Borssele Nuclear Power Plant'*. It then continues with the intended audience, the relationship of the 'stress test' with other safety evaluations, the organisation of the regulatory review and its planning, the regulatory body, and an overview of the national nuclear programme in the Netherlands. The introduction finishes with a description of the structure of the report.

Background

March 11 2011, Japan was struck by an earthquake of enormous magnitude, followed by a devastating tsunami that affected large parts of its eastern coast. This natural disaster killed thousands of people, and caused enormous damage to Japanese cities and infrastructure.

After the earthquake, the Japanese Nuclear Power Plant Fukushima Dai-ichi shut down automatically. However it failed to adequately maintain all of its safety functions after been hit by the tsunami that was initiated by the earthquake.

Not all causes of the accident in Japan have been analysed in detail yet. Thus not all lessons to be learned are known at this moment. The stress test can contribute to the learning process.

March 24th and 25th, the European Council declared that:

'*the safety of all EU nuclear plants should be reviewed on the basis of a comprehensive and transparent risk assessment ("stress test");..*'

The Western European Nuclear Regulators Association (WENRA) end of March 2011 produced a basis for a technical definition of a "stress test" and how it should be applied to nuclear facilities in Europe. The WENRA proposals were used as a basis for the official specifications[1] issued by the European Nuclear Safety Regulatory Group (ENSREG), issued on May 13th 2011. These were endorsed by EU Commissioner Öttinger on May 24th 2011.

First the licensees as being primarily responsible for the safety of their NPPs will produce their reports ('Licensee Reports') on the "stress test" of their facilities, in accordance with the ENREG specifications. In a second step, the regulatory bodies will evaluate these reports, and report their findings in 'National Reports' to be submitted to the EC. Later in a third step there will be a peer review of these reports.
ENSREG has issued templates for the "stress test" reports to be produced. These is a template to used by the licensees as well as one to be used by the regulatory bodies[2].

Purpose and Scope of this National Report

The operator EPZ of the Nuclear Power Plant (NPP) in Borssele, the Netherlands, performed what it calls a 'Complementary Safety margin Assessment' (CSA), according to the ENSREG guidelines for the post-Fukushima 'stress test' for NPPs. This exercise yielded – in ENSREG terminology – a Licensee Report. This report can be found on:

http://www.kerncentrale.nl/resultatenrobuustheidsonderzoek/EN/

On top of the ENSREG specifications, the Dutch government has demanded[3] that man-made events are also included in the Dutch stress test as possible initiating events.

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[1] Declaration of ENSREG, agreed with the European Commission on the 13th of May 2011, with Annex I : 'EU "Stress tests" specifications'

[2] Post-Fukushima "Stress Tests" of European Nuclear Power Plants – Contents and Format of National Reports, ENSREG, Draft-7, October 3rd, 2011

This National Report provides a regulatory review of the Licensee Report. It both summarizes licensee's findings as well as evaluates the completeness and adequacy of licensee's assessments. In summary, the National Report:

1) Evaluates the descriptions and assessments produced by the licensee in terms of adequacy and correctness;
2) Evaluates the conformance to the current design basis of the plant;
3) Shows the safety margins of the NPP whereas;
    a. the stress test is in fact a targeted reassessment of those safety margins, bringing more insight into the robustness of the NPP;
    b. the stress test imposes postulated extreme scenarios on the NPP without taking account of probabilities of the postulated extreme initiating events;
4) Notes measures (physical and procedural) and research considered or proposed by the licensee;
5) Notes measures (physical and procedural) and research considered or proposed by the regulatory body.
6) Only applies to the NPP Borssele; other nuclear facilities in the Netherlands are out of scope of this National Report.

On request of the regulatory body, in addition to the ENSREG-specified scenarios, licensee has analysed the robustness of its plant regarding 'deliberate disturbances'. This effort is beyond the scope of the European stress test and thus is not reported in the Netherlands' National Report.

Intended Audience

This National Report is mainly targeted at regulatory bodies of other EU Member States, to enable them to perform a peer review. As agreed the report is written in English. Nevertheless, it will be made available to the general public. The report and a comprehensive Dutch summary will also be sent to the Dutch Parliament.

Relationship 'Stress Test' with other Safety Evaluations

To date, the evaluation of the safety of the NPP Borssele and other nuclear installations in the Netherlands, involves deterministic as well as probabilistic studies. The 'stress test' is focussed on deterministic analyses of postulated scenarios with combinations of extreme challenges, in which the sequential loss of safety functions and lines of defence is assumed, irrespective of the probability of such losses.

Such extreme scenarios have not been considered in the past licensing and oversight procedures. Instead, the licensee was obliged to show how he assured that adverse effects of such extreme challenges could practically be excluded. The current 'stress test' will allow a review of the validity of current design assumptions and of possible areas for additional protective measures to further increase robustness.

Organisation of the Regulatory Review

The review of the Licensee Report (CSA) was undertaken by the regulatory body that oversees all activities that are governed by the Nuclear Energy Act. It coordinated the supporting review efforts of an Expert Group of staff from several ministries. The regulatory body also received support from German Technical Support Organisation GRS[4].

The regulatory body adopted the guidelines of ENSREG. It communicated these with the licensees in the Netherlands, after the addition of some specific requirements.

The progress of the Licensee Report was monitored during periodic meetings with the licensee.

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[3] Letter to utility EPZ from ministry of economic affairs, agriculture and innovation, ETM/ED/11074538, June 1, 2011.
[4] GRS, 'Gesellschaft für Reaktorsicherheit', a German Technical Support Organisation.

After publication of the Licensee Report, the regulatory body started its review of the report. During this review period, there has been contact with the licensee for clarification purposes. A site visit was part of the process.

During the review, the regulatory body could rely on sources, additional to the Licensee Report. Examples are the Technical Information Package (TIP) of the NPP, the Safety Report, strategic maintenance and surveillance plans and extensive maintenance and in-service inspection programmes.

Planning of the Review

October 31st, 2011, the licensee, EPZ, submitted its Licensee Report on the post-Fukushima 'stress test' to the regulatory body. In the period November through December 2011 the Regulatory Body performed the review of the Licensee Report and edited the National Report.
In the period January – April 2012, the international peer review of the various National Reports of the EU Member States will be undertaken. The National Reports will form the primary basis for the review, but for reference purposes the reviewers may also use the Licensee Reports. In addition regulatory bodies of Member States will also visit each other's NPP sites and have discussions with the hosting regulatory body as well as with the licensee(s).

Regulatory Body

All nuclear facilities in the Netherlands, including the NPP of Borssele, operate under licence, awarded after a safety assessment has been carried out. The licence is granted by the regulatory body under the Nuclear Energy Act.

For the purpose of this report, the 'regulatory body' is the authority designated by the government as having legal authority for conducting the regulatory process, including issuing licences, and thereby regulating nuclear, radiation, radioactive waste and transport safety.

In the Netherlands, the ministry of Economic affairs, Agriculture & Innovation (EL&I[5]) is the principal authority for conducting the regulatory process under the Nuclear Energy Act. However two ministries are involved, constituting the regulatory body for all things nuclear in the Netherlands:

- Ministry of EL&I, 'Directoraat voor Nieuwe kernenergie en veiligheid' (NKV), i.e. directorate for nuclear new build and nuclear safety, is involved in the preparation of legislation, formulating policies and licensing;
- The inspectorate the 'Kernfysische dienst' (KFD) is in fact the nuclear inspectorate. Within the general responsibility of the Ministry of EL&I it is responsible for the assessment and inspection of nuclear facilities. The KFD is an organisational subdivision of the 'Inspectorate for the environment and transport', which is a general inspection branch of the ministry of I&M[6].

Nuclear Programme

The Netherlands has a small nuclear programme, with only one NPP in operation, producing about 4% of the country's electrical power consumption. The programme features a number of steps of the nuclear fuel cycle. Some of the Dutch nuclear businesses have a global impact. Urenco supplies about 25% of world-demand for low-enriched uranium, of which its plant in Almelo, the Netherlands, provides a third. The company ET-NL in Almelo supplies all centrifuges for the enrichment plants of Urenco and AREVA – world-wide. The High Flux Reactor (HFR) in Petten, on average supplies 70% of the European demand for radio-isotopes – and no less than 30% of the global demand. The Nuclear Research & consultancy Group (NRG) operates the HFR and several nuclear research facilities on their site in Petten and in addition provides consultancy services to clients on several continents. In addition, scientists of the Dutch universities and NRG participate in many international nuclear research programmes.

[5] Dutch: 'Economische Zaken, Landbouw & Innovatie', EL&I
[6] Dutch: 'Infrastructuur & Milieu', i.e. Infrastructure & the Environment.

Structure of the Report

This report follows the guidance[7] provided by ENSREG on the contents and format of national reports on the post-Fukushima stress tests, dated 3rd of October 2011. The numbering of its chapters and sections corresponds to that of the guidance provided by ENSREG.

This report is designed to be a 'stand alone' document to facilitate the international peer review by regulatory bodies of other EU nations, undertaking the so-called 'stress test', although the Licensee Report will be available for further reference to technical details.

The report offers a review of the situation in the Netherlands as compared with the obligations imposed by the Annex I, EU 'Stress Test' specifications.

Chapter 1 relates to general data on the site and its NPP in Borssele.

Chapter 2 describes and evaluates the analysis of the 'robustness' of the plant when faced with earthquakes.

Chapter 3 does the same for the initiating event flooding.

Chapter 4 considers extreme weather conditions.

Chapter 5 is about loss of electrical power and loss of ultimate heat sink.

Chapter 6 considers Severe Accident Management (SAM), its current organisation, its adequacy to limit the consequences of various postulated scenarios and options for improvements.

Chapter 7 summarizes the general conclusions of the regulatory review of the Licensee Report.

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[7] Post-Fukushima "Stress Tests"of European Nuclear Power Plants – Contents and Format of National Reports, ENSREG, Draft-7, Oktober 3rd, 2011

1. General data about the site and the NPP

The Borssele NPP is the single operating NPP in the Netherlands. It is also known as the 'Kerncentrale[8] Borssele', the KCB.

The Licensee Report is quite comprehensive in its description of the site and the installation and meets the requirements defined by ENSREG. The information presented in chapter 1 of the Licensee Report has been presented before in various licensing documents – and in even more detail.

The Licensee Report states that if somewhere in the future MOX fuel will be used, the safety of the NPP will remain comparable to the safety in the current situation in which only natural enriched uranium is used. The regulatory position is that this does not automatically imply that with a mixed[9] core, the results of the analyses presented in the Licensee Report will still remain valid for all ENSREG-postulated scenarios.

1.1 Brief description of the site characteristics

Geography

The KCB is located in the province of Zeeland, the Netherlands' most South-western province. The area of Zeeland mainly consists of a collection of former islands, connected by bridges, dykes, and other waterworks. Roughly one third of the total area of the province is water.

The KCB is located on the Northern shore of the river Westerschelde, about 1.4 km Northwest of the village Borssele. The area belongs to the municipality of Borsele.

The site is located directly behind the dyke of the river Westerschelde. Due to the open connection to the North Sea, near the site this river can be considered to be an estuary. The area around the site is mainly flat.

Hosted facilities on the site

The KCB shares its site with a conventional power plant[10] that is mainly coal-fired and five wind turbines[11]. The site is owned by utility N.V. EPZ who is also license holder of the KCB and the other facilities on the site.

Land use of nearby areas

The nearest village is found 1.4 km northwest of the site. Larger urban areas are found at 10 km distance or more.

The use of the area directly east and south of the EPZ-site is mainly agricultural.

On its north side the EPZ-site is bounded by industrial area among which is the seaport of 'Sloehaven' with heavy industries like an oil refinery and production facilities for phosphor and aluminium. Distances of these facilities to the EPZ-site range from 1 to 3 km.

[8] 'Kerncentrale' is the Dutch word for Nuclear Power Plant.
[9] Mixed core refers to a core containing more than one type of fuel.
[10] Gross capacity 427 MWe, net capacity 404 MWe. Also fuelled with biomass and phosphorus gas from neighbouring industry.
[11] In operation since 2005, installed capacity 11.75 MWe.

Transportation

The river Westerschelde features intense shipping with ship movements over 40,000 per year. Origin or destination in many cases is the port of Antwerp. Commodities shipped include dangerous materials like LPG[12], flammable liquids and liquefied ammonia.

The major highway is the A58 (E312) which is located at a distance of 7.2 km. The local road is the N254, located at about 500 meters distance.

The nearest railway line is found at a distance of 500 meters. A local yard and sideline from the main line services the local ports and industries.

A small airport 'Midden Zeeland Airport' is located about 10 km north of the EPZ-site. It is host to small civilian aircraft up to a weight of 5.7 tons.

There are several prescribed airways for en route flying that pass over the Netherlands. The A5 airway from the south flying to Schiphol airport near Amsterdam is located 20 km east of the EPZ-site. The B29 airway for flights from Brussels to London is located 20 km south of the site.

The closest military airbase[13] is 40 km northeast from the EPZ-site. In the Netherlands NPPs have restricted airspace for military air traffic. Ground dimensions are a square with sides 3.6 km and the minimum vertical height is 0.5 km.

The most nearby military facility is the ammunition depot Ritthem at a distance of 5.5 km.

Figure 1-1 Location of the KCB in the province of Zeeland

[12] Liquefied Petroleum Gas, LPG
[13] Woensdrecht in the province of Brabant. The province of Brabant is located east of the province of Zeeland.

Demography

The village of Borssele has 1,500 inhabitants and is located about 1.4 km northwest of the site. The cities of Vlissingen, Middelburg, Goes and Terneuzen are at distances of 10, 10, 15 and 16 km respectively. Their number of inhabitants are 44,660, 47,850, 36,000, and 54,830 respectively.

Figure 1-2 Aerial photograph of EPZ site. Front middle: the NPP, and left of it the coal-fired unit. Front left: outlet coolant, front right: intake coolant. Behind the EPZ site: industrial area.

1.1.1 Main characteristics of the unit

The emphasis in this section is on providing an overview of the main features of the KCB unit. For somewhat more detailed descriptions of Structures, Systems and Components (SSCs) important to safety, reference is made to section 1.1.2 and its various subsections.

The KCB is a single unit NPP owned by the utility 'N.V. EPZ'. The unit is also known as 'unit BS30'. It was designed and built by Kraftwerk Union (KWU) and started commercial operation in 1973. Since that date several major modernization projects have been carried out.

Currently, the KCB has a thermal power of 1365 MW_{th}, a gross capacity of 512 MW_e and a net capacity of 485 MW_e.

Reactor core and fuel

The reactor is fuelled with 121 fuel assemblies 15 x 15 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 has a mix of two enrichment levels: 4.0% and 4.4%.

EPZ has the intention to use Mixed Oxide fuel (MOX) in the NPP Borssele in the near future. For that reason EPZ has performed a licensing procedure to get licensed the use of MOX fuel elements with maximum of 5.41% (weight) fissionable plutonium. The maximum allowed number of MOX fuel elements in the reactor will be 48 (i.e. 40%).

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

Primary system and turbine system

The nuclear reactor of KCB is a pressurized water reactor (PWR) with two loops, each with one primary pump and one steam generator. Refer to Figure 1-3. The steam generators are the original ones and still are in good condition. They are tubed with Incoloy 800 and only a small fraction of the tubes have been plugged.

The turbines were retrofitted in 2006 for better thermal efficiency, adding 35 MW_e. The turbine generator installation consists of one high-pressure and three condensing dualflow 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 is usual with the KWU/Siemens plant designs, the condensate is collected and de-aerated in a large feed water accumulator.

The hydrogen-cooled generator has 21 kV coils and a 150 kV main transformer.

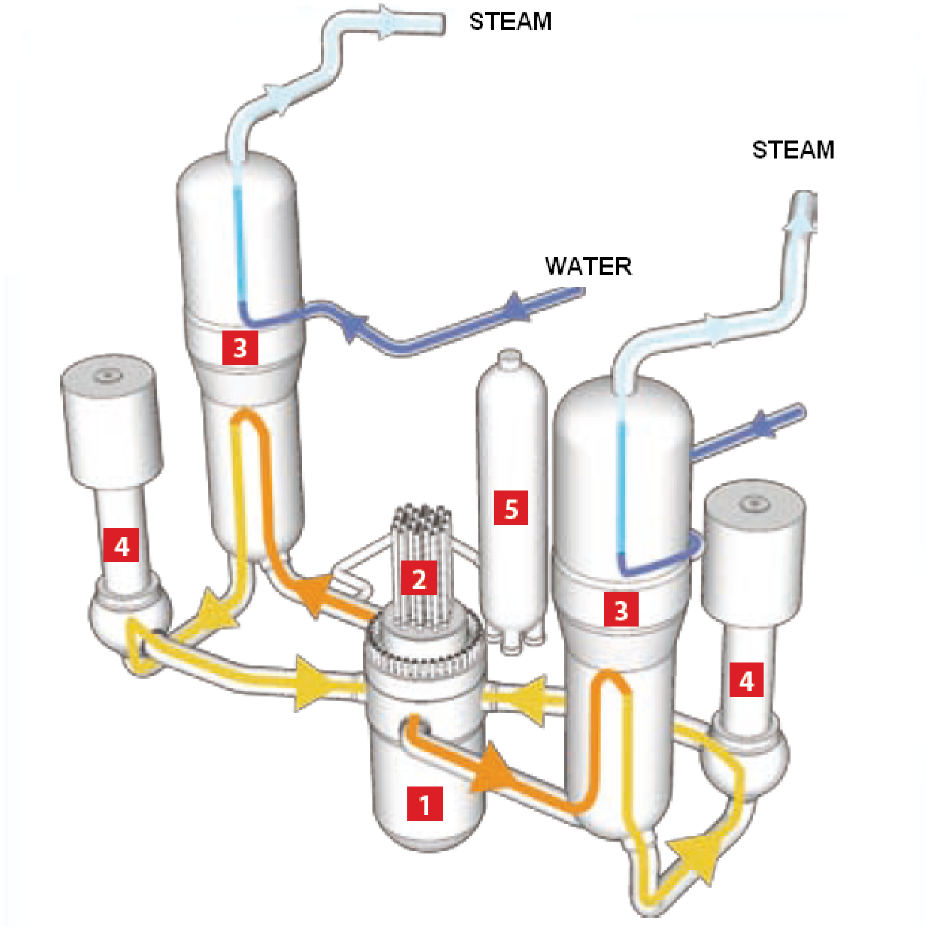

Figure 1-3 Primary system of KCB

1. Reactor vessel
2. Control rods
3. Steam generators
4. Main coolant pumps
5. Pressuriser

Containment

The containment is a 46-meter spherical steel shell, which is in turn encapsulated by the concrete reactor building. Refer to Figure 1-4. The spherical shell not only contains the reactor and steam generators, but also the spent fuel pool.

Spent fuel pool

There is no separate fuel storage facility outside the containment. The water in the spent fuel pool contains boron at 2,300 ppm. However boron is not required to guarantee a sub criticality of $K_{eff} < 0.95$.

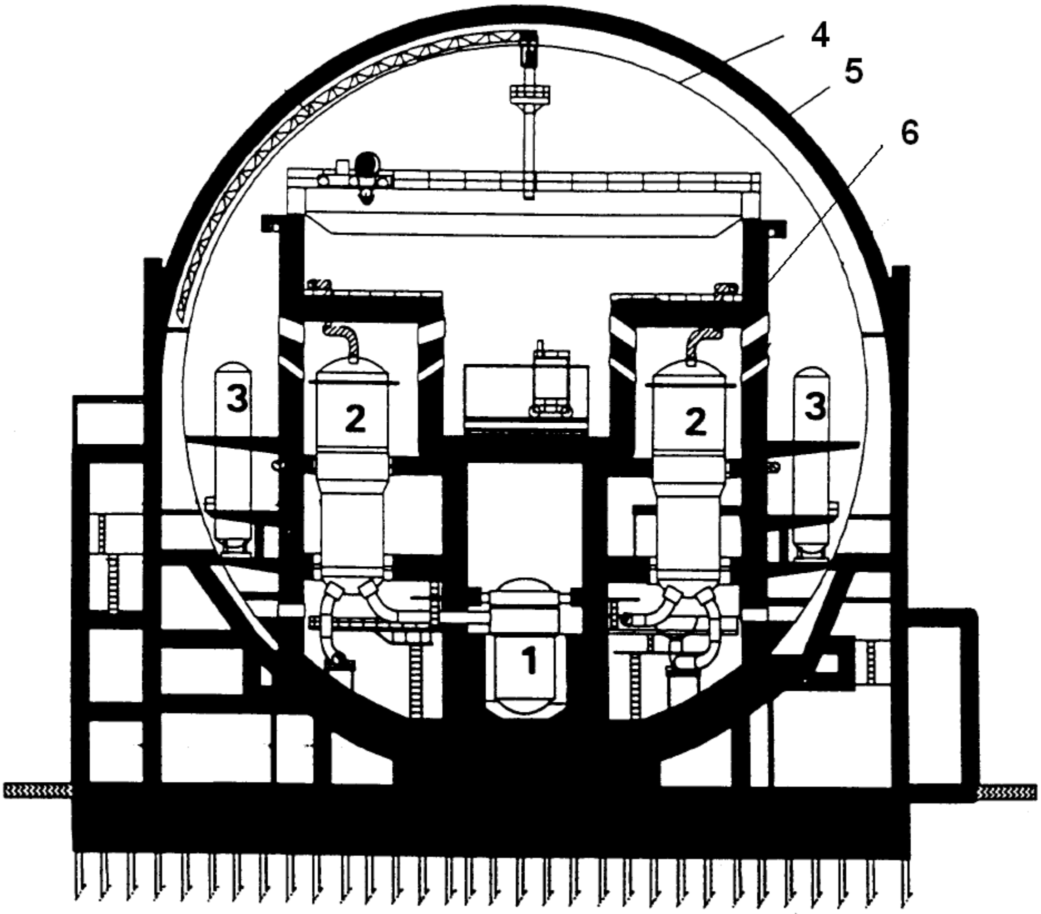

Figure 1-4 Cross section of KCB containment showing a small selection of major components
1. Reactor vessel
2. Steam generators (2 pieces)
3. Medium-pressure core inundation buffer tanks (4 pieces)
4. Primary containment: steel shell
5. Secondary concrete containment, the 'reactor building'
6. Cylindrical concrete housing of primary system

Control room

The main control room[14] is based on an ergonomically optimisation of plant operation procedures, including emergency procedures. A redundant bunkered control room is available

[14] The main control room was back-fitted during the second 10 yearly safety evaluation (Modification Project, 1997)

for controlled shutdown, core cooling and spent fuel pool cooling after occurrence of external hazards and in beyond-design conditions.

The reactor protection system was replaced in 1997 and is based on the principle for design base accidents no operator action is required in the first 30 minutes after the start of the event. 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 plant-specific full-scope control room simulator in Essen (Germany) is used for operator training with the full range of operational events.

More on safety relevant SSCs

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

There are two grids for the emergency AC power system (EY), for different levels of plant accident conditions. Emergency Grid 1 has 300% capacity (3 diesel generators) and the bunkered Emergency Grid 2 has two extra, smaller diesel generators (2 x 100%) in separate rooms. Likewise, other essential safety systems have been backed up in the bunkers. The 4-pump Safety injection system & residual heat removal system (TJ) is backed up by a 2-train bunkered Backup coolant makeup system (TW), and the 3-pump Main and auxiliary feed water system (RL) is backed up by a 2-train bunkered Backup feed water system (RS) system.

For conditions that result in the failure of all trains of the Conventional emergency cooling water system (VF) the plant is equiped with a redundant Backup cooling water system (VE). This ultimate heat sink can remove decay heat from the reactor core and the spent fuel storage pool, and provides cooling water to the emergency diesels. Its cooling water is groundwater, pumped from eight wells on the premises of the plant. The system is operated from the emergency control room.

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 main characteristics of the KCB have been summarized in Table 1-1.

Table 1-1 Main characteristics of safety relevant systems of the KCB

System(s)	Features / characteristics
Reactivity Control Systems	28 control rods; 3 volume control pumps (3 x 4.4 kg/s); 2 boron injection pumps (2 x 2.2 kg/s @ 4.7 bar, 21,000 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 @ 2,300 ppm).
Primary pressure protection system	3 tandem pressurizer relief valves with PORV[15] 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	Reactor coolant system: 2 trains of RHR[16] system with 2 pumps (2 x 167 kg/s @ 6.7 bar) each, seawater cooled (using the component cooling water system as interface); Separate heat removal system with 2 redundant pumps (2 x 61.1 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; 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 (4 x 28 m3, 25 bar) each; 2 trains of 2 storage tanks (178 m3 each) each
Steam Generator Heat Removal Systems	3 main feed water pumps (3 x 380 kg/s @ 66 bar, 3 x 50%); 3 auxiliary feed water pumps (3 x 24.4 kg/s @ 100 bar, 3 x 100%) one of them turbine driven; 2 trains of 1 backup feed water pumps (2 x 18 kg/s @ 80 bar, 2 x 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 92.2 bar each; Two 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 x 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[17]).
Key Safety Support Systems	Self testing reactor protection system; Emergency control room; Fire protection systems: Inergen[18], CO_2, fine water spray and Sprinkler systems, and crash tender.

[15] Power-Operated Relief Valve, PORV
[16] Residual Heat Removal, RHR
[17] Passive Autocatalytic Recombiner, PAR, installed and used to mitigate the impact of a possible hydrogen combustion and to avoid containment failure.

System(s)	Features / characteristics
Diesel generators	Two grids for emergency AC power system, for different levels of plant accident conditions. Emergency Grid 1: two air-cooled 6 kV diesel generators (2 x 100%, 2 x 4.343 MW) and one separated water-cooled diesel generator (1 x 100%, 1 x 4.343 MW). Bunkered Emergency Grid 2: 2 separately bunkered water-cooled 380 Volt, 0.88 MW diesel generator (2 x 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). Mobile diesel generator EY080: 400 V, 1 MW diesel generator, back-up for Emergency Grid 2. Batteries for the no-break power supply; capacity for at least 2 hrs.

Plant layout

In successive sections of this National Report, reference is made to various building numbers of the KCB. Figure 1-5 gives the plan of the Borssele NPP with identification of the most important buildings and their building numbers.

[18] Inergen, trade name of fire suppression system agent, consisting of the gases nitrogen, argon and carbon dioxide. It is used to lower the concentration of oxygen to a point that it cannot support combustion, but it is still safe for humans.

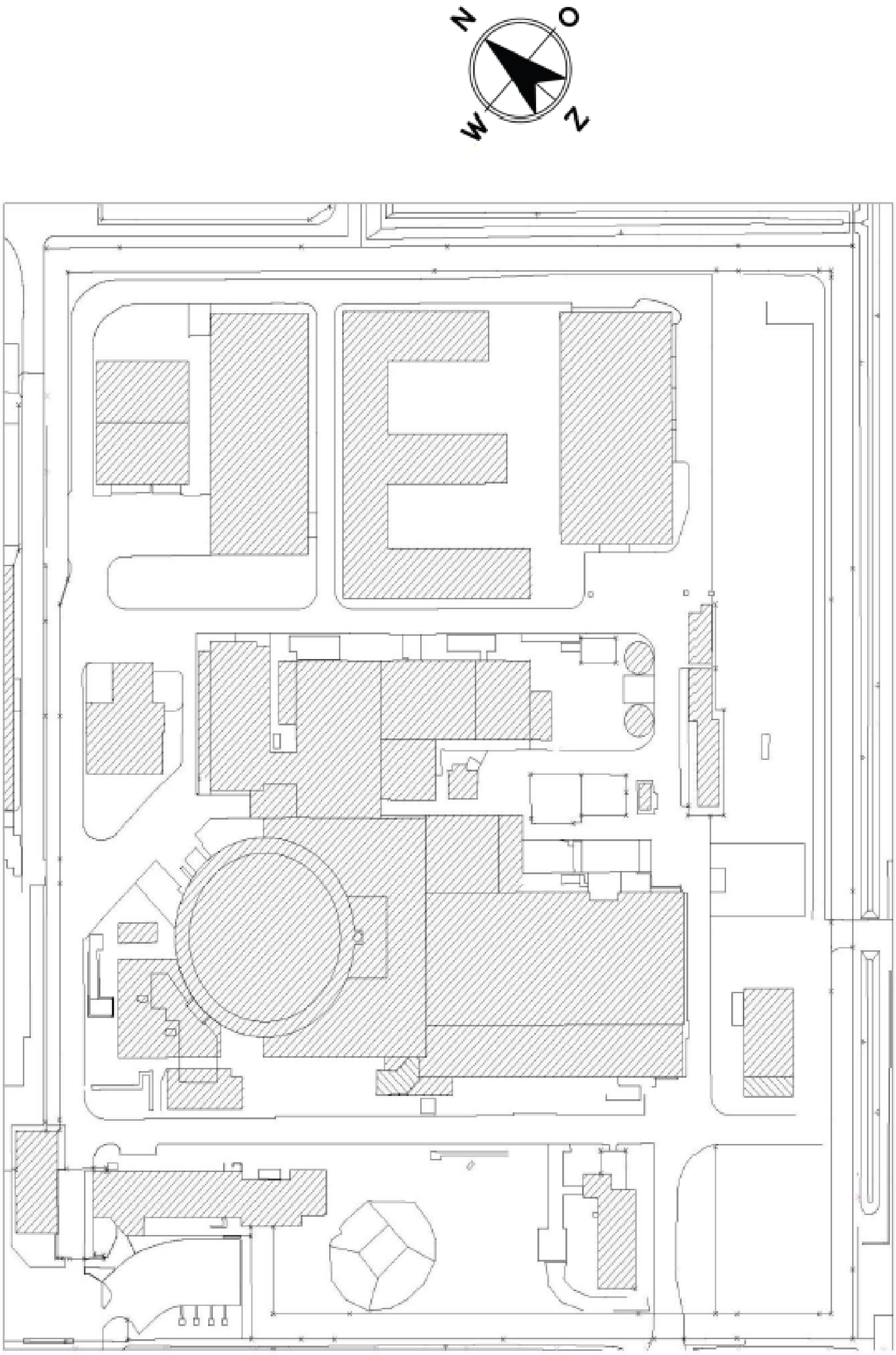

Figure 1-5 Plan of the Borssele NPP and identification of some important buildings[19]

01	Containment	12	Auxiliary transformer
02	Reactor building inside (annulus)	13	Ventilation stack
03	Nuclear auxiliary building	21	Cooling water inlet building (not on map)
04	Turbine building	23	Cooling water outlet building (not on map)
05	Electrical building /switchgear building	33	Backup systems bunker
06	Access building	34	Radioactive waste storage building
09	Condensate plant	41	Remote shutdown building / auxiliary transformer
10	Diesel generator building	48	Fire station
11	Generator transformer	72	Diesel generator building

[19] Numbers intentionally left out of map.

1.1.2 Description of the systems providing the main safety functions

This section of the National Report gives a factual description of the systems that provide the main safety functions. It largely matches the factual information provided in section 1.3.1 of the Licensee Report. The Licensee Report can be found on:

http://www.kerncentrale.nl/resultatenrobuustheidsonderzoek/EN/

A very detailed description of the systems providing the main safety functions can be found in the Licensee Report in its section 1.3. The Licensee Report in its subsections 1.3.2 through 1.3.7 provides even more detailed factual descriptions.

For the building numbers used in the successive sections, refer to Figure 1-5.

1.1.2.1 Emergency Grid 1 (EY010/020/030) and bunkered Emergency Grid 2 (EY040/050)

There are two grids for emergency AC power system (EY), which are used for different levels of plant accident conditions. Emergency Grid 1 (NS 1) has 300% capacity with 3 diesel generators, (EY010, EY020 and EY030) and the bunkered Emergency Grid 2 (NS 2) has 200% capacity with 2 smaller diesels (EY040 and EY050). For more specifications, refer to Table 1-1.

If the external grid fails, all the systems required for safe shut-down of the plant are powered by NS 1 and NS 2. One diesel generator from NS 1 or NS 2 is sufficient to power the safety systems during any design-basis accident.

Emergency Grid 1

If the external grid becomes unavailable, automatic switch over to house load operation occurs. In case of failure, power supply is switched to diesel generators of NS 1. NS 1 supplies power to the plant to shut down safely. NS 1 consists of two air-cooled 6 kV diesel generators EY010/020 (2 x 100%, 2 x 4.343 MW) and one separated water-cooled diesel generator EY030 (1 x 100%, 1 x 4.343 MW). These diesel generators supply AC power to NS 1.

NS 1 consists of two bus bars, BU and BV. Each bus bar has its own diesel generator, EY010 and EY020 respectively, and the backup diesel generator EY030 is available to feed either BU or BV.

In case EY010 or EY020 is not available due to maintenance, or in case either one fails to startup, EY030 takes over. The 6 kV rails BU and BV feed the 380 V rails CU and CV via two 2,000 kVA transformers. The 380 V bus bars CL and CM are fed via two separate 800 kVA transformers, Diesel generators EY010/020/030 are started automatically in case the external grid fails.

Each diesel generator (EY010, EY020 and EY030) is fuelled from a dedicated tank. The normal level in these tanks is sufficient for 72 hours of continuous operation.

Emergency Grid 2

If, besides the external grid, NS 1 would also fail, e.g. due to external hazards, NS 2 is available to supply power. NS 2 consists of two separately bunkered water-cooled 380 Volt, 0.88 MW diesel generators EY040/050 (2 x 100%). These diesels supply AC power to NS 2, which is designed to provide essential safety functions when specific accident conditions occur (essentially, for the bunkered feed water and primary injection systems).

NS 2 features two independent bus bars. Both of these bus bars can be fed from the external 10 kV grid, from the bus bars BU/BV of NS 1 or from its own diesel generators, EY040 and EY050 respectively. In addition, each bus bar is equipped with an auxiliary input for connecting the *mobile emergency power unit EY080*. NS 2 is housed in building 33 and thereby protected against external hazards. All the systems that are required to remain functional after external hazards, are powered by NS 2.

Each diesel generator (EY040 and EY050) can draw fuel for its operation either from a dedicated tank or from a main tank, which is located on the roof. The combined amount of fuel in these tanks is sufficient for 72 hours of continuous operation. The main tank on the roof can be refilled from an *external source*, e.g. from a *mobile* diesel storage tank.

1.1.2.2 Main steam system (RA)

The Main steam system (RA) is the connection between the two steam generators and the turbine. In the steam generators, the energy from the RCS is transferred to the secondary system (the steam / water cycle). By having a strict physical separation between the RCS and the secondary systems, the steam in the main steam system is clean (non-radioactive). Through the two main steam lines, the steam generators feed the turbo generator and it is there that the steam energy is converted into electricity.

The main steam pipes are welded to the steam generators (located in building 01). From there the pipelines run through building 02 into building 03. The main steam system is protected against overpressure by spring-operated safety relief valves (ten for each main steam line), which open at a defined pressure level, and atmospheric dump valves (two per main steam line), which are designed to control steam pressure and to cool down the installation automatically. The atmospheric dump valves (also called motor-operated relief valves) can be opened to reduce pressure down to atmospheric conditions by release to the environment.

Downstream the safety relief valves and the atmospheric dump valves, the main steam isolation valves are located with which the turbine and condensers can be isolated. Further downstream, the main steam isolation valves, the two main steam lines extend into building (04) via the roof of building (03). Here the two main steam lines enter the high-pressure stage of the turbine. The turbine can be bypassed completely (100%) using the bypass line which branches off from the steam lines.

1.1.2.3 Main and auxiliary feed water system (RL)

The task of the Main feed water system is to supply feed water to the steam generators, in order to enable heat transfer from the reactor coolant system to the secondary system. The task of the Auxiliary feed water system is to supply feed water to the steam generators in case the Main feed water system is not available. The Auxiliary feed water system is also used to provide the feed water at the required flow rates during normal plant start-up and shutdown by using the motor-driven emergency feed water pumps.

Main feed water system (RLM)

The RLM system consists of a feed water tank, three motor-driven centrifugal pumps, two feed water preheaters, two condensate coolers, a flow control station, and associated piping, valves and instrumentation. The feed water tank has a minimum capacity of 250 m^3. The water supply in the feed water tank is sufficient for approximately 6 minutes during full power operation in case no water is supplied to the tank. The feed water tank is equipped with two main safety valves which open if the pressure inside the tank becomes too high. The steam is vented through the roof until the pressure has dropped sufficiently. The possible exhaust flow rate corresponds to the maximum steam flow that can be fed to the feed water tank.

The feed water tank is maintained at a nominal pressure of 10.9 bar by steam normally supplied from the turbine. The three RLM pumps each consist of an assembly of a booster and a main pump, both driven by the same motor. Each pump processes water at a nominal rate of 1,361 t/h.

The discharge lines of the pumps are connected to the feed water header. The header splits and enters the feed water preheaters. The two feed water preheaters and the two condensate coolers heat the main feed water prior to injection through the feed water headers. Main feed water flow is controlled by motor-operated regulator valves on the main feed water lines. Each line has parallel low and high capacity regulator valves.

All of the RLM pumps are located in building 04 in the same room as the emergency feed water pumps. This room requires no room cooling to maintain acceptable operating temperatures. The feed water lines run from building 04 via the roof of building 03 into building 01 to the steam generators.

Emergency feed water system (RLE)

The RLE system consists of the feed water tank, one turbine-driven centrifugal pump, two motor-driven centrifugal pumps, a flow control station, and associated piping, valves and instrumentation. The RLE system shares the feed water tank with the main feed water system. The turbine-driven emergency feed water pump is powered by steam from the main steam line (RA), taken downstream of the main steam isolation valves (MSIVs). The turbine-driven pump and the motor-driven pumps have a flow rate of 120 t/h to the steam generators. RLE flow control is accomplished by motor-operated regulator valves on the emergency feed water lines. All three emergency feed water pumps and the main feed water pumps are located in building 04. No room cooling is required to maintain acceptable operating temperatures.

1.1.2.4 Main condensate system (RM)

The Main condensate system (RM) is designed to transfer the condensate from the condensers to the feed water tank. The RM system also provides cooling water to the generator during turbine-generator operation. In case there is a shortage, condensate can be injected using the Demin water supply system (RZ) into the RM system.

The Main condensate system RM consists of three motor-driven pump sets, 12 condensate heaters, and associated piping, valves, and instrumentation. The RM pump sets process water at a rate of 975 t/h at 22 bar and a rate of 1,260 t/h at 15 bar. All RM components are located in building 04.

1.1.2.5 Backup feed water system (RS)

The purpose of the Backup feed water system (RS) is to provide a backup source of make-up water to the steam generators in the event of the unavailability of the Main and Auxiliary feed water system (RL). The RS water also provides cooling to the diesel generator room and diesel generators (EY040 and EY050) until the Backup cooling water system (VE) is started. The RS system is designed to perform its task after the occurrence of an external event.

The RS system consists of two primary pumps which draw water from two storage tanks. These two pumps serve the RS injection trains which provide makeup via the main and emergency feed water lines to the steam generators. A manual cross-tie on the RS discharge lines provides the possibility for each RS train to inject in both steam generators. Each pump has a capacity of 50 m^3/hour at 120 bar. The storage tanks have a capacity of 496 m^3 and 469 m^3 respectively.

Two sets of two RS pumps recirculate the water from the RS tank and provide coolant for units of the UW system, which cools the diesel generator rooms, and for the coolers of diesel generator EY040 and EY050 themselves.

In order to extend the available RS operating time beyond 24 hours, the RS system can be refilled from water reserves of the UJ or RZ system. In addition, the RS system can be connected to an external water supply in case of an emergency (coupling by flexible hoses).

Most RS system components are located in building 33. The remaining RS components are located in building 01.

1.1.2.6 Demin water supply system (RZ)

The Demin water supply system (RZ) is an integral part of the demineralized water storage and transfer systems. The RZ system has three major functions:

1. to provide a transfer capability between the Demineralised water plant (UA) and the plant systems that require high-quality, steam-generator-grade water during normal plant operation;
2. to provide storage for demineralised water that is to be used as condensate in the event of a disruption of the normal condensate supply to the feed water systems, or in the event of a break in the feed water tank or a feed water line between the tank and check valves of the main feed water pumps. During operations, a minimum of 300 m^3 of demineralised water must be in the RZ storage basin. The RZ system also has access to an inventory of 407 m^3 in each of the demineralised water storage tanks;
3. to provide the means to transfer the stored condensate to the Main and auxiliary feed water system (RL). Each of the large capacity RZ pumps has the ability to deliver a maximum of 88 t/h at 16.5 bar to either the feed water tank or the suction lines of the emergency feed water pumps.

The RZ system is also used to refill the RS tanks through a normally closed manual valve. The RS tank draining and refilling is performed regularly.

The RZ condensate transfer system consists of three motor-driven centrifugal pumps. These pumps draw suction from the RZ storage basin and process fluid at a normal rate of 60 t/h at 19.5 bar. During normal plant operations, at least 2 of these pumps are required to be operable. The RZ system has a fourth pump. This pump is rated at a lower capacity than the other three and is conservatively not considered to be part of the condensate transfer function of the RZ system due to its lower capacity. The RZ storage basin is actually a combination of four separate compartments connected by common headers. Most RZ components are located in building 04.

1.1.2.7 Volume control system (TA)

The Volume control system (TA) is a three pump system providing charging and letdown flow to and from the RCS during normal plant operations. The TA system also provides water to the pressuriser sprays (as an alternate spray source), seal water to the reactor coolant pumps, and chemistry control for the RCS using chemicals from the Chemical control system (TB).

The TA system consists of three positive displacement pumps, each of which has a capacity of 16 m3/h. These pumps draw reactor coolant from the volume control tank, occasionally supplemented by the TB system, and discharge it into the Reactor Coolant System (RCS) cold legs. Primary coolant is then drawn from the RCS to the volume control tank via the recuperative and high-pressure heat exchangers in the letdown part of the TA system. The volume control tank has a capacity of 14.3 m3, and can be supplemented by highly concentrated borated water or demineralised water from the TB system.

The TA injection pumps are located in building 02. The recuperative heat exchangers and the high-pressure heat exchangers are located in building 01.

1.1.2.8 Chemical control system or Boron injection system (TB)

The Chemical control system, also called the Boron injection system (TB), delivers borated and demineralised water to the suction of the Volume control system (TA) pumps in order to compensate for gradual changes in reactivity, and also to fill the Reactor Coolant System (RCS), the Spent Fuel Pool (SFP) and several water storage tanks.

The borated portion of the Chemical control system consists of two centrifugal motor-driven pumps which provide borated water to the suction piping of the TA system pumps. Each boron injection pump has an injection rate of 4 m^3/hr through control valves. The boron injection pumps draw highly borated water (12%) from the boron holding tanks. Each of these tanks has a capacity of 16 m^3 and a normal operating volume of 12 m^3 of highly borated water. Via a boron-water mixing unit the TB system injects borated water in the requested concentration in the TA system. If manual control is selected, then both pumps can be run simultaneously.

Redundancy is provided on the suction and discharge side of the pumps by cross-ties. Therefore, a flow from both tanks is possible through either of the boron injection trains to each TA pump.

The demineralised water portion of the Chemical control system consists of two centrifugal motor-driven pumps, which automatically supply demineralised water to the boron-water mixing unit.

The boron injection pumps, the demineralised water pumps and the boron storage tanks are located in building 03.

1.1.2.9 Backup residual heat removal system (te) and backup cooling water system (VE)

The Backup residual heat removal system (TE) and Backup cooling water system (VE) are single train systems providing backup cooling for residual heat removal if the low-pressure operation mode of the Residual Heat Removal (TJR-RHR) system fails. The TE and VE systems are installed and designed for mitigation of:

- unavailability of TJR-RHR due to external events (earthquake, aircraft accident, explosions or high tides);
- unavailability of the Conventional emergency cooling water system VF.

The backup Ultimate Heat Sink (UHS) consists of the following three subsystems:

- Backup residual heat removal system (TE);
- Spent fuel pool cooling system (TG080);
- Backup cooling water system (VE).

The TE system is a backup for the Residual heat removal system (TJ). The TG080 part of the Spent fuel pool cooling system has an additional heat exchanger cooled by VE.

VE provides cooling water for the TE and TG080 systems. Furthermore the VE system provides cooling water to the RS system diesel generators (EY040 and EY050), the diesel generator rooms and the electronic cabinet rooms in building 33. The coolers of the spent fuel pool, the TE system coolers and the TE pump trains are placed in building 02.

The TE system is connected to the suction header of one TJ low-pressure train. The TE pump discharges via a series of manual and check valves to the cold legs of both RCS loops. During power operations the TE and VE systems are in standby mode and the system has to be manually aligned before being placed into service.

The VE backup cooling water pumps are submersible pumps in the ground water bore holes. The VE system consists of eight submersible pumps all delivering flow to a common header. The VE system discharges via the main cooling water system VC to the Westerschelde river.

1.1.2.10 Component cooling water system (TF)

The Component cooling water system (TF) is designed to transfer, in the different operation modes, the heat from the different coolers to the Conventional emergency cooling water system (VF). TF is a closed cooling system separated in two trains, each with two pumps. Three TF heat exchangers are available. Via a set of valves, the third one can be connected to one of the trains dependent on the heat loads. TF supplies cooling to numerous safety related and non-safety-related components.

1.1.2.11 Spent fuel pool cooling system (TG) including TG080

The Spent fuel pool cooling system (TG) is a closed loop, three pump system (TG020, TG025 and TG030) which circulates fuel basin water through a series of heat exchangers where heat is transferred to the component cooling system TF (section 1.1.2.10). The main purpose of the TG system is to remove the decay heat from the fuel rods stored in the spent fuel pool (SFP).

During the normal operation of decay heat removal from the SFP, water is drawn from the SFP through five overflow weirs and is returned to the SFP through a discharge nozzle. A part of the water flow of the TG system is filtered through a cation and anion resin system. Each of the three TG pumps can supply either set of two TG heat exchangers. The TG030 pump has a seal cooling by TF. The TG020 and TG 025 pumps operate independently of the TF and VF systems as they are cooled by the TG flow itself.

If there is a loss of the normal Ultimate Heat Sink (UHS), either TG020 or TG025 can be switched to the TG080 cooler, using the VE cooling system (well water cooling system, refer to section 1.1.2.9).

1.1.2.12 Safety injection system & residual heat removal system (TJ)

The TJ system fulfills a variety of safety and non-safety related functions during various plant conditions, including:

- high-pressure injection;
- low-pressure injection;
- low-pressure recirculation: residual heat removal and containment sump recirculation
- accumulator injection;
- containment spray.

High pressure injection system (TJH)

The High pressure injection system (TJH) is designed to supply a source of make-up water to the RCS. TJH is used in case of a loss of coolant accident (LOCA), or as a source of RCS makeup following a steam generator tube rupture, feed water or steamline break, or in conjunction with the power-operated relief valves for feed and bleed operation.

The TJH consists of four multistage centrifugal pumps which draw water from four inundation tanks and supply both RCS loops 1 and 2 through two separate and redundant injection trains.Each train consists of two redundant pumps, which inject into the hot and cold legs of one of the RCS loops. Cross-ties on the suction and discharge headers of the two trains can be manually opened in case of loss of one injection train (accident management measure). The TJH pumps are located in building 02 and the inundation tanks are located in building 03. The TJH injection trains run from the TJH pump outlets to the hot and cold legs of RCS loops 1 and 2.

Low pressure injection system (TJL)

The Low pressure injection system (TJL) is designed to supply make-up water at low pressure to the RCS. TJL is used following a large LOCA which rapidly depressurises the RCS and following an intermediate break LOCA which depressurises the RCS more slowly. TJL components also provide the long-term make-up and reactor core decay heat removal functions. These long-term functions are described in the low-pressure recirculation (TJR) section below. TJL consists of 4 single stage centrifugal TJL pumps which draw water from four inundation tanks and supply both RCS loops 1 and 2 through a series of check valves. The inundation tanks are common for the TJH and TJL systems and have each a useable volume of 143 m^3. The TJL pumps are arranged in two main injection trains, each injecting to the hot and cold legs of a RCS loop. Redundancy in the TJL system is provided by separating the TJL system into two completely separate and redundant injection trains.

Cross-ties on the suction and discharge headers of the two trains can be manually opened in case of loss of one injection train (accident management measure).

TJH and TJL make use of the same inundation tanks and injection lines. If the inundation tanks are empty, the TJH pumps are stopped and TJL switches over to recirculation mode by suction from the containment sump (TJR).

The TJL pumps are located in building 02. The TJL injection trains 1 and 2 run from the TJL pump outlets to the cold and hot legs of the RCS loops. The inundation tanks are located in building 03.

Low-pressure recirculation system (TJR)

The low-pressure recirculation mode of operation (TJR) of the Low-pressure injection system (TJL) provides two functions:

1. Provide a means for normal decay heat removal from the reactor vessel during normal plant cooldown: the residual heat removal mode (TJR-RHR)
2. Provide a source of make-up water and core cooling at low pressure from the containment sump after LOCA.

The Low-pressure recirculation system (TJR) uses the TJL system and pumps. For plant cooldown and decay heat removal, each train takes suction from a RCS hot loop. Primary water is cooled in the TJ residual heat exchangers and returned to the RCS cold loops. Residual heat removal mode is possible from RCS at 30 bars to mid-loop (RCS water level at middle of the primary loops). Each suction line from the hot loop has two motorized isolation valves, one electric and one hydraulic. Redundancy in the TJR system is provided by separation of the TJL system in two separate trains. Each train can remove 100% of the residual heat.

The recirculation from the containment sump can be used after LOCA and ensures the low pressure injection mode (TJL) after depletion of the inundation tanks. On low level of the inundation tanks the suction is automatiquely switched to the containment sump. Filters in the sump prevent blocking of the system. The water from the containment sump is cooled in the TJ heat exchangers before it is reinjected into the primary loops. The suction lines have two electric operated valves. Redundancy is provided by separation in two separate trains.

Accumulator injection system (TJB)

The Low-pressure accumulator injection system (TJB) consists of four, 28 m^3 tanks containing borated water. The water is injected through a series of check valves to both the cold and hot legs of the RCS's coolant loops 1 and 2. The tanks are pressurised to 24.5 bar with nitrogen gas.

Upon a drop in pressure below 24.5 bar in the RCS, the check valves will open, supplying the hot and cold legs of both RSC loops 1 and 2 with borated water. This system is designed as a totally passive safety system.

There are four main injection paths for the TJB system. Each of the four tanks independently supplies either the hot or cold legs of the RCS loops.

The TJB tanks are located in building 01.

Containment Spray system

For the design basis accidents containment spray is not necessary. The system can be used as accident management to decrease pressure, temperature and aerosols by spraying in the dome of the reactor building. The system consists of two redundants trains. Each train is equipped with a pump which sucks from two inundation tanks.

1.1.2.13 Nuclear ventilation system (TL)

The Nuclear ventilation system (TL) consists of 10 subsystems, which are dedicated to controlling the air conditions in the Reactor building (01/02) and building 03. The specific tasks of the TL system include:

- the maintaining of a focused air flow to prevent spreading radioactive materials through
- the air and to prevent an uncontrolled release to the environment;
- reducing the amount of radioactivity in the filtered air;
- the intercepting radioactive materials in the air by filtering it before it is discharged into
- the ventilation shaft;

- establishing and maintaining specific atmospheric conditions;
- disposing heat produced by the parts of the installation and lighting;
- monitoring the RCS by measuring the amount of condensed water generated in the circulation coolers.

1.1.2.14 Backup coolant makeup system (TW)

The Backup coolant makeup system (TW) is designed to:

- compensate with borated water for leakages after external hazards;
- compensate with borated water for primary inventory shrinkage and reactivity increase
- due to RCS cooldown;
- decrease RCS pressure and boron injection by spraying in the pressuriser in case of loss
- of TA/TB systems or in case of steam generator tube rupture;
- inject borated water in conditions with high RCS pressure (ATWS);
- inject borated water in open reactor vessel conditions.

The TW system consists of two redundant pump trains, which draw borated water from two storage tanks. The two positive displacement pumps, one for each train, have a capacity of 18.8 m^3/h at a pressure of 185 bar. The storage tanks have a net capacity of 243 m^3 and 262 m^3 respectively.

Most TW components are located in building 33. The remaining components are located in building 02 and 01.

1.1.2.15 Demineralised water plant (UA)

The Demineralized water plant (UA) is responsible for desalination and purification of the industrial water supply so that it can be used to fill the installations of the plant. This function must be performed during normal plant operation. The UA system has no safety function. The system consists of two identical filter stages including cation and anion filters, CO_2 degassers, booster pumps, an active carbon filter and two demin-water storage tanks. The demineralised water is stored in the storage tanks can be used as backup for the Demin water supply system (RZ).

1.1.2.16 Low pressure fire extinguishing system (UJ)

The buildings where production takes place at the Borssele NPP are provided with Low- (UJ) and High- (UF) pressure fire-water systems. In normal situations the UJ system has a constant (static) pressure of 4 bar. The fire-water pump (electrically driven) has a capacity of 6,000 l/min with a pressure about 10 bar. The electrical driven pump has a backup pump which is a diesel-driven pump with the same capacity and pressure as the electrical one. The UJ system also provides water to the automatic sprinkler and fine-water spray systems.

1.1.2.17 Conventional emergency cooling water system (VF)

The Conventional emergency cooling water system (VF) is designed to transfer the heat from several nuclear and non nuclear systems via an intermediate system (TF, VG) to the Westerschelde. VF supplies sea water to the following heat loads:

- Nuclear intermediate heat exchangers;
- Diesel generator coolers;
- Conventional intermediate heat exchangers;
- Chilled water system coolers;
- De-aeration system cooler.

The VF system consists of two independent sub-systems: train 1 and train 2. Each train has two pumps (rated capacity 584 kg/s) with one of them normally operating. The VF pumps draw sea-water from the Westerschelde in the cooling water intake building (21), through five independent suction trains. The discharge from the VF pumps flows via two main headers to the different heat exchangers. The VF water is then returned to the Westerschelde via the Main cooling water system (VC).

The VF system pumps are located in building 21.

1.2 Significant differences between units

The Netherlands has only one operating NPP which features only one nuclear reactor on its site.

1.3 Use of PSA as part of the safety assessment

In the Netherlands, PSA is an important tool to evaluate whether a nuclear installation meets the Dutch risk criteria. In fact these risk criteria apply to all industrial activities, including non-nuclear ones. There are two main risk criteria that have to be met:

1. The maximum allowable individual risk (mortality) as a consequence of operating an installation is 10^{-6} per year. It is calculated for one of the most vulnerable groups of people, the one year old children.
2. The societal risk has been defined as the risk of 10 or more fatalities, directly attributable to (an accident with) an operating installation. For an accident with 10 fatalities, the maximum permissible risk is 10^{-5} per year. With every tenfold increase of the number of fatalities, the maximum permissible risk should be divided by 100.

KCB has implemented a full scope 'living' Probabilistic Safety Assessment (PSA), ranging from PSA Level-1 to PSA Level-3. 'Living' means that the PSA for the Borssele NPP is updated yearly. This means that both plant modifications and updated failure data are included in the PSA model. The operator EPZ is using the Living PSA for many applications.

Scope of Level-1 PSA

The scope of KCB's PSA-1 includes power and non-power operation. It evaluates the core damage frequency and plant damage state frequencies. It is used to identify the main weak points in the plant's safety features.

The Level-1 analysis includes internal as well as external initiating events. The level of detail is enough to allow the demonstration of the effectiveness of hardware modifications. The PSA takes account of the operating experience. This enables to evaluate the influence of the maintenance policy on the

Scope of Level-2 PSA

The Level-2 analysis of the KCB is used to identify the likelihood and mechanisms for potential releases for radioactive materials from the containment. The total core damage frequency obtained from the Level-1 analysis forms the basis for the analysis.

Scope of Level-3 PSA

Using the source terms that form the output of the Level-2 analyses, the COSYMA computer program has been used to evaluate the compliance with the Dutch risk criteria.

2. Earthquakes

The Netherlands is a region with low seismic activity and this is certainly true for the area where the KCB is located, the province of Zeeland. This section evaluates the compliance of the KCB with its design basis regarding earthquakes and the available seismic safety margins.

The protection against potential earthquakes is adequate. The Licensee has proposed to undertake studies to reduce the uncertainty in the seismic margins by additional studies like a Seismic Margin Assessment (SMA) or a Seismic-Probabilistic Safety Assessment (Seismic-PSA). It is anticipated that such studies will benefit from recent advances in the techniques of seismic safety assessment.

2.1 Design basis

2.1.1 Earthquake against which the plant is designed

No Design Basis Earthquake was specified in the original design due to the low seismicity of the site. The DBE was established later and updated several times during various Periodic Safety Reviews (PSRs).

2.1.1.1 Characteristics of the Design Basis Earthquake (DBE)

The definition of the DBE for the KCB area is conservatively based on an intensity of VI½ MMI (Modified Mercalli Intensity). The largest historically observed intensity is V½. The medium return period associated with this intensity corresponds to around 30,000 years, thus the median frequency of exceeding the intensity of the DBE is around 3.10^{-5} per year.

The ENSREG guidance requires to express the DBE in terms of the maximum horizontal peak ground acceleration, the PGA. The Licensee Report provides two values: 0.6 $m.s^{-2}$ for the ground level and 0.75 $m.s^{-2}$ for the level at the base of the foundations. The PGA has been derived from the intensity, hence its return period equals that of the intensity of the DBE.

2.1.1.2 Methodology used to evaluate the DBE

The highest earthquake intensity ever recorded in the area of the KCB was approximately V½ MMI, caused by the earthquake with a magnitude of 5.6 on the Richter scale near Tournai, Belgium on June 11, 1938. This is based on recorded earthquakes in the period 217 and 1990 AD.

For the DBE this intensity level was increased with one unit, arriving at VI½ MMI. Ground response spectra for two ground conditions have been directly derived from the intensity of the DBE. These spectra have also been modified to account for the effects of larger earthquakes at greater distances. This does not lead to changes in the PGA values, but extends the plateau region of the spectra.

The resulting PGA values are:

- 0.6 m/s^2 for alluvial ground conditions (on the left in Figure 2-1), to be used if the input ground motion is applied at ground level;
- 0.75 m/s^2 for medium ground conditions (on the right), to be used if the input ground motion is applied at the basis of the pile foundation.

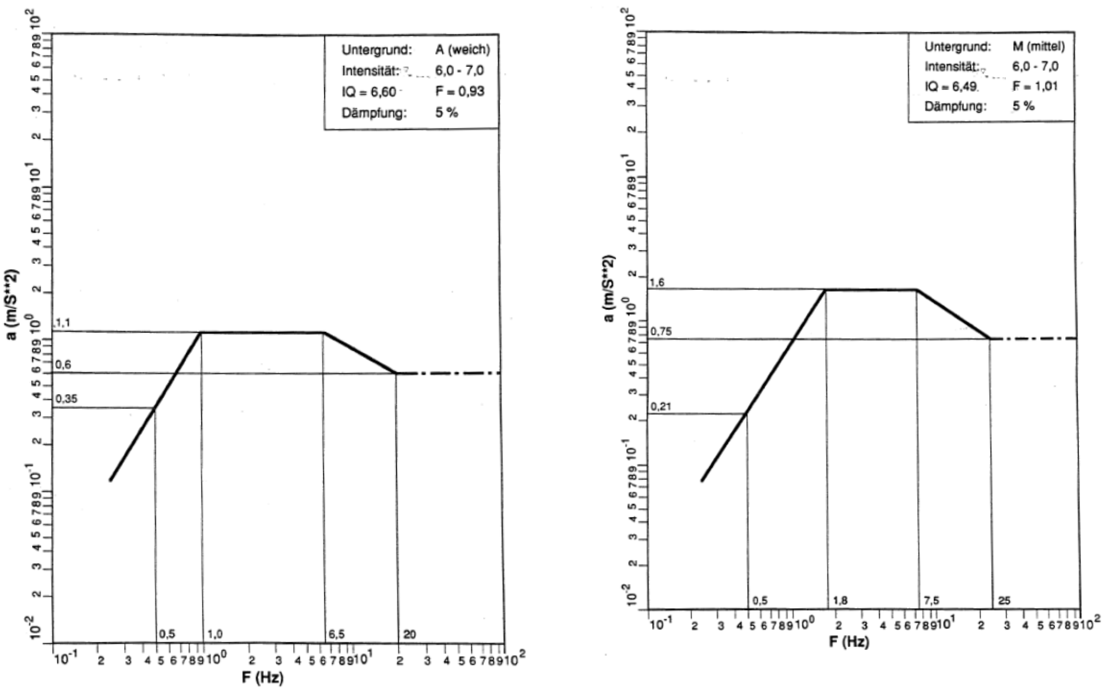

Figure 2-1 Modified 'Hosser' spectra used in the second 10-yearly Periodic Safety Review.

2.1.1.3 Conclusion on the adequacy of the DBE

The licensees judgement of the DBE makes strong reference to the German KTA standards, like KTA 2201.1[20] and KTA 2201.2[21]. The application of the German practice seems justified since the seismically 'calm' conditions near Borssele are generally comparable with those for – especially Northern – Germany.

2.1.2 Provisions to protect the plant against the DBE

The Licensee Report presents six levels of defence against the DBE and to achieve a safe shut down state. They are summarized in short:

1. Prevent the challenge to the safety systems due to earthquakes, this is achieved in two ways:
   - The proper site selection; the Borssele site is characterized by a very low seismicity;
   - Engineered design of all Structures, Systems and Components (SSCs) supporting normal and emergency plant operation.
2. Deterministic protection concept relying on seismically qualified safeguards only. A conservative assumption in this concept is that SSCs not designed for DBE loads will fail. SSCs have been designed for DBE loads or an assessment and retrofit programme has made certain that essential SSCs will perform successfully under seismic conditions;
3. Resistance of seismic class 1 and 2a SSCs against seismic loads exceeding the design basis. Significant margins have been established due to conservative assumptions applied at the different stages of the seismic design.
4. Preventive emergency measures implemented to cope with design-exceeding situations induced by a loss of vital safety functions. Ensure decay heat removal in scenarios induced by design exceeding earthquakes, like loss of AC power and loss of ultimate heat sink.

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[20] Design of Nuclear Power Plants against Seismic Events; Part 1: Principles, KTA-Geschaeftsstelle c/o Bundesamt fuer Strahlenschutz (BfS), Germany, 1990

[21] Design of Nuclear Power Plants Against Seismic Events, Part 2: Subsurface Materials (Soil and Rock) , KTA-Geschaeftsstelle c/o Bundesamt fuer Strahlenschutz (BfS), Germany, 2000

5. Mitigative emergency measures foreseen to maintain the containment barrier after the onset of core melt. There is an extensive set of severe accident management guidelines (SAMGs). They describe the use of for instance passive autocatalytic recombiners (PARs) eliminating the risk of hydrogen explosion and a containment filtered venting system[22]. More on this topic can be found in chapter 6.
6. Emergency preparedness programme ensuring proper on- and offsite emergency responses under accident conditions to protect employees, public and environment against the effects of radioactive releases from the plant.

The licensee has addressed these six levels of defence in the context of:

- Identification of SSCs that are required for achieving a safe shutdown state and are most endangered during an earthquake, including evaluation of their robustness in connection with the DBE;
- Main operating contingencies in case of damage that could be caused by an earthquake and which could threaten achieving a safe shutdown;
- Protection against indirect effects of an earthquake;

Depending on their functional role with respect to the safety functions, the SSCs are assigned to one of the following safety classes: Class 1, Class 2a and 2. The completeness of the SSCs assigned to class 1 and class 2a as well as their seismic design has been verified in the framework of the second 10 yearly safety evaluation.

All SSCs have been identified and the seismic design has been verified. This does also extend to the spent fuel pool, including the spent fuel racks.

The seismic class 2a has been exhaustively verified in the framework of the second 10 yearly safety evaluation. The integrity of these SSCs and their assignment to seismic class 2a after an earthquake was demonstrated. Further defence is established by physical separation of redundant safeguards wherever possible.

The indirect effects of an earthquake were considered, and the licensee reported that no consequences for coping with the DBE could be identified. Building structures and components that have no seismic design are not needed for controlling safety functions after an earthquake event.

A total loss of AC power is excluded from the DBE but could also be mitigated by the preventive emergency measures of 'secondary bleed and feed' and 'primary bleed and feed'. A total loss of ultimate heat sink, i.e. a simultaneous loss of the Conventional emergency cooling water system (VF) and the Backup cooling water system (VE), is excluded from the design-basis earthquake. However it could be mitigated for a longer period by the Back-up feedwater system RS, provided that demineralised water make-up to the storage tanks is available in the longer term.

If offsite power is lost, the available emergency power supplies are challenged. Those are designed to withstand the DBE. In some cases, an additional mobile diesel generator or other electrical power sources are provided.

Additionally a description for the spent fuel pool (SFP) exists. With regard to spent fuel pool cooling, a loss of the Conventional emergency cooling water can be durably mitigated with the seismically qualified cooling chain TG080/VE. Also a total loss of ultimate heat sink would not induce a cliff edge since heat removal from the spent fuel is ensured by thermal inertia of the water in the spent fuel pool for at least six hours (in shutdown modes with the reactor core fully unloaded) or one day (in power operation mode). By injecting cold water into the pool and the

[22] The filtered containment venting system ensures that containment pressure can be reduced to and maintained at acceptable level without uncontrolled release of radioactive materials.

excess water spilling over into the containment sump, this grace time can be significantly stretched. A corresponding emergency procedure is presently under elaboration.

The on- and offsite infrastructure needs generally not to be assumed as destroyed after an earthquake which does not exceed the design basis of MSK intensity VI½. This is also valid when regarding the accessibility of the plant site via the local road (Europaweg). Due to the low intensity of the DBE, the assumption is that the infrastructure can continue to be used after the earthquake. Hence no prevention or delay of the access of personnel and equipment is expected under DBE conditions.

Potential sources of consequential (internal) fire inside the safety-related buildings have been identified and excluded by the measures resulting from the second 10 yearly safety evaluation. The fire-fighting systems in buildings 01, 02 and 35 are not designed for operability under DBE, the licensee has listed this as a weakness and subsequently has identified possible modifications.

Soil liquefaction has been studied during the site investigation and can be ruled out for this location because of the very limited strong motion duration (< 5s) and the limited peak ground acceleration < 1 $m.s^{-2}$. In case of a longer duration of strong motion (> 9 s), however, liquefaction of certain ground layers cannot be ruled out.

2.1.3 Compliance of the plant with its current licensing basis

The regulatory position is that the plant complies with its current licensing basis. The position is based on decades of regulatory oversight.

Evidence of compliance is documented in many licensing documents that are subject to regulatory oversight like the Safety Report, the Technical Specifications, strategic maintenance and surveillance plans and (in more detail) to extensive maintenance and in-service inspection programmes.

The licensee notes that further improvement is possible. Corrective measures have been taken, or are planned to be executed by the licensee. Self-assessment has shown that the availability and preparedness of auxiliary mobile equipment further improvement is possible. Examples of such equipment are mobile generators, fire-fighting equipment, hoses etc.

2.2 Evaluation of the safety margins

2.2.1 Range of earthquake leading to severe fuel damage

Neither a seismic PSA nor an explicit Seismic Margin Assessment was performed in the past. In the Licensee Report, the concept of seismic margin is introduced first, followed by description of the way sources of seismic margin applicable to the KCB are derived. These are concluded by an estimation of these margins with respect to the fundamental safety functions: ability to achieve and maintain subcriticality, decay heat removal and confinement of radioactive substances. In addition the capacity of the buildings that support the safety functions has been estimated.

A seismic margin is generally understood to be the plant's capability to withstand seismic loads exceeding the design basis. If these margins are to be expressed quantitatively, it is necessary to describe the plant seismic capacity in terms of representative ground motion parameters. The use of the peak ground acceleration (PGA) as a characteristic ground motion parameter has found the widest application and is used in the Licensee Report.

The seismic capacity of the plant cannot be expressed easily as a discrete figure since there are various sources of variability (due to randomness and uncertainty). It is common practice to express the seismic capacity either as median seismic capacity A_m, i.e. the 50-percentile of the variability distribution, or as HCLPF (high confidence low probability of failure) capacity. The HCLPF thus represents the peak ground acceleration at which the probability of seismic induced failure level is low (< 5%) at high confidence (= 95%). HCLPF values can be elaborated for individual SSCs but also for safety functions and the entire plant. Often there are different success paths ensuring a safety function. The seismic capacity of a safety function is then determined using the MIN-MAX rule: the minimum (MIN) seismic capacity of the SSCs required in each success path is first derived, then the HCLPF capacity of the safety function is given by the maximum (MAX) capacity of each success path.

The Licensee Report explains that a global analysis of the entire path of the vibrational energy of the surrounding soil to a certain component is not practicable. Instead the overall system is split into various subsystems which have been analysed separately.

However this approach introduces several sources of conservatism like:

- Excitation characteristics and physical parameters are chosen conservatively to ensure that conclusions made will be valid with a high degree of certainty. By combining several models in a chain of calculation, one arrives at results with considerable conservatism;
- Bounding assumptions with respect to input characteristics. ASME and KTA require smoothing, broadening or even increasing of input spectra;
- Interaction, especially damping effects between subsystems are neglected, resulting in conservatism;
- Realistic anchorage characteristics that allow small displacements and significantly influence vibrations behaviour (avoiding resonant situations) are neglected;
- Non-linear effects like plastic deformation and friction are not modelled and corresponding energy dissipation is neglected or only considered with simplified approaches.

In a simplified assessment, licensee has shown that there are significant seismic margins with respect to the fundamental safety functions. The lowest HCLPF capacity of all considered SSCs has been estimated to be 0.15 g. The HCLPF capacity for many safety-relevant systems and bunkered buildings is higher.

2.2.2 Range of earthquake leading to loss of containment integrity

The capacity with regard of confinement failure is judged to be dominated by sliding of the steel containment against the concrete structures. This would cause significant damage to the concrete internal structures or shear failure of the outer shield. The EPRI NP-6041 screening value of 0.3 g could be taken as a reasonable estimate in absence of a detailed fragility analysis.

Taking into account that an increase of the intensity level by one unit equals a doubling of the peak ground acceleration, the following can be concluded:

- earthquakes up to an intensity of VII-VIII (VII½), i.e. exceeding the design basis by one unit of intensity level, will not lead to core damage or even confinement failure under high confidence;
- there is also a high probability that the plant can withstand earthquakes up to an intensity of VIII-IX (VIII½) – the licensee report refers to real earthquake experiences from fossil power plants in the Pacific region which support this statement.

The PGA that would result in loss of integrity of the reactor containment was not estimated, but no confinement failure is expected with high confidence even then the DBE is one unit of intensity higher.

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

A beyond design earthquake is not expected to lead to external flooding of the KCB site. No consequential links have been discovered between the possible occurrence of an extreme flood (like tsunami) and an initiating earthquake. A second mechanism for flooding is failure of dikes induced by earthquakes. The chances of damages to 'soil structures' like dikes due to an earthquake is estimated to be small in this region (< 10^{-4} per year). Liquefaction of the soil has been investigated. The distance of the Westerschelde embankment and the KCB buildings is large enough to prevent liquefaction that could influence stability of KCB buildings and / or flood defences.

Earthquakes and extremely high tides are considered to be largely unrelated phenomena at the Borssele site. However, adequate measures are in place to cope with a combination of these phenomena. With a beyond design earthquake of 0.3 g combined with flooding of KCB premises, buildings essential for safe shutdown will remain available; they are designed against both earthquakes and external flooding.

2.2.4 Measures which can be envisaged to increase robustness of the plants against earthquakes

2.2.4.1 Cliff-edge effects

Some potential cliff-edge effects have been identified by the licensee and modifications and /or investigations proposed. The cliff-edge effects listed are:

Unavailability of shift personnel after 10 hours

There is a potential for a cliff-edge with design-exceeding earthquakes if the main control room is destroyed combined with a situation in which the site becomes inaccessible. This would lead to a high pressure core melt scenario after the Back-up feedwater system (RS) storage tanks were drained.

Structural failure of missile shield inside containment at PGAs > 0.3 g

Such a scenario may induce a core melt while containment integrity is not ensured.

Possible failure of the containment filtered venting system.

The filtered venting system is not qualified for the design-basis earthquake.

Possible inoperability of the fire-fighting systems in buildings 01, 02 and 35.

Unlike the fire-fighting system in building 33, the fire-fighting systems in buildings 01, 02 and 35 in principle are not qualified for the design-basis earthquake.

2.2.4.2 Possible modifications / investigations:

Measures proposed by licensee EPZ

The related modifications / investigations that according to the licensee could be envisaged are:

- Emergency Response Centre facilities that could give shelter to the alarm response organisation after all foreseeable hazards would enlarge the possibilities of the alarm response organisation;
- storage facilities for portable equipment, tools and materials needed by the alarm response organisation that are accessible after all foreseeable hazards would enlarge the possibilities of the alarm response organisation;
- ensuring the availability of fire annunciation and fixed fire suppression systems in vital areas after seismic events would improve fire fighting capabilities and accident management measures that require transport of water for cooling/suppression;
- by increasing the autarky-time beyond 10 h the robustness of the plant in a general sense would be increased;
- ensuring the availability of the containment venting system TL003 after seismic events would increase the margin in case of seismic events;

- uncertainty of the seismic margins can be reduced by a Seismic Margin Assessment (SMA) or a Seismic-Probabilistic Safety Assessment (Seismic-PSA). In 10EVA13 either a seismic-PSA will be developed and/or an SMA will be conducted and the measures will be investigated to further increase the safety margins in case of earthquake;
- in 10EVA13 the possibilities to strengthen the off-site power supply will be investigated. This could implicitly increase the margins in case of LOOP as it would decrease the dependency on the SBO generators;
- develop a set of Extensive Damage Management Guides (EDMG) and implement a training programme;
- develop check-lists for plant walk-downs and needed actions after various levels of the foreseeable hazards.

Regulatory position statements

- The information provided by EPZ on the vulnerability of the NPP for earthquakes is plausible;
- The Licensee Report states that the licensee plans to perform a seismic PSA or a Seismic Margin Assessment, a SMA. The regulatory body endorses this proposal. It is known that the Netherlands Royal Meteorological Institute (KNMI) will contribute data and knowledge to this project. In the seismic study attention should be given to among others characterization of the subsurface of the site and possible seismic impact of foreseeable future mining activities in the neighbourhood.

3. Flooding

In the Netherlands, flooding is an external hazard that is quite relevant to assess when siting any (industrial) activity. There is considerable national expertise on estimating flooding risks and engineering of structures that are needed to protect the land from flooding.

The Water Act ('Waterwet') replaces former acts on water management, like the Flood Defences Act and Public Works Act, both of which were important for the implementation of the governmental policy on flood risk.

In the Netherlands, the design of dykes is based on a defined exceeding frequency. A norm for the exceeding frequency of one in 4,000 years means that the water defence must be able to withstand all combinations of water levels and waves that have an occurrence probability of one in 4,000 years. The norm varies for the various areas in the Netherlands and depends on among others aspects like population density and scale of industrial activity in the respective areas.

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)

Originally the NPP Borssele was designed to withstand a flood level of 5 meters above NAP[23]. Currently, the height of the Design Basis Flood (DBF) is set at 7.3 meters above NAP. The DBF includes the dynamic wave height. Upon occurrence of a flood equal to the DBF, overtopping of the dykes that protect the NPP is assumed.

3.1.1.2 Methodology used to evaluate the DBF

The original flood level of 5 meter + NAP is probably based on the highest known water level at Borssele that was reached during a storm surge of 1 February 1953, which amounted to 4.7 meters + NAP. The level of 5 meter nowadays is referred to as the 'Laag Hoogwater Concept'(low high-water concept). Within this concept all systems essential for operating the plant and all installed (safety) systems for safe shutdown stay available up to at least the level of 5 meters + NAP.

In various periods reassessments have been made, resulting in modified high-water concepts. Currently the flood protection level (DBF) is 7.3 meter + NAP due to several back-fitting measures carried out in various periods.

The Nuclear Base Level (in Dutch: 'Nucleair BasisPeil', or N.B.P.) is calculated for Borssele at 6.18 m + NAP and according to the Licensee Report appears with a return period of 1 million year[24] (frequency of 10^{-6} per year). For flood-resistant design of Borssele NPP, the design level is obtained by adding various factors to the N.B.P., as defined in the regulations of the IAEA. The resulting level (N.B.P. + factors) is the calculated nuclear design level (calculated N.O.P., 'Nucleair Ontwerp Peil'). The N.O.P. features two variants, the Static N.O.P. (6.29 m + NAP) and the dynamic N.O.P. (7.29 m + NAP). In effect, the dynamic N.O.P. equals the DBF.

[23] The 'sea level' is defined by the height of the calm sea surface in relation to a horizontal standard level. The Normal Amsterdam Water Level (NAP in Dutch) is used as this standard level.

[24] Refer to section 3.1.1.3 for comments on this statement.

3.1.1.3 Conclusion on the adequacy of protection against external flooding

With regard to systems, structures and components (SSCs), it can be concluded that the current design basis of NPP Borssele regarding external flooding is adequate.

In order to ensure that NPP Borssele will continue to be able to withstand possible flooding in the future, design levels are evaluated every ten years during the 10 yearly safety evaluations. Based on the outcome of these evaluations, modification projects are initiated if necessary. A surveillance programme is put in place to ensure these design levels. Currently the N.O.P. is under review within the fourth 10 yearly periodic safety review (PSR). This PSR-report in 2012 will contain a new statistical analysis of the return periods of extreme high water levels (including the DBF) based on a recent dataset.

The regulatory body has the opinion, that the impact of floods with a long return period needs further assessment. Refer to section 3.2.2 for the full regulatory position.

3.1.2 Provision to protect the plant against the DBF

3.1.2.1 Identification of SSCs that are required for achieving and maintaining safe shutdown state and are most endangered when flood is increasing

The licensee has identified all SSCs required for achieving and maintaining a safe shutdown state within the defence concept against flooding.

3.1.2.2 Main design and construction provisions to prevent flood impact to the plants

The site is enclosed by three dykes. Sea dykes A (sea dyke, 9.4 m + NAP) and B (inland dyke, 7.75 m + NAP) are part of the dyke ring protecting a part of Zuid-Beveland (peninsula, part of province of Zeeland) and are the main defence against flooding of the site. The difference in crest height between both dykes bears no relation to their reliability. Both dykes are designed against the same allowable failure frequency, as required by law ('de Waterwet'), namely once in 4,000 years. The different crest heights result from differences in orientation, foreshore, obstacles, etc. The Waterwet also ensures that with regular inspections and a five-yearly review of the design, the condition of the dykes is kept up to date. As a result of this, dyke A will be improved in 2012. The failure frequency of the dyke will remain once in 4,000 years. The regulatory body has the opinion, that the impact of floods with a long return period needs further assessment. Refer to section 3.2.2 for the full regulatory position.

3.1.2.3 Main operating provisions to prevent flood impact to the plant

In case of a threatening flooding, procedure S-VF-01 is initiated. The procedure is initiated by a water level of 3.05 m + NAP or a storm warning which is issued by the province of Zeeland at 3.10 m + NAP. In the unlikely case that a dyke failure can be expected, actions are foreseen. To control a situation in which all systems have failed, Severe Accident Management Guidelines (SAMGs) are available.

The procedure includes:

- continuous monitoring of the cooling water inlet building;
- bringing Site Emergency Director (SED) on site;
- consultation of the management by the SED;
- communication with the CCB shift supervisor and possible request for assistance.

By taking these measures the threat of a possible flooding is closely monitored and anticipated. The next steps that are required in case an actual flooding would follow are determined during this process and are adapted to the pending situation.

In the unlikely case that a dyke failure can be expected, the following actions must be initiated:

- Cooling down to 'cold shutdown' mode. All cooling options remain available in case of LOOP, plus most efficient use of cooling water.
- Mobilisation of additional personnel (operators en maintenance crew).

Whereas the infrastructure will still be intact, two additional shifts will be called on site to occupy both the Emergency control room and the (backup) Emergency Response Centre, for the sake of emergency preparedness.

The existing alarm procedure should be extended with a straightforward decision model for the above mentioned points and additional necessary actions. For example, the logistical implications of bringing in staff for an undefined period must be further elaborated. Since increase of water levels can be forecasted and is a relatively gradual process, sufficient time should be available to carry out the required actions to achieve the desired conditions.

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

Even though the installation is safeguarded, accessibility of the plant becomes increasingly difficult, even at a water level of several centimetres. As a precaution, additional personnel (operators and maintenance crew) is mobilized at the site of KCB. Before the infrastructure is affected, these additional shifts will be called on site to occupy both the emergency control room and emergency coordination locations, for the sake of emergency preparedness. Next to mobilization, communication will be a major problem in case the region is flooded.

Whereas the weather must be so extreme in order to generate dyke failure and flooding, also Loss Of Off-site Power (LOOP) must be anticipated. Failure of electricity pylons due to extreme weather conditions is credible, but will not result in the loss of safety systems required for safe shutdown of the plant (refer to Chapter 5).

3.1.3 Plant compliance with its current licensing basis

3.1.3.1 Licensee's processes to ensure that plants 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 operable condition.

In order to ensure that NPP Borssele will continue to be able to withstand possible flooding in the future, design levels are evaluated every ten years during the 10 yearly safety evaluations. Based on the outcome of these evaluations, modification projects are initiated if necessary. A surveillance programme is put in place to ensure these design levels.

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.

An extension to the precautions is proposed. This measure will include revision of procedures for use and surveillance, next to determining the specifications (and number) of the mobile equipment. It also includes adaptations to the mobile diesel generator EY080, which is currently located at the switchyard and not available at flood levels higher than 5 m + NAP.

It is not clear if mobile equipment is required to cope with flooding. No licensee's processes are described to ensure that mobile equipment and supplies will be available under flood conditions. However licensee has indicated that the availability of mobile equipment will be assessed in more detail.

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

No direct deviations from the current licensing basis have been found. The evaluation of the N.O.P., which will be reported in 2012, is discussed in the Licensee Report and it is concluded that a moderate change is possible. Depending on the exact outcome of this evaluation appropriate measures shall be taken. The descriptions provided are comprehensive and adequate enough.

3.2 Evaluation of safety margins

3.2.1 Estimation of safety margin against flooding

The regulatory position is that the information provided by licensee is plausible, given the model employed. Below the margins resulting from employing that model are reproduced. However the regulatory body has the opinion the impact of floods with a long return period are not known in much detail yet and that further assessments may be needed. Refer to section 3.2.2 for the complete regulatory position.

In the low high-water concept (5 m + NAP) of licensee, the weakest link is the cooling water inlet building which is designed against a static water level of 5 m + NAP, but which is water tight to 7.4 m + NAP. However, a possible margin could exist, even when taking wave and run-up effects into account.

If the water level reaches the 6.7 m + NAP floor of building 04, 05 and 10, the electrical power supply from Emergency Grid 1 will be heavily affected. However, most of the 6 kV / 0.4 kV transformers, including the transformer feeding bus bar CU of Emergency Grid 1 are located in building 05 at the 6.7 m + NAP floor. The air intakes of the cooling of these transformers (via natural convection) are openings in the wall of building 05 at 5 m+ NAP. This means that these transformers are subject to the dynamic water level as is present outside the buildings. This does not apply to the transformer feeding bus bar CV which is fed by bus bar BV; all these components are located in building 10 and are thus not subject to a dynamic water level. As a consequence, this part of Emergency Grid 1 is available up to a static level of 6.7 m+ NAP. The availability of the main control room is not guaranteed. But its functionality is to be expected because of the availability of (part of) Emergency Grid 1, rectifiers, batteries and the dispatcher.

If the water level reaches the 7.3 m + NAP the flooding is covered by the RS-concept. The availability of the main control room is not guaranteed. However, its functionality is to be expected because of the availability of (part of) Emergency Grid 1, rectifiers, batteries and the dispatcher. (Emergency) communication to outside parties must be assumed to be lost as no specific protection of the external communication lines against wide-spread flooding is foreseen.

A margin of 1 m exists above the DBF of 7.3 m + NAP, before the situation worsens considerably and prevention of core damage becomes difficult.

3.2.2 Measures which can be envisaged to increase robustness of the plants against flooding

Measures proposed by licensee

The licensee envisages additional studies on the return-periods of extreme floods and the results of these have to be considered when deciding on possible measures to increase robustness. In the current 10 yearly safety evaluation the DBF is under review. Depending on the exact outcome of this evaluation, in 10EVA13 measures will be investigated to further increase the safety margins in case of flooding.

As discussed dyke A will be improved in 2012. Regarding structural measures, wave protection beneath the entrances to the bunkered backup injection- and feed water systems and to the bunkered emergency control room would mitigate the sensitivity to large waves combined with extreme high water and would make the plant less dependent from the dike.

To improve plant robustness during actual flooding situations, the following measures are proposed by the licensee:

- develop a set of Extensive Damage Management Guides (EDMG) and implement a training program. Examples of Issues to be addressed are:
    - procedures to staff the Emergency Control Room;

  - use of autonomous mobile pumps;
  - procedure to transport own personnel to the site;
  - procedure for the employment of personnel for long term staffing;
- an Emergency Response Centre facility that could give shelter to the alarm response organisation after flooding (and all foreseeable hazards) would increase the options of the alarm response organisation;
- storage facilities for portable equipment, tools and materials needed by the alarm response organisation that are accessible after flooding (and all foreseeable hazards) would increase the options of the alarm response organisation;
- establishing independent voice and data communication under adverse conditions, both on-site and off-site, would strengthen the emergency response organisation;
- improvement of plant autonomy during and after an external flooding, for example by establishing the ability to transfer diesel fuel from storage tanks of inactive diesels towards active diesel generators would increase the margin in case of loss of off-site power.

Regulatory position statements

- Before implementation of any measure proposed by licensee, careful consideration has to be given to its effectiveness.
- The regulatory body has the opinion that the impact of floods with long return periods (ten thousand, one hundred thousand, one million years etc) is not known in much detail yet and that further assessments may be necessary. Several governmental bodies are involved in the assessment of the adequacy of the protection of the Netherlands against flood risks. Models have been developed and continuously are being improved to aid this assessment. It is required that a reassessment tailored to the needs of the Borssele site be undertaken considering: (1) the maximum challenge floods with long return periods will pose to the NPP and its dykes, (2) the various failure mechanisms of the dykes, (3) the impact of the maximum challenge by floods on the safety of the NPP, and (4) the various options to protect the plant against this challenge like improving dykes and/or adding other engineered structures.

4. Extreme weather conditions

4.1 Design basis

The licensee in its Licensee Report has considered the following weather conditions:

- maximum and minimum water temperatures of the River Westerschelde;
- extremely high and low air temperatures;
- extremely high wind (including storm and tornado);
- wind missiles and hail;
- formation of ice;
- heavy rainfall;
- heavy snowfall;
- lightning;
- credible combinations of the conditions mentioned above.

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 plants systems, structures and components: maximum temperature, minimum temperature, various types of storms, heavy rainfall, high winds, etc.

Water temperature

A minimum allowable water temperature is not specified as a design basis. A maximum daily-average cooling water inlet temperature of 21.6 °C from the Westerschelde estuary is the original design basis for the maximum water temperature for NPP Borssele. However, analyses have shown that sufficient cooling is guaranteed up to a Westerschelde water inlet temperature (daily-average value) of at least 25 °C. The lowest temperature was recorded in January 1997: −1.1 °C; the highest temperature, 23.2 °C was recorded in July 2006 at Vlissingen.

Air temperature

The minimum or maximum allowable outside air temperature is not specified in the design basis. However, sufficient cooling capacity is available to maintain allowable temperatures inside operational areas, under outside temperature conditions that can be expected in the Netherlands. Design provisions guarantee the availability of the emergency diesels and the diesel fuel inventory at very low temperatures. To prevent coolant of generators freezing, additives are used.

Pipes of the Low pressure fire extinguishing system (UJ) are located at a depth of approximately 0.8 meters below ground level. As frost in the soil may occur up to a depth of 0.7 m, freezing of the UJ piping does not seem not very likely, but further assessment may be useful.

Wind, wind missiles, hail and salt deposits

The design load of buildings is higher than the design wind load. The resulting thrust pressure at a wind speed of 450 $km.h^{-1}$ is below the maximum expected static pressure in the event of an explosion. It is demonstrated that the maximum expected wind speed is sufficiently covered by the design explosion pressure wave.

Wind missiles and hail is covered by the resistance against a small aeroplane crash since the design-basis aeroplane crash.

Although not likely, failure of electricity pylons due to extreme storms is credible. Failure of the pylon on the KCB site would lead to a Loss Off Offsite Power (LOOP), but will not result in loss of safety systems required for the safe shutdown of the plant. Refer to chapter 5 for more information on LOOP.

Dry winds can carry salt which can deposit on among others electrical circuits. Measures against these deposits exist.

Icing

The formation of drifting ice on the Westerschelde is a relatively slow process that can be adequately anticipated.

To prevent icing of the cooling water intake screens (VA[25]), the coarse and fine intake screens in building 21 can be equipped with hot air guns during the winter. In case of complete loss of the Main cooling water system VC and the Conventional emergency cooling water system VF due to complete icing, the Backup cooling water system VE can take over the ultimate heat sink function. VE pipes are located at a depth of approximately 1.5 meter below ground level. Because frost in the soil may occur up to a depth of 0.7 m and because the water is brackish, freezing of the VE piping can be ruled out.

Rain and snow

Extreme rainfall will not cause loss of relevant SSCs, under the assumption that extreme rainfall will not last longer than 48 hours in case drainage pipes are blocked and water accumulates on the roofs. For some roofs, analysis of accumulation of water was not needed, because of their shapes (dome, saddle shape or not roof edges).

Situations are possible, that water accumulates on the roofs as a consequence of fire-fighting activities if drain pipes are blocked. It should be evaluated if in such situations the allowable load will not be exceeded.

All the building structures of KCB are designed so that they can withstand all the credible consequences of snowfall.

Lightning

The lightning protection of the containment (building 01/02) and building 05 fulfils protection class I of KTA 2206. The protection of buildings 33 and 35 and the associated connections to the unprotected zones (cable conducts) has been designed according KTA 2206.

If the plant is subjected to lightning pulses with amplitudes above the designed levels, damage or spurious actuation of I&C safety channels can be initiated. In a worst-case scenario, a Loss Of Coolant Accident (LOCA) could be assumed as a result. However these actions can be overruled by emergency operation procedures (manual operation or switch gear).

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

Apart from the weather conditions discussed, there are no additional weather conditions that may have an impact on the reliable operation of the safety systems, which are essential for heat transfer from the reactor and the spent fuel to the ultimate heat sink.

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

A more thorough analysis is ongoing in the current 10 yearly safety evaluation. A first examination shows that no major changes are expected in the return frequency of the discussed phenomena. In general, the degree of resistance against external influences that is required is defined so that the probability of an accident with serious consequences caused by external

[25] VA: Cooling water filtering system

weather influences is small compared to the risk of serious accidents by causes within the plant, i.e. a probability of less than 10^{-6} per event per year.

4.1.1.4 Consideration of potential combinations of weather conditions

Credible combinations of extreme weather conditions have been considered and no significant deficiencies have been identified:

- High air temperature + high water temperature;
- Low air temperature + low water temperature;
- Snow + extreme wind;
- Extreme wind + extreme rainfall + lightning.

4.1.1.5 Conclusion on the adequacy of protection against extreme weather conditions

It can be concluded that there are no flaws in the protection, although there is some room for improvement. These possible improvements are discussed in the evaluation of the safety margins in section 4.2.

4.2 Evaluation of safety margins

4.2.1 Estimation of safety margin against extreme weather conditions

No cliff edge effects will occur as a result of the margins, although for lightning no quantifiable margins are given. Below the various extreme weather conditions are addressed.

Water temperature

The intake temperature of water from the River Westerschelde at which sufficient cooling is fully guaranteed is 25 °C. The reactor will be shut down at a daily average seawater temperature of 25 °C, meeting the requirements specified in the Technical Specifications. In the unlikely case that this is not done and the seawater temperature rises further, the reactor will be shut down automatically around 27 °C. After reactor shutdown the seawater temperature will be less limiting because of the decreasing decay heat. Directly after reactor shutdown the limiting seawater temperature is 30 °C.

Because warming up of seawater is a slow process and the above mentioned seawater temperature is unrealistically higher than the maximum measured peak seawater temperature of 23.2 °C, it can be concluded that there is sufficient margin for cooling at any credible seawater temperature. In all cases the reserve Ultimate Heat Sink (UHS) is available as backup. This reserve UHS consists of the Backup cooling water system (VE), the Backup residual heat removal system (TE) and part of the Spent fuel pool cooling system (TG080)), which is independent of seawater temperature.

Air temperature

Exceeding the allowed air temperatures in the containment (building 01) or the control room (in building 05) will have no direct safety impact but will lead to shutdown based on the requirements in the Technical Specifications with regard to conventional health safety requirements. This means that a sufficient margin exists although this is not quantifiable.

Wind

The safety margins are known for buildings 01/02 (containment), 33 (Backup systems bunker) and 35 (housing the emergency control room). They have to be able to resist a wind load of 0.1 bar, where their design enables them to take 0.36 down to 0.3 bar, resulting in a margin of 0.26 down to 0.2 bar.

For building 03, 04, 05, 21 and 72 no margin is quantified, although it is noted that they are resistant to wind speeds of at least 12 Bft[26]. Extremely high wind speeds may cause fragments of wall cladding to detach in very rare cases, but no cliff edge effects will take place.

Ice formation

No quantifiable margin can be given for the ice formation on the estuary Westerschelde. Ice formation is no acute problem because its formation in the Netherlands always can be anticipated and in principle, the Conventional emergency cooling water system (VF) will be available as a backup. If ice blocks or damages the cooling water intake, prolonged cooling is possible using the Backup cooling water system (VE), which pumps its water from ground water bore holes. In conclusion, no cliff edge effects are expected from ice formation.

Rainfall

In the unlikely event of a combination of continuous rainfall lasting longer that 48 hours and completely blocked drainpipes, the water level on top of building 03 (nuclear auxiliary building) may exceed the maximum allowable design load. However, loss of building 03 will not lead to loss of safety systems required for safe shutdown of the plant. No cliff edge effects are expected.

In addition, collapse of building 03 is highly unlikely because of the presence of multiple drainpipes, the statistics on rainfall presented by the Royal meteorological institute (KNMI), and procedures for monitoring plant safety and building integrity.

Snowfall

All relevant building roofs meet norm NEN 6702 which requires resistance to a load of 0.7 $kN.m^{-2}$. The existing safety margin in relation to this norm varies for the various buildings from 0.3 (turbine building 04) up to 16.7 $kN.m^{-2}$ (dome 01/02). Building 04 should be monitored first with heavy and prolonged snowfall. However operators perform daily checks in and around the plant to monitor plant safety and building integrity. Snow and ice will be removed when needed.

If the usual checks would not be performed during extreme and prolonged snowfall, and maximum load of the building roof of 04 would be exceeded, this roof might collapse. However the floor beneath this roof is strong enough to carry this extra load. Systems located below this floor – like some parts of Conventional emergency cooling water system (VF) – would remain unaffected by the collapse. So the VF will remain available after a collapse. Some feed water systems can be affected by the collapse like the Main and auxiliary feed water systems (RL-M and RL-E). However alternative and backup sources are available and the reactor can still be brought into cold undercritical state.

Lightning

The plant satisfies the NEN 1014 and KTA 2206 rules on lightning protection, but no quantifiable margin for lightning can be given.

4.2.2 Measures which can be envisaged to increase robustness of the plant against extreme weather conditions

Measures proposed by licensee EPZ

- Develop check lists for plant walk-downs and needed actions for various levels of foreseeable hazards.

Regulatory position statements

- The regulatory body endorses licensees proposal, however evaluation of its effectiveness is needed before implementation;

[26] 12 Bft (Beaufort): hurricane force, featuring wind speeds larger than 117 $km.h^{-1}$, averaged over 10 minutes.

- Heavy rain does not pose extreme challenges to the plant. A special case is the accumulation of water resulting from fire-fighting activities if drain pipes are blocked. The possible consequences of this need to be studied;
- Further recommended topics for additional study are: the minimum depth of underground piping required for proper protection against freezing, possibility to operate diesel generators at extremely low temperatures and the potential effect of accumulation of wind-transported snow on roofs.

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

The emphasis of this chapter is on consecutive measures that could be attempted to provide necessary power supply and decay heat removal from the reactor and from the spent fuel. It is focussed on prevention of severe damage to the reactor and the spent fuel, including any last resort means. In addition, the time available for these measures is estimated.

For mitigating actions to be taken in case of occurrence of damage to reactor and / or spent fuel, refer to chapter 6 'Severe accident management'.

For more detailed information on technical provisions, refer to chapter 5 of the Licensee Report.

5.1 Loss of electrical power

For the 'stress test' ENSREG postulates that all offsite power is lost and remains so for 72 hours. In addition it is postulated that the site remains isolated from delivery of heavy material for the same period, whereas light equipment may reach the site after 24 hours.

The Licensee Report follows the ENSREG-prescribed format for Licensee Reports and features two separate sections 5.1 and 5.2 for 'Nuclear power reactor' and for 'Spent fuel storage pool' respectively. The ENSREG-prescribed format for *National Reports* is different, in that findings for these separate areas need to be merged in the National Report.

EPZ has defined the following combinations for the LOOP-SBO issues (plant states):

- LOOP, characterized by:
    - unavailability of supply from external grids and of the 6 kV connection between the NPP and the coal fired plant (the CCB);
- LOOP and station black out (referred to as SBO 1), characterized by:
    - unavailability of supply from external grids and of the 6 kV connection between the NPP and the coal fired plant (CCB);
    - unavailability of the emergency grid 1 (NS1);
- LOOP and total loss of AC-power (referred to as SBO 2), characterized by:
    - unavailability of supply from external grids and of the 6 kV connection between the NPP and the coal fired plant (CCB);
    - unavailability of the emergency grid 1 (NS1);
    - unavailability of the emergency grid 2 (NS2).

For all these plant states it is initially assumed the batteries (DC) and uninterrupted AC (380V) power system are available.

5.1.1 Loss of off-site power (LOOP)

This section covers the measures to cope with the plant state 'Loss Of Offsite Power' (LOOP) for the nuclear power reactor as well as for the spent fuel pool (SFP).

5.1.1.1 Design provisions to take account of LOOP

LOOP can be initiated by loss of grid supply or loss of connections (to external grid and CCB). There are several provisions to prevent LOOP or better reduce its probability, and there are provisions to mitigate its consequences.

Design provisions to prevent plant state LOOP

The electrical energy of the turbine generator is transformed to 150 kV and transported to the (150 kV) external grid. For 'house' use there is the house load transformer which feeds the 6 kV house grid. The house grid has two redundant parts. For start-up and shut-down there are two auxiliary transformers that take power from the external grid.

There are various redundancies in grid connection. E.g. a house transformer failure can be coped with by two auxiliary transformers. The house grid is connected with two separate lines to the coal fired plant (CCB) which shares[27] its site with the NPP.

The plant can operate in house-load modus, producing power for only its own needs. The reactor will operate at 30% of its power, the turbine generator at 5%. This mode is preferred above shut-down when the grid will be restored within a short time frame. With extended loss of grid, shut-down may be chosen for which emergency grid 1 (NS1) will be available to provide power.

Internal backup provisions to cope with LOOP

The *uninterrupted power system* (UPS) provides DC power to the reactor protection system (RPS) and other SSCs that in the event of LOOP are necessary to achieve a safe shut-down state.

The NPP´s *two emergency grids* N1 and N2 provide AC power in emergency situations. They have been described in chapter 1 of this National Report and in much more detail in chapter 1 of the Licensee Report.

- NS1 features three emergency diesel generators (EDGs) (3x 100%) that will start automatically in case of LOOP - after two seconds. After that it takes 10 seconds to power up. The diesels can be operated manually locally and from the control room if needed.
- NS2 will be activated if NS1 is not available. It is housed in the backup systems bunkered building, and protected against external hazards like flooding, earthquakes and explosions. NS2 has two separated EDGs, each of which is capable to support all the systems that are needed to achieve a safe shutdown.
- Interconnection of NS1 and NS2: if NS2 receives no power from its own EDGs, NS1 can supply power to the NS2 emergency grid.

The *coal fired plant* (CCB) is seen as an on-site AC provider, providing two options:

- If CCB still has off-site power, then via a 6 kV transmission line, power can be fed to the NPP.
- If CCB does not have off-site power, its emergency diesels can provide enough power to feed NS2 (via NS1).

The *on-site mobile emergency diesel generator* (EY080) is available if NS1 and NS2 are lost. EY080 has enough power (1 MW) to feed NS2. However, for physical separation it is located at a distance from the other emergency diesels. To move it, external support is needed. Therefore it is not a completely internal backup option. If a truck is considered to be light equipment, in the postulated scenario of ENSREG it will be available 24 hours after LOOP started. If the truck is considered to be heavy equipment, the first 24 hours no credits can be given to the mobile EDG.

5.1.1.2 Autonomy of on-site resources and provisions to prolong on-site AC service time

AC Power

There are various on-site resources for AC power. There are several emergency diesel generators, each having their own stocks and various shared reserve stocks. The maximum running time of a particular EDG can be extended by using available stocks from several tanks.

[27] The NPP and the coal fired plant are both owned by utility EPZ.

The minimum amount of available diesel in stock is 245 m^3. During an event, the challenge will be to get the fuel at the right place timely – this means internal refuelling. With this collected stock, NS1 can operate about 280 hours (using one of the emergency diesels, EY010 or EY020). If the events would be limited to LOOP only, external replenishment of the fuel stock would enable NS1 to supply power to the plant indefinitely. NS2 (with one emergency diesel, EY040 or EY050) with the on-site stock would have a running time of about 1300 hours (54 days) before the stock will be exhausted. However, there are no dedicated hardware provisions and procedures available to support the required fuel transfers, staff will have to improvise.

Below all available on-site AC resources are addressed separately.

The emergency diesel generators of NS1 start automatically after *two seconds*, when the voltage drops below 80% of the nominal voltage or the frequency deviates more than 5% of the nominal frequency. The NS1 features three emergency diesel generators (EDGs). Two of these have enough diesel stock to last for about 79 hours if operated at *full* power. The Technical Specifications Packages (TIP) states that they will function at least 72 hours. The third diesel of NS1 – separated from the others and serving as a backup – has enough diesel to last 25 hours.

The diesel generators of NS2 start automatically after two seconds, when the voltage drops below 80% of the nominal voltage or the frequency deviates more than 5% of the nominal frequency. The EDGs of NS2 have enough diesel stock to operate for 72 hours at maximum expected loads.

If NS2 needs power from NS1, it takes one hour to establish the connection from NS1 to NS2 to supply the users of emergency power grid 2.

The coal-fired unit CCB, if connected to the grid and not experiencing LOOP, can transmit off-site power to the NPP via 6 kV house grid, and there will be no limitation in time to supply. If the CCB experiences LOOP, its emergency diesels can provide power to the NS2 of the NPP. If there is a acute emergency at the NPP, power to NS2 from CCB's EDGs can be provided in 30 minutes. If there is more time allowed to make the connection, the CCB staff can take more time (four hours) to make the connection and meanwhile protect its own equipment. The runtime of the CCB's EDGs is about nine hours.

With the mobile EDG (EY080), once in position, it takes four hours to make the connection to NS2. Transporting it over the site may take another two hours. A truck is needed to transport it, which has to be provided by an external company.

Batteries

The autonomy periods for the systems to provide DC power in case of LOOP are presented below. As soon as AC power is lost, DC powered systems will be supplied by the batteries.

Table 5-1 Battery discharge times

Battery	Discharge time (h)
220 V (NS 1)	2.8 (5.6)
+ 24 V building 5 (NS 1)	2.3
- 24 V building 5 (NS 1)	2.6
+ 24 V building 33 (NS 2)	7.3
- 24 V building 33 (NS 2)	10.5

Following procedures described in the Licensee Report, the discharge time of the 220 V batteries can be increased to 5.7 hours. This comes at a cost, switching off the turbine oil pump will safe battery time but will damage the turbine.

Competence of shift staff to make the connections

EPZ's staff is competent and qualified to make the necessary connections. However the actions required in the postulated scenario's are not periodically trained. The licensee proposes to especially train the procedure for 'mid-loop operation'. Mid-loop operation is a short period during outage in which the water level in the primary system is lowered to 2/3 of the primary coolant lines. An SBO may have a large impact on the safety. Several procedures exist to perform the necessary actions to achieve a safe plant state. Periodic training of these will guarantee their successful execution when needed.

5.1.1.3 Robustness of provisions in connection with seism and flooding

All safety relevant SSCs are able to maintain functioning at PGAs up to at least 0.15 g. Damage due to seismic events will be limited. SSCs needed for coping with floods will still be available after seismic events.

Flooding of the plant's premises will render the mobile emergency diesel generator useless. It is placed on a lorry and as such is not protected against flooding. Also its limited on board fuel limits its use during flooding conditions.

5.1.2 Loss of off-site power and loss of the ordinary back-up AC power source: LOOP-SBO 1

This section covers the measures to cope with the plant state which includes a Loss Of Offsite Power (LOOP) combined with a Station Blackout (SBO). This SBO is limited to the loss of emergency grid 1, NS1 (SBO 1).

5.1.2.1 Design provisions to take account of LOOP and SBO 1

In this postulated scenario, NS1 will not be available. Many systems required for cooling are connected (for their electrical power) to NS1 and/or NS2.

Design provisions to prevent plant state LOOP - SBO 1

Provisions against LOOP have been addressed in section 5.1.1.1 of this National Report. In the same section implicitly, provisions to prevent failure of NS1 have been named:

- Physical separation of redundant parts of NS1, 3rd diesel in separated building as backup;
- Two external redundant connections from the coal-fired unit CCB to the main grid.

Internal Backup provisions to cope with LOOP – SBO1

With LOOP – SBO 1, emergency grid 1 (NS1) is not available. The main backup is provided by emergency grid 2 (NS2), already described in section 5.1.1.1. In summary, after failure of NS1, the backup resources are:

- Emergency grid 2 (NS2), will start in two seconds;
- Supply by coal-fired plant CCB with options:
    - Use of its 6 kV connection to transmit power from main grid to the NS1 of the NPP – if CCB does not suffer LOOP;
    - Supply from emergency diesels of CCB to NS2 of the NPP.
- Installation and connection of mobile diesel generator EY080, sufficient to power emergency grid 2 (NS2).

5.1.2.2 Autonomy of on-site resources and provisions to prolong on-site AC service time during LOOP – SBO 1

The autonomy of on-site resources has been addressed in section 5.1.1.2 for the LOOP conditions, as well as the required competence of the staff to make the necessary connections. This section provides autonomy data for the LOOP – SBO 1 plant state.

- With LOOP – SBO 1, the main AC source will be emergency grid 2 (NS2). Its diesels can start in two seconds. In section 5.1.1.2 it has been stated that its fuel storage is adequate to last about 72 hours. If other onsite fuel storages can be used, one EDG of

NS2 would have enough fuel to supply power during 1300 hours. However, there are no dedicated hardware provisions and procedures available to support the required fuel transfers, staff will have to improvise.

- In case of complete SBO, mobile EDG EY080 is available as a backup AC source. Transporting it onsite and connecting it to NS2 takes six hours. However, transportation requires external support, so it is not a real internal backup option.
- As long as AC power is available, the DC power system is in operation (via converters). As soon as AC power is lost, the DC-system will be powered by batteries. Discharge times are more than two hours, refer to section 5.1.1.2. The discharge time of the 220 V batteries can be extended to 5.7 hours, following certain procedures. Recharging batteries (when AC power is available again) takes eight hours.

5.1.2.3 Robustness of provisions in connection with seism and flooding

All safety relevant SSCs are able to maintain functioning at PGAs up to at least 0.15 g. Damage due to seismic events will be limited. SSCs needed for coping with floods will still be available after seismic events.

Flooding of the plant's premises will render the mobile emergency diesel generator useless. It is placed on a lorry and as such is not protected against flooding. Also its limited on board fuel will limit its use during flooding conditions.

5.1.3 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: LOOP-SBO 2

5.1.3.1 LOOP and SBO 2

(time to activate diverse AC power sources in this scenario, preparedness to take them into service, capability to make connections", tijd hiervoor nodig, tijd beschikbaar om kernbeschadiging en SF beschadiging te voorkomen)
In this postulated scenario, power from the main grid, NS 1 and NS2 will not be available. In addition it is assumed that power from emergency diesels of the coal fired plant CCB as well from the mobile diesel generator is not available. In this scenario, only power from batteries and the uninterrupted power systems are in operation.

Design provisions to prevent plant state LOOP - SBO 2

Provisions to prevent plant state SBO 2 are:

- Two physically separated parts of the NS 2 with each part having its dedicated EDG;
- Separate connection of NS 2 to public grid;
- Supply to NS 2 by the emergency power supply of the coal-fired plant CCB;
- Installation and connection of mobile diesel generator EY080 to NS 2.

Internal Backup provisions to cope with LOOP – SBO2

For the true LOOP – SBO 2 plant state, and not allowing external support to transport the mobile EDG, only the uninterrupted battery power system remains.

5.1.3.2 Autonomy of on-site resources and provisions to prolong on-site AC service time during LOOP – SBO 2

The autonomy of on-site resources has been addressed in section 5.1.1.2 for the LOOP conditions, as well as the required competence of the staff to make the necessary connections. This section provides autonomy data for the LOOP – SBO 2 plant state, in which the emergency grid 2 (NS2) is postulated not to be available.

If equipment like a truck is postulated to arrive not earlier than 24 hours after initiation of LOOP – SBO 2, these first hours, the mobile EDG can not be connected to NS 2.

The batteries will last more than two hours. The discharge time of the 220 V batteries can be extended to 5.7 hours by disabling the turbine oil pump. When AC power is available, it takes eight hours to charge the batteries.

If failure of NS 2 is caused by failure to provide fuel to its EDGs, transfer of fuel from other resources on-site is possible. This will bring the plant in the state 'LOOP – SBO 1', in which SSCs essential for safe shutdown will be available. However, there are no dedicated hardware provisions and procedures available to support the required fuel transfers, so staff will have to improvise.

If a truck succeeds to arrive at the site, on-site transport of the mobile EDG (EY080) and connection of it to NS 2 will take six hours.

There are arrangements to provide an alternative EDG, which will need to be transported from Rotterdam to Borssele. Transport of this mobile EDG and connection of it to NS 2 will take about eight hours.

If transport of fuel is possible, external supply of diesel may help to move from plant state LOOP – SBO 2 to plant state LOOP – SBO 1 or better.

5.1.4 Conclusion on the adequacy of protection against loss of electrical power.

KCB is sufficiently protected against a Loss of Offsite Power (LOOP). Two independent and redundant (3 x 100% and 2 x 100%) emergency power systems NS 1 and NS2 are available, of which NS2 is protected against external events like flooding, earthquake and explosion. These emergency grids are available to challenge a LOOP. As a defence in depth measure the emergency power system of the coal-fired plant CCB and/or a mobile diesel generator and an external diesel generator are available. If available, these equipment should be adequate in providing electric power to safe shut down, cooling (core and spent fuel) and preventing a radiological release.

The regulatory body recognizes that in the Licensee Report, for the assessment of ENSREG-postulated scenarios, licensee has given credit to various structures, systems and components (SSCs) that are not designed, classified or tested for their purpose in severe accident management. This is a common and acceptable approach for accident management past the design basis and for the purpose of the stress test this is acceptable too. However further assessments are recommended to establish the validity of the assumptions made regarding the associated SSCs.

Regarding the alternative power sources refer to section 5.1.5 for a regulatory position.

5.1.5 Measures which can be envisaged to increase robustness of the plants in case of loss of electrical power

Measures proposed by licensee to better challenge the plant state LOOP:

- in the Periodic Safety Assessment '10EVA13' the possibilities to strengthen the off-site power-supply will be investigated. This could implicitly increase the margins in case of LOOP as it would decrease the dependency on the SBO generators;
- establishing the ability to transfer diesel fuel from storage tanks of inactive diesels towards active diesel generators would increase the margin in case of loss of off-site power;
- reduction of the time necessary to connect the mobile diesel generator to emergency Grid 2 to two hours, would increase the margin in case of loss of all AC power supplies including the SBO generators;
- Develop a set of Extensive Damage Management guides (EDMG) and implement a training program. Issue to be addressed will be connecting CCB/NS1.

Measures proposed by licensee to better challenge the plant state LOOP – SBO 1

The measures to improve the robustness of the plant, so that the plant better matches SBO 1 conditions are identical to those that are listed for LOOP.

Measures proposed by licensee to better challenge the plant state LOOP – SBO 2

Measures to improve the robustness of the plant, in a way that the plant better matches SBO 2 conditions, are identical to those that are listed for LOOP and SBO 1. For SBO 2 additionally the following measures have to be completed:

- by training of the procedure ensure that during mid-loop operation, the actions for water supply that are needed in case of loss of all AC power supply, are performed in a timely manner;
- more extensive use of steam for powering an emergency feed water pump and for example an emergency AC generator could increase the robustness in case of loss of all AC power supplies including the SBO generators.

Regulatory position statements

- Licensee has proposed a set of measures that in principle can contribute to safety and can be endorsed by the regulatory body. However before implementation, their effectiveness needs to be assessed;
- The amount of lubricating oil or better a shortage of this oil in crisis situations, can be considered as a potential cliff edge, assessment of this topic is recommended;
- A set of clear criteria needs to be established as a basis for deciding when to switch the turbine oil pump off to increase the battery time. Disabling this pump will damage the turbine.
- The regulatory body has the opinion that the various alternative power sources should be re-evaluated:
    - As an alternative to emergency grid 2 (NS2) licensee mentions the emergency diesel generator (EDG) of the coal-fired plant. While this is a credible option, the EDG of CCB is not a nuclear class component and not tested for its purpose in accident conditions at the NPP. The mobile EDG (EY080) currently needs off-site support, which can not relied upon in all crisis situations. Also procurement arrangements for externally supplied EDGs are mentioned, which still have to be concluded. Some crisis conditions may make it impossible to transport an EDG to the site. Alternate and independent means to recharge batteries should also be part of the study.
    - Connection equipment and connection points for power sources, fuel resources etc. are available. Their ease of use and availability for all foreseeable circumstances should be analysed.

5.2 Loss of the decay heat removal capability/ultimate heat sink

ENSREG postulated: the connection with the primary ultimate heat sink (UHS) 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.

Many systems important to supply cooling become unavailable when certain AC power sources are lost. In addition cooling also becomes more complicated when some heat sinks become unavailable. The licensee has defined four plant states with combinations of loss of ultimate heat sink (LUHS) and station black out (SBO).

The following two plant states are variations on loss of ultimate heat sink, without loss of AC sources, and they are addressed in this section 5.2 and its subsections:

1. Loss of primary ultimate heat sink (LPUHS). The following systems are unavailable:
    a. the Main cooling water system (VC) (needed for bypass operation), and;
    b. the Conventional and emergency cooling water system (VF);
2. Loss of primary and alternate ultimate heat sink (LPAUHS). The following systems are unavailable:
    a. the Main cooling water system (VC), plus
    b. the Conventional and emergency cooling water system (VF), and
    c. the Backup cooling water system (VE);

The following two plant states feature combinations of loss of ultimate heat sink and loss of AC power sources. They are addressed in section 5.3 and its subsections, but are listed here for completeness:

3. Loss of primary ultimate heat sink (LUHS) and station black out (referred to as SBO 1). The following systems are unavailable:
    a. the Main cooling water system (VC),
    b. the Conventional and emergency cooling water system (VF),
    c. off-site power (LOOP), and
    d. Emergency Grid 1; NS 1;
4. Loss of primary ultimate heat sink (LUHS) and total loss of AC-power (referred to as SBO 2). The following systems are unavailable:
    a. the Main cooling water system (VC),
    b. the Conventional and emergency cooling water system (VF),
    c. off-site power (LOOP),
    d. Emergency Grid 1; NS1,
    e. Emergency Grid 2; NS 2,
    f. the emergency power system of the coal-fired power plant (CCB), and
    g. the mobile diesel generator EY080.

LOOP SBO combinations have been addressed before in section 5.1. In this section 5.2 it will be addressed in relation to versions of LUHS only.

5.2.1 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.

Provision to prevent LPUHS for the reactor and for the spent fuel pool

The primary ultimate heat sink (PUHS) is the water from the river Westerschelde. It is supplied to the NPP via the main cooling system (VC) and the emergency cooling system (VF). Pumps of these systems are located in the cooling water inlet building. Also refer to chapter 1 for systems descriptions.

The depth of the inlet channel and the depth directly in front of the inlet building should enable undisturbed water intake under most conditions. Regular dredging ensures this depth is maintained.

The main cooling system (VC) is further protected by mussel filters and other various filter arrays of the cooling water filtering system (VA). If there is not enough water to feed both VC and emergency cooling system VF, the VC is switched off-line since it is not safety relevant and priority is given to feed the emergency cooling system VF. Additional provisions to prevent failure of VF are:

- Emergency electricity supply (NS 1) for the VF pumps; and
- Two redundant VF trains with two redundant VF pumps each, fed by two redundant electric NS 1 systems.

In chapter 2 it has been explained that the Licensee Report has made plausible that seismic margins of safety relevant SSCs (including those related to cooling) will allow these to maintain their functionality up to PGAs of at least 0.15 g. The cooling water intake building housing the pumps for the main cooling system (VC) and emergency cooling system VF is watertight to a height of 7.4 meters.

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

5.2.2.1 Reactor cooling

The primary ultimate heat sink is the water of the Westerschelde, supplied by the main cooling water system VC and the conventional emergency cooling water systemVF. LPUHS is within the design basis of the plant. In the plant state LPUHS, two alternative heat sinks are:

- the atmosphere, in case of steam venting (over the roof) via the main steam relief valves of the Main steam system (RA):
  - the valves are redundant and feed water is supplied by two feed water systems, one of them redundant and protected against external events;
- eight deep-water wells supplying the Backup cooling water system (VE).
  - Only six out of eight wells are needed for design capacity;
  - Earthquake and flooding are part of the design basis of VE;
  - VE pumps are connected to emergency grid 2 (NS2);
  - VE connects to reactor cooling as well as to spent fuel basin cooling
  - VE can supply water in less than 13 hours;

For the reactor there are two cooling down phases; (1) the cooling down phase and (2) the decay heat removal phase. In parallel, for the spent fuel pool, pool cooling is needed.

The Licensee Report in its section 5.1.2.2 gives an extensive description of all available circuit configurations depending on the available systems. In addition the achievable cooling time is shown in diagrams for each option. The cooling status is summarized in Table 5-2 (copy of Table 5.6 of Licensee Report).

Table 5-2 Cooling status in case of Loss of Primary Ultimate Heat Sink (LPUHS)

Operational state	Means of cooling	Duration (h, approximated)	Remarks
Cooling down	RL supply until RL tank is empty RL continues cooling by RZ water supply	3 15	Decay heat removal conditions can be met after approximately 3 hours
	RS supply	60	
Decay heat removal	Re-establishment of the TJ/TF/VF cooling line by	6, 7[28] (3, 0)[29]	UJ stock is 1200 m^3, replenishment is

[28] It lasts for six or seven hours once the decay heat removal starts which will be three or 13 hours respectively after shut down using a water stock of about 1,000 m^3 that is effective when used for cooling.

[29] The results for combined decay heat removal and spent fuel pool cooling are presented in brackets. The differences in periods result from differences in pool loading.

	feeding VF by UJ Replenishment of UJ by public water supply system		required
	Switch over to TE/VE (ground water) cooling	unlimited	

The plant state LPUHS is within the design basis. It can be controlled by on site systems. No external actions are needed. It is preferable to stretch the cooling down phase (which uses evaporation). This will reduce water consumption. Heat removal by heating water (as with decay heat removal) requires about six times as much water stocks as cooling by evaporation.

5.2.2.2 Spent fuel pool cooling

In case regular method of spent fuel pool (SFP) over the conventional emergency cooling water system VF is not available due to loss of the primary ultimate heat sink (LPUHS), three main alternative options (with several variations) exist, that depending on availability of systems and resources can be used:

- Start a cooling chain via the normal spent fuel cooling system TG, component cooling system TF, conventional emergency cooling water system VF, and supply this chain with water – depending on availability – from:
  - low pressure fire extinguishing system UJ;
  - fire truck taking suction form fire fighting pond of coal fired plant CCB or from river Westerschelde (unlimited resource);
- Start a cooling chain via normal spent fuel cooling system TG, using the reserve heat exchanger 'TG080', and connect to Backup cooling water system VE, and supply this chain with water – depending on availability – from:
  - deep water wells connected to VE;
  - low pressure fire extinguishing system UJ;
  - fire truck taking suction form fire fighting pond of coal fired plant CCB or from river Westerschelde (unlimited resource);
- Start cooling via evaporation of the water of the spent fuel pool,
  - With replenishment of water from:
    - Safety injection system & residual heat removal systems TJ;
    - Demineralised water distribution system;
    - Fire extinguishing systems indirectly draining from fire fighting pond or river Westerschelde.
  - No replenishment, just evaporation.

Note that LPUHS is within the design base.

The Licensee Report in its section 5.2.2 gives an extensive description of all available circuit configurations depending on the available systems. In addition the achievable cooling time for each option is given and shown in diagrams. The cooling status is summarized in Table 5-3.

Table 5-3 Cooling status for SFP with LPUHS

Plant state	Means of cooling	Duration (h, approximated)	Remarks
Cooling pool	By re-establishing the TJ/TF/VF cooling line, pool cooling via TG/VF is also re-established Supply to UJ by public water supply system	6, 13 unlimited	Normal UJ stocks are limited
	Switch over to TG080/VE cooling	unlimited	

5.2.3 Loss of the primary ultimate heat sink and the alternate heat sink - LPAUHS

5.2.3.1 Reactor

In case of a loss of primary and alternate ultimate heat sink (LPAUHS), not only is the heat sink of the conventional emergency cooling water system VF (the River Westerschelde) not available anymore, but also the heat sink of the backup cooling water system VE (deep water wells). As a result cooling will be provided by other available water reservoirs.

Also in this scenario, a cooling down phase, decay heat removal phase and spent fuel cooling phase can be identified.

All options available for the cooling down phase with LPUHS are also available for LPAUHS. The options for the cooling down phase meet the 13 hours criterion[30] set by the decay heat removal chain TE/VE which relies on the starting of the use of well water.

However in the decay removal phase, no credit can be given to the TE/VE system, assuming the unlikely event of unavailability of VE. Replenishment of UJ by public water supply will extend the decay heat removal for the course of the event. Additional supply may come from the fire fighting pond at the CCB and an unlimited supply from the river Westerschelde. Refer to Table 5-4 (copy of Table 5.7 of Licensee Report).

Table 5-4 Cooling status in case of Loss of Primary and Alternate Heat Sink (LPAHS)

Operational state	Means of cooling	Duration (h, approximated)	Remarks
Cooling down	RL supply until RL tank is empty	3	Decay heat removal conditions can be met after approximately 3 hours
	RL continues cooling by RZ water supply	15	
	RS supply	60	
Decay heat removal	Re-establishment of the TJ/TF/VF cooling line by feeding VF by UJ	6 - 7[31] (3, 0)[32]	UJ stock is 1200 m^3, replenishment is required
	Replenishment of UJ by public water supply system	unlimited	
	Supply from CCB’s fire fighting pond or	8, 10	
	The river Westerschelde	unlimited	

5.2.3.2 Spent fuel pool

With loss of primary and alternative ultimate heat sink (LPAUHS), the option to use water from the deep water wells via VE, does not exist. The three main cooling options listed in section 5.2.2.2 remains valid, except that water source ‘deep water wells’ needs to be excluded.

The Licensee Report in its section 5.2.2 gives an extensive description of all available circuit configurations depending on the available systems. In addition the achievable cooling time for each option is given and shown in diagrams. The cooling status is summarized in Table 5-5

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[30] Thus these options last long enough to bridge the 13 hours needed to get VE at the required cooling water output.

[31] It lasts for six or seven hours once the decay heat removal starts which will be three or 23 hours respectively after shut down using a water stock of about 1,000 m^3 that is effective when used for cooling.

[32] The results for combined decay heat removal and spent fuel pool cooling are presented in brackets. The differences in periods result from differences in pool loading and the start and end points of several activities.

Table 5-5 Cooling status for SFP with LPAUHS

Plant state	Means of cooling	Duration (h, approximated)	Remarks
Cooling pool	By re-establishing the TJ/TF/VF cooling line, pool cooling via TG/VF is also re-established Supply to UJ by public water supply system	6, 13 unlimited	Normal UJ stocks are limited
	Supply from the fire fighting pond at CCB Supply from river Westerschelde	8 (0, 14)[33] unlimited	

5.2.4 Conclusion on the adequacy of protection against loss of ultimate heat sink

The plant state of LPUHS is controlled by on-site systems and is within the design basis.

The plant state of LPAUHS is beyond the design basis but can be controlled provided that additional supply of water from the public water system or ultimately from the river Westerschelde can be provided. Otherwise the application of the main options and their alternatives will end when the stocks are exhausted. Time periods for LPAUHS depend on the following available options:

- the cooling down phase can be extended for more than 14 days by applying all available on-site stocks;
- the decay heat removal phase only relies on UJ or fire truck supply, which will last 10 hours and 13 hours respectively (relying on on-site stocks) when decay heat removal starts three hours after reactor shut down, and 11 hours and 16 hours respectively when decay heat removal starts 13 hours after reactor shut down;
- the spent fuel pool cooling can be extended for more than 14 days when evaporation is accepted. With replenishment from public water system or river Westerschelde, cooling can be sustained for a unlimited time, assuming enough power sources are available.

It is noted that the licensee gives much credit to the fire extinguishing system UJ, which is used in many circuit combinations to control the LPAUHS state. Assessment of its proper classification for such tasks is advised.

5.2.4.1 Spent fuel pool

5.2.5 Measures which can be envisaged to increase robustness of the plants in case of loss of ultimate heat sink

Measures proposed by licensee EPZ

The licensee has proposed to further develop a set of Extensive Damage Mitigating Guides (EDMGs) and implement an associated training programme.

With regards to the spent fuel cooling system, licensee proposes several additional measures to further improve robustness of the plant, they are:

[33] Results for combined decay heat removal and SFP cooling in brackets. Variations due to differences in pool loading possible, and start and end points of several activities.

- a reserve spent fuel pool cooling system that is independent of power supply from the emergency grids, could expand accident management possibilities. In 10EVA13 this will be investigated;
- a possibility for refilling the spent fuel pool without entering the containment would increase the margin to fuel damage in certain adverse containment conditions;
- additional possibilities for refilling the spent fuel pool would increase the number of success paths and therefore increase the margin to fuel damage in case of prolonged loss of spent fuel pool cooling;
- develop a set of Extensive Damage Management Guides (EDMG) and implement a training program. Issues to be addressed:
  - description of the alternative ways to replenish the fuel storage pool;
  - injection of fire water directly into the fuel storage pool by a flexible hose;
  - cooling the fuel storage pool by TG080/VE supplemented by UJ;
  - connection of TN to the suction side of the fuel storage pool cooling pumps;
  - procedure for spent fuel pool cooling (over spilling, make up);
  - flexible hose connections to the TG system and the spent fuel pool;
  - procedure for direct injection of VE by UJ;
  - use of autonomous mobile pumps;
  - possible leak repair methods for larger pool leakage.

Regulatory position statements

- Licensees proposals seem to contribute to safety and in principle can be endorsed by the regulatory body, after assessment of their effectiveness;
- It is noted that the licensee gives much credit to the fire extinguishing system UJ, which is used in many circuit combinations to control the LPAUHS state. Assessment of its proper classification for such tasks is advised.

5.3 Loss of the primary ultimate heat sink, combined with station black out (see stress tests specifications).

The following two plant states feature combinations of loss of ultimate heat sink and loss of AC power sources:

1. loss of primary ultimate heat sink and station black out (referred to as SBO 1). In this situation, power from emergency grid 2 (NS2) is still available, as well as power from batteries, the UPS and the mobile EDG as backup of NS2. The following systems are unavailable:
   a. the Main cooling water system (VC);
   b. the Conventional and emergency cooling water system (VF);
   c. off-site power (LOOP), and
   d. Emergency Grid 1; NS 1.
2. loss of primary ultimate heat sink and total loss of AC-power (referred to as SBO 2). The following systems are unavailable:
   a. the Main cooling water system (VC);
   b. the Conventional and emergency cooling water system (VF);

c. off-site power (LOOP);
d. Emergency Grid 1; NS1;
e. Emergency Grid 2; NS 2,
f. the emergency power system of the coal-fired power plant (CCB), and
g. the mobile diesel generator EY080.

5.3.1 Time of autonomy of the site before loss of normal cooling condition of the reactor core and spent fuel pool (e.g., start of water loss from the primary circuit).

5.3.1.1 Reactor

LPUHS and SBO1

No fuel damage will occur with the combination of LPUHS and SBO1. This situation is under control; cold shutdown will be achieved and alternative cooling options are available. It will be preferable to make the cooling down phase as long as possible because it uses evaporation, which uses much less of the water stocks than heat removal by heating water.

The Licensee Report in its section 5.1.3 gives an extensive description of all available circuit configurations depending on the available systems. In addition the achievable cooling time is shown in diagrams for each option. The cooling status is summarized in Table 5-6.

Table 5-6 Cooling status with LPUHS in combination with SBO

Operational state	Means of cooling	Duration (h, approximated)	Remarks
Cooling down	Start RL cooling down using RL tank supply RS will take over as soon as SG level is low	3 75	
	RL (alternative to RS) including UJ supply	3 111	RS can also be started without the preceding RL cooling
Decay heat removal	TE/VE combination provides sufficient decay heat removal	Basically unlimited	
	UJ Public water supply system	7 (0)[34] unlimited	

LPUHS and SBO2

Compared with the 'normal' LPUHS scenario's, all options using electrical systems connected to the ordinary grid or one of the emergency power supply systems are not available anymore. The main and auxiliary feedwater system (RL) remains available but only until its water reserves are exhausted. The steam turbine driven pump can be operated or secondary bleed and feed can be applied for about three hours. The low pressure fire extinguishing system (UJ) with its own diesel driven pump can operate (using its own stock of diesel) for about eight hours.

[34] For combined decay heat removal and spent fuel cooling, results between brackets.

No fuel damage needs to occur if the reactor is kept at hot shut down conditions. The secondary system can sustain this situation as long as external supply of water will exist. Ultimately, the water is supplied by the public water system and/or drained from the river Westerschelde. Table 5-7 lists the remaining options.

Table 5-7 Cooling status in case of LPUHS and SBO2

Operational state	Means of cooling	Duration (h, approximated)	Remarks
Cooling down	RL start of cooling down by supply of RL tank	3	Either the supply is provided by steam driven RL pump or by secondary bleed & feed
	Supply from UJ trough RS Public water supply	11 (48)[35] Unlimited	
	Depressurize steam generators Supply with fire trucks from fire fighting pond Use of Westerschelde water	< 1 131 Unlimited	
Decay heat removal	No option available		

5.3.1.2 Spent fuel pool

LPUHS and SBO1

With SBO1, two cooling options remain:

- using cooling chain via normal spent fuel cooling system TG, using the reserve heat exchanger 'TG080', and connect to Backup cooling water system VE, and supply this chain with water – depending on availability – from several sources like deep water wells, public water supply system or fire trucks, drawing from fire fighting pond or Westerschelde;
- using evaporation , with replenishing from public water supply system, or just evaporation of no pool cooling system is active. This will decrease the water level.

LPUHS and SBO2

With SBO2 only evaporation remains an option. With replenishment from the public water supply system this situation can be maintained indefinitely. If replenishment is not possible, water level in the pool will drop which will result in increasing dose rates in this area.

Next table lists the situation for SBO1 and SBO2.

[35] In brackets the combination of cooling down and spent fuel pool cooling with allowing the pool to heat up.

Table 5-8 Cooling status spent fuel pool for LPUHS combined with SBO1 and SBO2

Plant state	Means of cooling	Duration (h, approximated)	Remarks
Loss of primary ultimate heat sink with station black-out type 1, LPUHS-SBO 1	TG080/VE combination provides sufficient cooling	Basically unlimited	
	UJ	6 (13)[36]	
	Public water supply system	unlimited	
Loss of primary ultimate heat sink with station black-out type 2, LPUHS-SBO 2	Evaporation of pool water and refill via UF, fire truck and UJ	180 (48)	
	Public water supply system	unlimited	
	Heat up and evaporation of pool water until top of fuel reached	84	

5.3.2 External actions foreseen to prevent fuel degradation

Emergency diesel reserves for emergency diesel generators (EDGs) are designed to last for several days (72 hours) but could be extended to about 1,300 hours while exhausting all on-site stock. After this period, new diesel supplies from external suppliers should be brought in to extend the period of EDG operation.

Contracts for delivering diesel fuel exist.

The on-site mobile EDG (EY080) currently needs external support to be moved over the premises of the NPP.

5.3.3 Measures, which can be envisaged to increase robustness of the plants in case of loss of primary ultimate heat sink, combined with station black out

Measures proposed by licensee

Measures described in section 5.1.5 regarding improving robustness against LOOP, SBO1 and SBO apply also to combinations of these with LPUHS.

Measures described in section 5.2.5 to increase robustness against LPUHS and LPAUHS also apply to combinations of SBO and LPUHS.

A potential action can be envisaged to prevent running out of diesel of on site supply for fire extinguishing system UJ and the fire brigade. An increased number of diesel supplies at separate locations of the plant can be a solution.

Regarding the spent fuel pool cooling, the licensee proposes to consider the following measures:

- a reserve spent fuel pool cooling system that is independent from power supply from the emergency grids, could expand accident management possibilities. In 10EVA13 this will be investigated;

[36] The variant is due to a difference in pool charge (full core inventory vs 1/3 core inventory) and core cooling by UJ via the steam generator 13 hours after reactor shut down the differences in periods result from the differences in pool loading; Decay heat removal supplied by UJ should start when the UJ stock has emptied after 13 hours of spent fuel pool cooling

- a possibility for refilling the spent fuel pool without entering the containment would increase the margin to fuel damage in certain adverse containment conditions;
- additional possibilities for refilling the spent fuel pool would increase the number of success paths and therefore increase the margin to fuel damage in case of prolongued loss of spent fuel pool cooling;
- implementing the following procedures:
  - description of the alternative ways to replenish the fuel storage pool;
  - injection of fire water directly into the fuel storage pool by a flexible hose;
  - cooling the fuel storage pool by TG080/VE supplemented by UJ;
  - connection of TN to the suction side of the fuel storage pool cooling pumps;
  - procedure for spent fuel pool cooling (overspilling, make up);
  - flexible hose connections to the TG system and the spent fuel pool;
  - procedure for direct injection of VE by UJ;
  - use of autonomous mobile pumps;
  - possible leak repair methods for larger pool leakage.

Regulatory position statements

- Measures described in section 5.1.5 regarding improving robustness against LOOP, SBO1 and SBO apply also to combinations of these with LPUHS;
- Measures described in section 5.2.5 to increase robustness against LPUHS and LPAUHS also apply to combinations of SBO and LPUHS;
- The highest temperature in the core during the cooling statuses mentioned in the Licensee Report is not analyzed by the licensee. So further study is recommended;
- The case of a SBO-2 situation without Main and auxiliary feed water system RL or Main steam system RA (systems required for secondary feed and bleed) is not analyzed. So further study is recommended;
- The regulatory body recognizes that in the Licensee Report, for the assessment of ENSREG-postulated scenarios, licensee has given credit to various structures, systems and components (SSCs) that are not designed, classified or tested for their purpose in severe accident management. This is a common and acceptable approach for accident management past the design basis and for the purpose of the stress test this is acceptable too. However further assessments are recommended to establish the validity of the assumptions made regarding the associated SSCs.

6. Severe accident management (SAM)

This chapter focuses on mitigating actions to be taken in case of and after occurrence of damage to reactor and / or spent fuel, to prevent mainly large release of radioactive substances into the environment. As such, the focus is mainly on protection of the containment integrity.

For a detailed description of the SAM arrangement, refer to the corresponding Licensee Report. In this National Report, the main aspects will be highlighted and commented where necessary.

6.1 Organization and arrangements of the licensee to manage accidents

There are no statutory regulations in the Nuclear Energy Act requiring the presence of an on-site emergency preparedness plan. The basic requirements for the station's emergency preparedness are given in the operating licence. The licensee is responsible for on-site emergency responses and for providing plant status information to the authorities for off-site response. The public authorities are responsible for off-site response and for providing information to the general public.

6.1.1 Organisation of the licensee to manage the accident

Staffing and shift management in normal operation

There are 7 operations shift team, each managed by a shift supervisor and each composed of at least eight operators. Within a team there is a clear hierarchy and description of tasks, therefore the chain of command is always clear.

Measures taken to enable optimum intervention by personnel

The licensee has an emergency plan. It describes among others, when the plan is applicable, the emergency response organisation (ERO), possible measures, overview of emergency centres, equipment, and various types of periodic drills.

The ERO supports plant operation in accident and severe accident conditions. ERO combines an industrial safety and nuclear emergency organisation. Its functions are:

- notification to and cooperation with, external response organisations in case of an accident;
- provision of protective actions on the site, mitigating the consequences should an accident occur;
- administering first aid to injured persons;
- recovery of endangered persons and fire-fighting on the site;
- notification to and assembling of people on the site in case of an emergency;
- site security;
- aftercare of an accident.

The shift supervisor is the first to decide on the extend of the ERO to be activated. Once ERO is operational, the Site Emergency Director (SED) takes over this responsibility. The SED has an emergency management team that oversees all duties of the ERO.

There is a liaison officer who liaises with the Regional Operational Team (local government) in Middelburg. This staff member will be sent to the ROT to provide technical explanations on the plant status and proposed actions.

ERO is a scalable organisation. Number of ERO-staff members altered will depend on the scale of the incident at hand.

The majority of ERO staff is on call via pagers and sometimes cell phones. Shift personnel, security personnel and first aid personnel are always present. There are enough qualified persons available for each function in ERO.

Use of off-site technical support for accident management

Emergency communication starts with a call from EPZ to the notification point of the local authorities which is sited in Middelburg, and then to the notification point of the national authorities at the Ministry of Infrastructure and Environment in The Hague. After alerting the authorities, the plant's ERO will communicate with the ROT at local level and with the think tank of the Kernfysische Dienst (KFD; Nuclear Safety Department) at national level. The ERO uses written reports or Situation Reports (SITRAPs) to inform and advise both local and national authorities. The ROT will distribute this information within the local authorities.

The KFD has a process computer station in The Hague and is therefore able to monitor plant parameters online.

EPZ has a contract with the vendor of the plant (now named AREVA) to assist the plant's ERO with calculations and technical support in case of an emergency. This assistance is provided by the so called 'Krisenstab' (crisis staff). If needed this group of (nuclear) engineers can provide on-site assistance. The Krisenstab has all the engineering details and plant procedures in their offices in Germany that they might need to give this assistance. It is also possible to use an online data connection with the process computer (the PPS) of the plant.

EPZ has agreements with the Admiraal de Ruyter Hospital in Goes and the Academic Medical Center in Leiden to treat radioactive contaminated casualties and irradiated persons. This means that these hospitals have trained staff, equipment and procedures to provide specific treatment.

The ERO has a direct telephone line to the combined call centre for the police, fire department and ambulance in Middelburg. The shift supervisor or SED can ask directly for assistance when needed.

Procedures, training and exercises

Procedures are in use for all operational states (from shutdown through to full power) to operate the plant in all possible plant (damage) states:

- normal conditions (all operational states);
- abnormal conditions (all operational states);
- accident conditions;
- severe accident conditions.

Severe Accident Management Guidelines (SAMGs) are entered upon criteria that identify imminent or occurring core melt conditions. The Borssele SAMGs are based on the generic WOG SAMGs.

Control room personnel and relevant technical staff get periodical re-training in handling emergency procedures.

Licensee EPZ has its own full-scope simulator at the simulator centre in Essen (Germany). It is used to train control room personnel in the plant specific emergency procedures. This simulator is also used in so-called integrated emergency exercises. During these exercises, a shift group works in the simulator control room, while the plant's ERO is in the emergency response centre (ACC: Alarm Coördinatie Centrum) in Borssele. A data link connection between the ACC in Borssele and the Process Presentation System (PPS) of the simulator in Essen enables the plant's ERO in Borssele to monitor the 'plant status' constantly and to observe the effects of actions taken. The KFD in The Hague and the AREVA Krisenstab in Germany can also see live data from the simulator's Process Presentation System during the exercise.

The use of the simulator enhances the sense of realism of the integrated emergency exercises. The integrated exercises are the ultimate tests for evaluating of overall emergency preparedness.

Training has gradually shifted from predominantly technical training, towards training of skills required for the organisation and communication by the plant's ERO. This approach should benefit the emergency exercises which are held annually in cooperation with local and national authorities.

Apart from integrated exercises for the entire plant's ERO, there are several other types of training, such as on-the-job training, exercises on separate tasks of the emergency organisation, table-top exercises and separate instructions for groups within the emergency organisation.

A report on the overall emergency planning and preparedness is issued annually. These reports furnish information for the two- and ten-yearly safety evaluations of the nuclear power plant.

Plans for strengthening the site organisation for accident management

Currently the procedures are such that during a fire, the deputy shift supervisor takes the lead of the fire response organisation. The intended improvement will be that the team leader of the fire response team will be provided by the security department. This will leave the deputy shift supervisor in the control room during such incidents.

In case of physical isolation of the plant by an external hazard (like a flooding) additional personnel will be mobilized to strengthen the staff of the plant for the sake of emergency preparedness.

6.1.2 Possibility to use existing equipment

Provisions to use mobile equipment

A mobile diesel generator EY080 is available on the site and a supply of fuel is foreseen. A delivery contract for a second mobile diesel generator from an offsite location is available. A externally provided truck is needed to transport the onsite diesel generator to the connection point. That means that for any mobile diesel to be employed, external support is needed.

For the onsite diesel generator about 6 hours is needed to transport and connect, for the offsite diesel generator approximately 8 hours is needed. Both times assuming the infrastructure is not too much damaged.

The on-site fire brigade has several fire trucks available, and a modified crash tender as used in airports. Accessory equipment like fire hoses is available too, also to be used as a last-resort option of cooling water transportation.

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

The water and diesel reserves available at the KCB premises can be found in Chapter 5.

There are contracts for delivery of chemicals and fuel for the diesel generators, these should guarantee delivery within eight hours. However there are no special statements about emergency situations in the contracts.

There are two warehouses on site with various equipment and spare parts, among which material that can be used during emergency situations.

Management of radioactive releases, provisions to limit them

The strategies used to limit radioactive releases after core melt are provided by the SAMGs. An extensive summary of the SAMGs can be found in the Licensee Report in its Annex 6.1. The plant has provisions for filtered venting in emergency situations.

Contaminated water produced during an accident can be stored in the controlled area in the storage and waste water tanks which are normally used for contaminated process water.

Communication and information systems (internal and external)

During the use of the SAMGs, information will be communicated from the control room to the plant's ERO operating in the ACC bunker and from the plant's ERO to the control room, the (safety) authorities and AREVA's Krisenstab. During implementation of a severe accident management strategy, some dialogue will be required to complete the implementation steps.

The plant Process Presentation System (PPS) computer is used by the safety authority KFD and AREVA to obtain process information.

The ERO at Borssele uses telephones for external and internal communication. There is an emergency telephone network that serves as a backup. Furthermore the national emergency telephone network can be used. The plant's fire brigade can also communicate with the national C2000 (emergency partners) communication network.

For communication purposes, the use of fax, e-mail and pagers is also allowed. Radio communication is used for contact between different field teams.

A potential measure is the establishment of independent voice and data communication under adverse conditions, both on-site and off-site, which would strengthen the emergency response organisation.

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

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

Normally, the plant can be reached from three directions. Use of mobile resources will be hindered if roads are destroyed or they and/or the premises of the NPP are flooded.

Shift personnel are always on site. In case all roads are closed, an alternative means to relief them may be sending a new shift by helicopter – if (weather) conditions allow this. Currently a limited number of arrangements are in place for off site support measures.

Loss of communication facilities / systems

Extensive damage to the infrastructure around the plant could include the communications facilities like external telephone lines.

In case of an emergency situation, emergency response is coordinated from the emergency response centre (ACC: Alarm Coördinatie Centrum), which is located in a separate building on site. This centre is designed for internal events and emergencies and is not protected against flooding of the site, an earthquake or a large airplane crash in the vicinity of the reactor building. The meeting room above the main control room or any other meeting room on the site not damaged by the event could be used as a backup for the ACC. Because these meeting rooms do not have the provisions of the ACC, it is recommended to prepare a facility that is available as emergency response centre during and after the occurrence of large external events. This emergency response centre could give shelter to the emergency response organisation after all foreseeable hazards and would enlarge the possibilities of the emergency response organisation.

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

Work performance during severe accidents can be impaired due to high local doses rates. Dose measurement, shielding, protective clothing, respirators and limitation of the exposure time will be used to keep the dose of the workers within the required limits.

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

Licensee has analysed dose rates in various locations as a function of time after core melt with a major occurrence of fuel damage, conservatively set at 100%. It can be established that areas essential to manage the situation are accessible and habitable in this situation. An exception is a scenario with flooding of the site; in this case the ACC bunker for the ERO may not be accessible.

With filtered release from the containment, the containment filtered venting system will be used. It is efficient at filtering fission products, except noble gases. Nevertheless its effectiveness will allow habitation of the main control room, emergency control room and the ACC.

With unfiltered release from the containment a large spreading in the consequences can be considered, depending on the applicable conditions. Ventilation of main control room can be switched to internal filtered circulation. In addition a sufficient amount of respirators with compressed air is available. The ACC bunker has gas-tight doors and a filtered air supply.

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

The emergency control room is designed to withstand external hazards initiators and is equipped with the controls that are necessary to bring and keep the plant in a stable situation.. The ACC where the ERO team will be located during an accident is protected against radioactive releases but will be lost after flooding and probably also after a severe earthquake. The meeting room above the main control room or any other meeting room on the site not damaged by the event could be used as a backup for the ACC, but they lack the provisions of the ACC. The regulatory body endorses licensee's recommendation to prepare a facility that is available as emergency response centre during and after the occurrence of large external events.

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

The plant's severe accident management guidelines (SAMGs) deal with the management of accident situations induced by external hazards like earthquakes and floods. The regulatory body recommends further assessments regarding the impact of flooding, also refer to chapter 3 and chapter 7 'General conclusions' for the regulatory position.

Unavailability of power supply

Provisions regarding LOOP, SBO etc have been addressed in chapter 5 of this National Report. Transport of the emergency diesel EY080 is currently not mentioned or described in a separate procedure. Procedures to put the generator in operation are available. Reducing the connection time would increase the margins in case of loss of all AC supplies. The regulatory body recommends re-assessment of the various alternative power sources regarding their availability under accident conditions and proper classification for their intended purposes during accident management.

Potential failure of instrumentation

Instrumentation important for SAM purposes is qualified for operation under severe accident conditions. The power required by the instrumentation is provided by the emergency supply system and batteries.

It should be noted however that the Severe accident management guidelines (SAMGs) are developed on the basis that any instrument believed to provide useful information will be used, whether it is qualified for beyond design conditions or not. Environmental conditions during (a part of) a severe accident scenario may not exceed the conditions the instrument is qualified for.

In case of (potential) failures of some vital instrumentation, procedures for operating or stabilising 'by hand' are available. Also reserve instrumentation for nuclear-safety relevant systems is available.

Potential effects from the other neighbouring installations at site, including considerations of restricted availability of trained staff to deal with multi-unit, extended accidents.

The KCB is a single unit NPP. The presence of the nearby coal fired plant (the CCB) on the premises of EPZ is beneficial in the sense that during severe accident management some of its resources can be used. A potential threat posed by the CCB could be generation of missiles from its turbines. However in previous assessments it has been established that these will not pose a hazard to the SSCs of the NPP that are needed to bring it into a safe shutdown state.

6.1.4 Conclusion on the adequacy of organisational issues for accident management

Legal arrangements like the Nuclear Energy Act and the National Plan for Nuclear Emergency Planning and Response are in place. There is clarity about the responsibilities for emergency response and provision of information by licensee and public authorities. An emergency plan is available and an ERO is in place. The KCB emergency organisation has been synchronized with the local and national crisis organisation and is in conformity with the new 'Wet Veiligheidsregio's' released 1 October 2010. Agreements with external organisations, like the inspectorate KFD, the local authorities (ROT Veiligheidsregio), the crisis staff of the plant vendor (AREVA) and the local hospitals have been made for off-site support.

Based on the abovementioned arrangements it can be concluded that licensee satisfies the requirements for accident management, as required by its licence.

Nevertheless, further improvements are possible.

6.1.5 Measures which can be envisaged to enhance accident management capabilities

Measures proposed by licensee EPZ

Below the associated proposals of the licensee are reproduced:

- emergency response centre facilities that could give shelter to the emergency response organisation after all foreseeable hazards would enlarge the possibilities of the emergency response organisation;
- establishing independent voice and data communication under adverse conditions, both on-site and off-site, would strengthen the emergency response organisation;
- storage facilities for portable equipment, tools and materials needed by the emergency response organisation that are accessible after all foreseeable hazards would enlarge the possibilities of the emergency response organisation.
- develop a set of Extensive Damage Management Guides (EDMG) and implement a training program. Below are examples of the issues to be addressed:
    - description of the alternative ways to replenish the fuel storage pool;
    - injection of fire water directly into the fuel storage pool by a flexible hose;
    - cooling the fuel storage pool by TG080/VE supplemented by UJ;
    - connection of Demineralised water distribution system TN to the suction side of the fuel storage pool cooling pumps;
    - procedure for spent fuel pool cooling (over spilling, make up);
    - flexible hose connections to the Spent fuel pool cooling system TG and the spent fuel pool;
    - procedures to staff the emergency control room;

  - procedure for direct injection of Backup cooling water system VE by Low pressure fire extinguishing system UJ;
  - use of autonomous mobile pumps;
  - possible leak repair methods for larger pool leakage;
  - procedure to transport own personnel to the site;
  - procedure for the employment of personnel for long term staffing;
  - develop check-lists for plant walk-downs and needed actions after various levels of the foreseeable hazards;
  - connecting power from coal fired plant CCB to emergency grid 1 (NS1);
  - alternative supplies for Low pressure fire extinguishing system UJ.

Regulatory position statements

- Licensee in its Licensee Report already has identified options for further improvement of its severe accident management. In principle many of these options can be endorsed by the regulatory body. However before implementation their effectiveness needs to be assessed;
- It is recommended to study the world-wide post-Fukushima developments regarding SAMGs and improve on those in use with licensee EPZ where necessary;
- It is recommended to re-assess the staffing of licensee's emergency response organisation (ERO), considering:
  - Criteria for deploying additional shifts based on (predicted) evolution of the external hazard – also the number of additional people needed in different situations should be part of the evaluation;
  - How to guarantee to keep available 24/7 throughout the whole year two additional shifts;
  - How to act if for some reason ERO is not complete.
- It is recommended to re-assess the contents and frequency of the SAMG training program. Important aspects to consider are harsh conditions like:
  - Reduced accessibility of the site;
  - Reduced number of ERO staff;
  - Reduced availability of instrumentation;
  - Long duration of the accident.

6.2 Accident management measures in place at the various stages of a scenario of loss of the core cooling function

6.2.1 Before occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes (including last resorts to prevent fuel damage)

The Licensee Report in its Annexes to its chapter 6 provides Function Restoration Procedures and Emergency Operating Procedures that are tailored to prevent fuel damage. If these procedures fail, these procedures transit into an associated severe accident management guideline (SAMG).

Relevant procedures mentioned in the Licensee Report are (numbers refer to numbering in Licensee Report):

- Function Restoration Procedure C-1: Actions in case of insufficient core cooling (Annex 6.3);

- Function Restoration Procedure H-1: Actions on loss of secondary heat removal (Annex 6.4);
- Function Restoration Procedure S-1: Actions to restore subcriticality (Annex 6.5);
- Emergency Operating Procedure ECA-0-0: Actions in case of loss of auxiliary power (Annex 6.6);
- S-EY-01: Recovery instruction for emergency power system Emergency Grid 1;
- S-EY-02: Recovery instruction for emergency power system Emergency Grid 2.

6.2.2 After occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes

Relevant procedures mentioned in the Licensee Report are (numbers refer to numbering in Licensee Report):

- injection into the reactor coolant system: Severe Accident Management Guideline SAG-3;
- depressurising the reactor coolant system: Severe Accident Management Guideline SAG-2.

More than one SAG may be evaluated at a time and the implementation of strategies should follow the priorities dictated. Licensee mentions other SAGs that might be important with respect to core cooling after the occurrence of fuel damage:

- injection into the steam generators: Severe Accident Management Guideline SAG-1,
- injection into the containment: Severe Accident Management Guideline SAG-4.

6.2.3 After failure of the reactor pressure vessel/a number of pressure tubes

After failure of the reactor vessel debris will leave the primary system. The licensee has accident management guidelines that provide measures for cooling ex-vessel debris outside the cavity and also for scrubbing fission product releases. Examples are from SAMG SAG-4 (numbers refer to numbering in Licensee Report):

- Injection of water from the Safety injection system & residual heat removal system TJ storage tanks by the containment spray pumps;
- Injection of water from the TJ storage tanks by gravity drain.

Licensee reports studies with respect to cooling core debris inside the cavity. Conclusion is that the only reliable way to get water into the reactor cavity in the KCB design is via the reactor system after vessel failure. This is addressed in Severe Accident Management Guideline SAG-3. Refer to Annex 6.1 of the Licensee Report for details.

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

6.3.1 Elimination of fuel damage / meltdown in high pressure

Different SSCs can be used varying from the pressuriser relief valves, different pressuriser spray options to the opening of venting lines. There are no extra, special SAM design provisions installed for the elimination of fuel damage in high pressure.

The accident management measures to decrease the primary pressure before core melt and so for eliminating the possibility of fuel damage at high pressure are described in the following procedures (numbering from Licensee Report):

1. Function Restoration Procedure C-1 (Annex 6.3);

2. Function Restoration Procedure H-1(Annex 6.4).

The SAM guideline SAMG-SAG-2 gives multiple approaches to decrease the primary pressure after a core melt. An overview of all the possible systems is given in the Licensee Report's Annex 6.1.

6.3.2 Management of hydrogen risks inside the containment

The NPP is equipped with Passive Autocatalytic Recombiners (PARs) located in the containment. They do not require electrical energy for their functioning. The PARs can recombine hydrogen faster than it will be produced during the molten core concrete interaction (MCCI) phase of a severe accident. Since the spent fuel pool (SFP) is located in the containment, the PARs are effective in recombining hydrogen releases from the SFP should these occur.

In the unlikely event that the PARs would malfunction there are two SAMGs that provide guidance to safely manage the occurrence of hydrogen by among others opening of relief hatches, increase of steam concentration, injection of nitrogen, filtered venting etc.

Hydrogen production by molten core concrete interaction (MCCI) might be a smaller challenge for the KCB NPP than for other NPPs because of the high content of carbonates in the concrete. MCCI in the KCB will lead to relatively higher production of carbon dioxide, which will have an inerting effect.

The KCB has a dedicated hydrogen measurement system with sample points in the operational area and in the installation area. The SAMG strategies include the use of handheld sampling systems as a backup option.

6.3.3 Prevention of overpressure of the containment

The plant has a containment venting line with a wet scrubbing filter system TL003. The TL003 filter system is qualified for severe accidents and is highly efficient for fission product releases, except noble gases. Chemicals are added to the water content of the filter to enhance the scrubbing. No electric supply is needed to operate the filtered venting system as the valves can be opened manually from the outside.

The licensee currently has three accident management measures that are applicable (numbering from Licensee Report):

- control the containment conditions: Severe Accident Management Guideline SAG-6 (see Annex 6.1 of Licensee Report);
- reduce the containment pressure: Function Restoration Procedure FHP-Z-1 (see Annex 6.7 of Licensee Report);
- reduce the containment pressure: Severe Accident Management Guideline SCG-2 (see Annex 6.1 of Licensee Report).

6.3.4 Prevention of re-criticality

To prevent recriticality boron can be injected in the primary system by use of the operational boron suppletion system in combination of the volume control system, or by using the safety injection system. Licensee has a procedure that gives guidance to restoring sub-criticality.

6.3.5 Prevention of basemat melt through

There are several international research programmes focusing on debris cooling strategies and basemat melt-through strategies. The lessons learnt from these programs are assessed in the current periodic safety review[37] (PSR).

In the Borssele NPP, the KCB there is no core catcher or similar design arrangement for retention of corium in the vessel. There are two complementary strategies to prevent basemat melt-through. One focuses on in-vessel retention of the core and the other on in-containment retention after vessel failure.

In-vessel corium retention is an accident management strategy in which the reactor vessel is cooled preventing its failure and successive relocation of the core in the containment. In the KCB the possible means to accomplish this are:

- Corium cooling by restoring primary circuit reflooding – this is part of the current SAM strategy;
- External vessel cooling, a potential approach in which the gap between the vessel and the biological shield (concrete wall) is used to bring water and steam flows for cooling. In the KCB the gap is very narrow and this option therefore looks less promising.

In the unlikely event that the vessel fails, corium needs to be cooled in the cavity, which is more challenging. Cavity flooding is the strategy most adopted in SAMGs of utilities world-wide. It is being investigated in the current PSR. In the KCB after vessel failure the cavity can be flooded via the reactor system. In one of the current SAMGs of KCB, this option is addressed. There is also a SAMG which aids to control the sump water level (i.e. level outside the cavity) to ensure cooling of any debris that might escape from the cavity. This action will aid in scrubbing of fission product releases from ex-vessel debris outside of the cavity.

Additional supporting SAMGs exist for injecting of water into the containment, control of containment conditions and reducing containment pressure.

A specific time line for the delay between reactor shutdown and core meltdown can not be given because of the many scenarios that are possible. If core cooling keeps failing and cannot be restored, this is to be considered as a cliff edge.

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

The containment isolation function requires closed containment isolation valves. Most containment isolation valves will be closed automatically at an early stage of an accident, initiated by the containment isolation signal. As the containment isolation valves are battery powered these valves can also be closed in case of SBO. A proportion of the containment isolation valves will be intentionally kept open for cooling purposes during an accident. These valves can also be closed in case all power is lost. There is no need for compressed air to close the containment isolation valves. The containment isolation valves that are opened by the use of compressed air are spring closed with a battery powered actuation valve.

For operating of the containment filtered venting system, the isolation valves of this system to the environment need to be opened. These valves can also be operated manually from outside the buildings without electrical power.

If all electric AC power is lost the following options are available:

- increase the steam concentration in the containment by opening certain relief valves and reducing hydrogen concentration;
- stop heat removal from the containment by stopping the following systems, that will automatically stop because they are AC powered:

[37] The current PSR in Borssele NPP is the '10EVA13', the current 10-yearly safety evaluation.

  - containment spray;
  - air coolers TL030-032;
  - biological barrier coolers TM001/002;
  - annulus coolers;
  - various other coolers.
- Active opening of the relief hatches in the containment building between the installation area and the operations area to achieve a more even distribution of hydrogen in the containment. This is possible without AC power; hatches are opened with compressed air from little containers in the containment and electrical power for actuation valves comes from batteries.

There are other accident management measures and associated procedures that require AC power. They are described in the following procedures (numbering from Licensee Report):

- reducing the containment pressure: Function Restoration Procedure FHP-Z-1 (see Annex 6.7 of Licensee Report);
- injection into the containment: Severe Accident Management Guideline SAG-4 (see Annex 6.1 of Licensee Report);
- controlling the containment conditions: Severe Accident Management Guideline SAG-6 (see Annex 6.1 of Licensee Report);
- reducing the containment pressure: Severe Accident Management Guideline SCG-2 (see Annex 6.1 of Licensee Report);
- controlling hydrogen flammability: Severe Accident Management Guideline SCG-3 (see Annex 6.1 of Licensee Report).

Alternative power sources have been addressed in chapter 5 of this National Report.

6.3.7 Measuring and control instrumentation needed for protecting containment integrity

The plant features diverse instruments for measuring parameters like containment temperature and pressure, and in various ranges. The readings of the instruments is available in the main control room, the plant process computer, the emergency control room. The readings can trigger alarm annunciations. There are two instruments for measuring hydrogen concentration[38] in the containment. In addition radiation levels[39] are monitored.

6.3.8 Capability for severe accident management in case of simultaneous core melt/fuel damage accidents at different units on the same site

Not applicable, the Borssele site features a single unit NPP.

6.3.9 Conclusion on the adequacy of severe accident management systems for protection of containment integrity

The NPP is equipped with accident management systems to protect the containment integrity. To be mentioned are the passive autocatalytic recombiners (PARs) and the filtred venting system. Both aid to protect against high hydrogen concentrations and over-pressurization of the containment. Severe accident management guidelines (SAMGs) exist and are in place for all operational states. The Licensee Report might be improved by differentiating more clearly the actions to be taken for the different plant states.

[38] Reliable H_2 measurements are possible in the range 0 to 10%.
[39] Dose rates in the range 10^{-2} to 10^{5} $Sv.h^{-1}$ can be measured.

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

Measures proposed by licensee EPZ

- A potential additional measure in the phase after the occurrence of fuel damage might be filling the outside of the reactor vessel ('Verlorene Schalung') and cavity with water in order to cool the outside of the reactor vessel. Note that there is considerable doubt about the effectiveness of water to cool ex-vessel debris in such a small reactor cavity.
- In previous periodic safety reviews an extensive set of formal analyses has been performed to address the threats of hydrogen to the containment. In 10EVA13 these studies will be reviewed and where necessary renewed and extended.

Regulatory position statements

- Licensee in its Licensee Report has identified options for further improvement of its severe accident management. In principle many options can be endorsed by the regulatory body. However before implementation their effectiveness needs to be assessed;
- It is recommended to re-assess hydrogen combustion vulnerability and hydrogen management, including possible accumulation of this gas in other buildings than the containment;
- The Licensee Report presents a SAM-strategy which uses existing resources and equipment based on its availability, including those designed for the control of design base accidents or non-nuclear class components. This is a common and acceptable approach for accident management past the design basis. However further assessments are recommended to establish the validity of the assumptions made regarding the associated SSCs;
- It is recommended to consider accident management measures after loss of containment, addressing issues like repair and possible mitigating actions from outside the containment.

6.4 Accident management measures to restrict the radioactive releases

6.4.1 Radioactive releases after loss of containment integrity

On occurrence of loss of containment integrity, a required measure is to reduce the containment pressure. The following design provisions are available: (1) containment spray, (2) air coolers inside the containment, (3) containment recirculation filter system and (4) the containment filtered venting system.

Operational provisions include various SAMGs. They are described in the following procedures (numbering from Licensee Report):

- reducing the fission product releases: Severe Accident Management Guideline SAG-5 ;
- controlling the containment conditions: Severe Accident Management Guideline SAG-6
- mitigating fission product releases: Severe Challenge Guideline SCG-1 ;
- injection into the containment: Severe Accident Management Guideline SAG-4.

6.4.2 Accident management after uncovering of the top of fuel in the fuel pool

Hydrogen management

The spent fuel pool (SFP) is located in the containment. The passive autocatalytic recombiners (PARs) in the containment will recombine hydrogen released by reactions[40] in the SFP.

Operational provisions include various SAMGs. They are described in the following procedures (numbering from Licensee Report):

- active opening of relief hatches between the installation area and the operations area in the containment. This will improve/start the natural circulation between the installation area and the operations area in order to reduce the probability of high local hydrogen concentrations;
- controlling the containment conditions: Severe Accident Management Guideline SAG-6 ;
- controlling hydrogen flammability: Severe Accident Management Guideline SCG-3.

Providing adequate shielding against radiation

When the water level in the spent fuel pool (SFP) lowers, the radiation levels (dose rates) will increase in this area. Refilling the SFP is a remedy to lower the dose rates. The following accident management measures apply:

- Filling of the spent fuel pool with water from the tanks of the Safety injection & residual heat removal system. An instruction (BTG-01) is available. The TJ tanks are located outside the containment, in the auxiliary building 03;
- A connection can be made between the demineralised water system TN to the suction side of the pool cooling pump of the Spent fuel pool cooling system TG by use of a flexible hose. In this way water can be injected into the fuel pool. This accident management measure is currently not mentioned in a separate procedure;
- Furthermore other connections to the TG system and the spent fuel pool can be made by use of a flexible hose, e.g. via the Radioactive waste water system TR. This accident management measure is currently not mentioned in a separate procedure;
- Injection of water from the TJ tanks by the containment spray pumps. This accident management measure is mentioned in Severe Accident Management Guideline SAG-4.

After loss of cooling of the SFP the water will evaporate and the water level will lower gradually. It will take several days before the top of the fuel assemblies will surface, refer to chapter 5 for more information. Several alternatives exist to restore pool cooling, amongst others:

- Start a cooling chain via the normal spent fuel cooling system TG, component cooling system TF, conventional emergency cooling water system VF, and supply this chain with water – depending on availability – from:
    - low pressure fire extinguishing system UJ;
    - fire truck taking suction form fire fighting pond of coal fired plant CCB or from river Westerschelde (unlimited resource);
- Start a cooling chain via normal spent fuel cooling system TG, using the reserve heat exchanger 'TG080', and connect to Backup cooling water system VE, and supply this chain with water – depending on availability – from:
    - deep water wells connected to VE;
    - low pressure fire extinguishing system UJ;
    - fire truck taking suction form fire fighting pond of coal fired plant CCB or from river Westerschelde (unlimited resource);

[40] Such reactions may include a zirconium-water reaction.

More details can be found in section 5.2.2.2 of this National Report.

Restricting releases after severe damage of spent fuel in the spent fuel pool

The spent fuel pool is located in the containment. The plant has a containment venting line with a wet scrubbing filter system. The filter system is qualified for severe accidents and is highly efficient for fission product releases, except noble gases. Chemicals are added to the water content of this filter to enhance the scrubbing of iodine.

The containment spray system can be used to wash out airborne material and iodine from the containment's atmosphere. This will also give a reduction of the amount of radioactive material that is released in case of containment leakage.

Instrumentation needed to monitor the spent fuel state and to manage the accident

The level and temperature of the spent fuel pool are being measured. The radiation level near the spent fuel pool is also measured. These instruments are qualified for (severe) accident conditions and readings are available in the main control room and the emergency control room.

Availability and habitability of the control room

Licensee has analysed dose rates in various locations as a function of time after core melt with a major occurrence of fuel damage, conservatively set at 100%. It can be established that areas essential to manage the situation are accessible and habitable in this situation. An exception is a scenario with flooding of the site; in this case the ACC bunker for the ERO may not be accessible.

More details can be found in section 6.1.3 of this National Report.

6.4.3 Conclusion on the adequacy of measures to restrict the radioactive releases

Regulatory position statements

- The NPP features various design provisions and procedures that aid to restrict radioactive releases from the containment. The provisions satisfy the licence of the plant. Notable provisions are autocatalytic recombiners (not needing electric power) and the filtered venting system (also independent from AC power). SAMGs are in place to guide the staff to make the necessary decisions under accident conditions.
- The Licensee Report has presented for a certain core-melt scenario an assessment of the dose rates that may be encountered in various areas of the plant. It is recommended to further assess the expected dose rates for various scenarios.
- Contaminated water can arise from various accident management strategies. It is recommended to study procedures that can be envisaged for handling of large amounts of radioactively contaminated water.

7. General conclusions

The operator of the KCB has submitted a Licensee Report to the Regulatory Body, that addresses all topics prescribed in the ENSREG guidelines for the 'stress test' and meets the prescribed format. The description of the plant, its design basis and the analyses performed in general appear comprehensive. Overall it can be concluded the report is of good quality and is realistic in its assessments of the consequences for the nuclear power plant of the extreme external challenges postulated by ENSREG.

Based on decades of regulatory oversight compliance of the nuclear power plant with its licence base has been established. The regulatory review of the Licensee Report has not provided evidence for a different position.

Considering the very tight planning it is understood that in the current Licensee Report, some issues have been addressed to a limited degree and may need further assessment and successive reporting. During the regulatory review of its report, the licensee already provided a tentative planning of its intended measures and topics for further research. Implementation of these measures and associated planning will be further discussed with the regulatory body.

Some main remarks from the regulatory review are listed here in summary; for more details refer to section 7.2 and 7.3 of this chapter.

- The regulatory body recognizes that in the Licensee Report, for the assessment of ENSREG-postulated scenarios, licensee has given credit to various structures, systems and components (SSCs) that are not designed, classified or tested for their purpose in severe accident management. This is a common and acceptable approach for accident management past the design basis and for the purpose of the stress test this is acceptable too. However further assessments are recommended to establish the validity of the assumptions made regarding the associated SSCs.
- The licensee has proposed measures to further increase robustness of its plant and topics for further study. Most of these proposals in principle can be endorsed by the regulatory body. Before implementation, their effectiveness needs to be assessed;
- Regarding severe accident management (SAM) measures some aspects need further addressing like long-term measures. Furthermore the effectiveness of some procedures may need to be established by conducting tests. Training of long-term SAM measures should improve the reliability of existing procedures under crisis conditions;
- The description of 'cliff edges'[41] in the Licensee Report for most scenarios has been elaborated to a limited detail;
- The regulatory body has the opinion that the impact of floods with a very long return period (e.g. ten thousand, one hundred thousand or one million years) is not known in much detail yet and that further assessments are necessary.

7.1 Key provisions enhancing robustness (already implemented)

Continuous improvement processes

The Borssele NPP features 10-yearly safety evaluations, the Periodic Safety Reviews (PSR). Based on the outcome of these evaluations, modification projects are initiated if necessary. This is part of a continuous improvement process. Borssele has implemented a full scope 'living'

[41] Cliff edge effect is the effect of an abrupt transition from one status to another when exceeding a certain limit or applying a small change to a parameter.

Probabilistic Safety Assessment (PSA) which is updated yearly. Plant modifications and updated failure data are included in the PSA model.

The PSRs have contributed to established modifications like the ones listed below.

<u>1986:</u>

- Introduction of the 'bunker concept', which is a 'two-train'-bunker protected against high flood level and earthquakes, containing:
  - two 100% trains of diverse systems for primary make-up and emergency feedwater supply.
  - 2x 100% diverse Station Blackout Diesels.

<u>1997</u>

- Introduction of a Reserve Ultimate Heat Sink, by eight 17% deep wells as part of the 'bunker concept';
- Introduction of a reserve Residual Heat Removal System as part of the 'bunker concept';
- Introduction of a Spent Fuel Pond cooling system pump in the 'bunker concept';
- Expansion of the (primary) emergency power system to 3x 100% EDG's in different locations around the plant;
- Introduction of the leak-before-break-concept for the primary system, main steam lines up to the main steam isolation valves, and the feedwater lines up to the isolation valves;
- Introduction of 30 minutes grace time for all design base events, 10 hours autarky time and 24 hours autonomy time for design base external events;
- Filtered containment venting;
- Passive Autocatalytic Recombiners for hydrogen control (only needed for core melt scenarios);
- Separation (electrically, hydraulically and geographically) of safety systems to realize independency between trains;
- Plant specific full scope simulator;
- Adoption of the Westinghouse Owners Group Generic Severe Accident Management Guidelines (in operation since 2000).

<u>2006</u>

- Possibility to supply the bunkered systems by the primary (large capacity) EDG's;
- Extension of the autonomy time to 72 hours for design base external events.
- Protection against hazardous gasses from Westerschelde shipping accidents (toxic, combustible);
- Addition of a 2nd SFP cooling system pump in the 'bunker concept';
- Addition of a 2nd pump in the reserve Residual Heat Removal System as part of the 'bunker concept';
- Increasing the flooding margin of the bunkered systems by raising the SBO EDG's air intakes;
- Introduction of a crash tender for protection against large kerosene fires;
- Introduction of external connections for alternative feed water supply by mobile pumps;
- Automatic actions to counter loss of coolant incidents in mid-loop conditions;
- Expansion of the SAMGs to shut-down conditions.

The licensee is in the process of developing a set of Extensive Damage Mitigation Guidelines (EDMGs). This is a model developed by industry, specifically in the USA. The regulatory body endorses the idea of developing the EDMGs but can not review it yet since its development by utility EPZ has not been completed yet.

Based on SOER (Significant Operating Experience Report) 2011-2, issued by WANO (World Association of Nuclear Operators) the licensee has performed, shortly after the Fukushima events, an evaluation based on the SOER 2011-2 guideline.

The results of this evaluation has been discusses with the regulator and have been made public. The evaluation shows no discrepancies related to the design bases of the power plant. However some potential improvements have been identified. These improvements can be categorized in three groups:

a. Improvement of the internal test and surveillance programs;

b. Improvement of the availability of existing safety provisions in the case of flooding or a seismic event;

c. Improvements in the field of accident management after beyond design bases accident situations.

The actions to implement the improvements have been discussed with the regulatory body that agreed with the implementation plan submitted by the licensee.

Safety margins

Safety margins for some external hazards are provided below.

- Earthquakes
    - The lowest seismic capacity of all considered structures, systems and components (SSCs) has been assessed to be 0.15 g, based on calculations and engineering judgement. The information provided seems plausible.
    - A full seismic margin assessment has not been performed yet, but the licensee proposes further studies, refer to section 7.3 for more information.
- Flooding
    - Safety margins on top of the design basis for flooding exist. The DBF for the NPP is 7.3 m + NAP. Margins allow a height of at least 8.55 m +NAP.
    - The information provided by licensee seems plausible, based on the models used. Other flooding models exist for coastal locations and comparison with the results of these is useful. Refer to section 7.3 for the full regulatory position on this issue.
- Extreme weather
    - Licensee evaluated all relevant (combinations of) credible weather conditions and provided safety margins. Some of these are listed below with some comments.
    - *Water temperature*: the original design base considers a maximum cooling water inlet temperature of 2.6 ^{o}C but the margins allow inlet temperatures of at least 25 ^{o}C. The design base does not specify a minimum temperature. Lowest temperature measured was –1.1 ^{o}C.
    - *Air temperature*: the design base does not specify minimum and maximum values. However diesel fuel storage tanks allow a minimum temperature of -18 ^{o}C before degradation of fuel can occur, which is reasonable considering the local climate.
    - *Win*d: For some safety relevant buildings the design basis requires resistance against at least 0.1 bar, where their design allows them to take 0.36 down to 0.3 bar. For others it is noted they are able to resist wind speeds of 12 Bft.
    - *Rainfall*: Extreme rainfall in combination with blocked drainage pipes and no intervention of plant staff can result in accumulation of water on some roofs. This will not cause loss of safety relevant SSCs if it does not last longer than 48 hours. A special case is the accumulation of water because of fire-fighting activities in combination with blocked drainage pipes, refer to section 7.3.
    - *Snowfall*: All buildings of the NPP can withstand all credible consequences of snowfall, considering the local climate. A Dutch norm requires resistance to a load of 0.7 $kN.m^{-2}$ whereas the resistance of the NPP buildings varies from 1 (turbine building) to 17.7 $kN.m^{-2}$ (dome).

  - *Lightning*: The plant satisfies the relevant norms on lightning protection[42] but no margins for lightning have been given.
- LOOP/SBO and LUHS
  - There are margins and alternatives for loss of electrical power and emergency power grids. In addition the plant has various alternatives for the main ultimate heat sink. Notable is the unlimited supply of fresh water from the deep water wells.

7.2 Safety issues

Potential cliff edge effects

Several cliff edge effects have been identified. Exceeding certain values or occurrence of certain incidents may introduce an abrupt transition of the plant from one state into an other. Such a transition may or may not have major consequences for the safety of the plant and its environment. Identification of cliff edge effects will help to learn lessons about improving robustness for beyond design basis events.

A number of cliff edges has been identified by the licensee in its Licensee Report. The cliff edges indentified to date can be categorized as follows:

- Cliff edges related to the availability of staff;
- Cliff edges related to the failure of systems, structures and components;
- Cliff edges related to the limited availability of diesel fuel;
- Cliff edges related to the limited availability of water for cooling purposes;

Most of the cliff edges are relatively obvious given the nature of a nuclear power plant. Nevertheless additional studies and analyses will be done by the licensee in order to develop measures to further improve the robustness of the installations with respect to the cliff edges recognized.

Shortfalls

Based on decades of regulatory oversight, compliance of the nuclear power plant with its licence base has been established. The regulatory review of the Licensee Report has not provided evidence for a different position. Therefore there are no shortfalls to report.

7.3 Potential safety improvements and further work forecasted

The licensee in its Licensee Report has proposed various measures, alterations and extensions of its procedures and topics for further study. In principle the regulatory body endorses most improvements proposed. However, before implementation, their effectiveness needs to be assessed.

The licensee in its proposals for further studies suggests to implement some of them in the PSR (10EVA13) that is being undertaken. The regulatory body will consider if all or some of these studies should be undertaken as part of the current PSR, and/or if a separate track is needed to perform these studies.

Below licensee's full set of tabled and numbered proposals is reproduced in the subsections 7.3.1 (Measures), 7.3.2 (Procedures) and 7.3.3 (Studies). Each table is followed by regulatory position statements related to a selection of some important issues. Where applicable in the position statements, reference is made to licensee's numbered proposals (format: Mx, Px, and Sx).

[42] KTA 2206 and NEN 1014

7.3.1 Measures

Table 7-1 Measures proposed by licensee EPZ

No.	Measure proposed by licensee EPZ
M1	Emergency Response Centre facilities that could give shelter to the emergency response organisation after all foreseeable hazards would increase the options of the emergency response organisation.
M2	Storage facilities for portable equipment, tools and materials needed by the emergency response organisation that are accessible after all foreseeable hazards would increase the possibilities of the emergency response organisation.
M3	A possibility for refilling the spent fuel pool without entering the containment would increase the margin to fuel damage in certain adverse containment conditions.
M4	Additional possibilities for refilling the spent fuel pool would increase the number of success paths and therefore increase the margin to fuel damage in case of prolonged loss of spent fuel pool cooling.
M5	Reduction of the time necessary to connect the mobile diesel generator to Emergency Grid 2 to 2 hours would increase the margin in case of loss of all AC power supplies including the SBO generators.
M6	Establishing the ability to transfer diesel fuel from storage tanks of inactive diesels to active diesel generators would increase the margin in case of loss of off-site power.
M7	Establishing independent voice and data communication under adverse conditions, both on-site and off-site, would strengthen the emergency response organisation.
M8	Ensuring the availability of fire annunciation and fixed fire suppression systems in vital areas after seismic events would improve fire fighting capabilities and accident management measures that require transport of water for cooling/suppression.
M9	By increasing the autarky-time beyond 10 h the robustness of the plant in a general sense would be increased.
M10	Ensuring the availability of the containment venting system TL003 after seismic events would increase the margin in case of seismic events.
M11	Wave protection beneath the entrances to the bunkered back-up injection- and feedwater systems and to the bunkered emergency control room would mitigate the sensitivity to large waves combined with extreme high water and would make the plant fully independent from the dike.

Regulatory position statements on measures

- The improvement of the accessibility under extreme conditions of rooms, warehouses and others which house (reserve) equipment needed for severe accident management (M2) is important. It should also take into account a re-evaluation of dose rates in these areas for representative scenario's.
- Improvement of the possibilities to sustain cooling of the spent fuel pool (SFP) under all foreseeable conditions and replenish its water (M3). Any implementation should keep dose to workers ALARA. Some possibilities have been mentioned in the Licensee Report but they need further study before implementation.
- The fire fighting systems in buildings 01/02 (dome) and 35 (backup control room) are not designed for operability after occurrence of the design base earthquake (DBE). To enhance their reliability after a DBE they should be qualified. However, any enhancement should be based on the results of the proposed advanced seismic analysis (refer to section 7.3.3). This position is linked to licensee's proposal S3 but also (for implementation) to its measure M8.

- Technical and organisational improvement of availability under earthquake conditions of systems for containment filtered venting and fire fighting (M8, M10). However, any enhancement should be based on the results of the proposed advanced seismic analysis (refer to section 7.3.3).
- Increasing the autarky time beyond 10 hours (M9). The autarky time is the time that the plant is kept in a controlled, stable state without the need for human interaction[43]. This measure requires further study to define its proper implementation.

7.3.2 Procedures

Table 7-2 Procedures (or their development) proposed by licensee EPZ

No.	Procedure (or its development) proposed by EPZ
P1	Develop a set of Extensive Damage Management Guides (EDMG) and implement a training program. Below are examples of the issues to be addressed:
	• Description of the alternative ways to replenish the fuel storage pool • Injection of fire water directly into the fuel storage pool by a flexible hose • Cooling the fuel storage pool by TG080/VE supplemented by UJ • Connection of TN to the suction side of the fuel storage pool cooling pumps • Procedure for spent fuel pool cooling (over spilling, make up) • Flexible hose connections to the TG system and the spent fuel pool • Procedures to staff the Emergency Control Room • Procedure for direct injection of VE by UJ • Use of autonomous mobile pumps • Possible leak repair methods for larger pool leakage • Procedure to transport own personnel to the site • Procedure for the employment of personnel for long term staffing • Connecting CCB/NS1 • Uncoupling of lower rails in time in case of flooding • Alternative supplies for UJ
P2	By training of the procedure ensure that during mid-loop operation, the actions for water supply that are needed in case of loss of all AC power supply, are performed in a timely manner.
P3	Develop check-lists for plant walk-downs and the necessary actions after various levels of the foreseeable hazards

Regulatory position statements on (development of) procedures

- Regarding severe accident management (SAM) measures, some aspects need further addressing like long-term measures. Furthermore the effectiveness of some procedures may need to be established by conducting tests. It is understood that internal rules exist for the chain of command under crisis conditions. However training of long-term SAM measures should improve the reliability of existing procedures under these conditions.
- The licensee is in the process of developing a set of Extensive Damage Mitigation Guidelines, the EDMGs (P1). This is a model developed by industry, specifically in the

[43] The autarky time allows bridging the time between an external event that disables the crew, and the arrival of a replacement crew. Autarky time is an entirely different quantity than the autonomy time. The autonomy time of the KCB is much longer than its autarky time. The autonomy time is the time that the plant can be kept in a controlled, stable state without the need for off-site supply of equipment or consumables. It allows bridging the time between an external event with extensive infrastructural damage, and the arrival of replacement equipment, emergency equipment or consumables (diesel, water).

USA. The regulatory body endorses the idea of developing the EDMGs, but can not review it yet since its development by utility EPZ has not been completed yet.

- It is recommended that the systems, structures and components (SSCs) important to handle severe accidents will be tested functionally (as far as they are applied past their suitable qualification) and their handling be trained according to procedures that need to be established.
- A set of clear criteria needs to be established as a basis for deciding when to switch the turbine oil pump off to increase the battery time. Disabling this pump will damage the turbine.

7.3.3 Topics for further study

Table 7-3 Topics for further study proposed by licensee EPZ

No.	Study proposed by EPZ
S1	A reserve spent fuel pool cooling system that is independent of power supply from the emergency grids could expand accident management possibilities. In 10EVA13 this will be investigated.
S2	In 10EVA13 measures will be investigated to further increase the safety margins in case of flooding.
S3	Uncertainty of the seismic margins can be reduced by a Seismic Margin Assessment (SMA) or a Seismic-Probabilistic Safety Assessment (Seismic-PSA). In 10EVA13 either a seismic-PSA will be developed and/or an SMA will be conducted and the measures will be investigated to further increase the safety margins in case of earthquake
S4	In 10EVA13 the possibilities to strengthen the off-site power-supply will be investigated. This could implicitly increase the margins in case of loss-of-offsite power as it would decrease the dependency on the SBO generators.
S5	More extensive use of steam for powering an emergency feed water pump and for example an emergency AC generator could increase the robustness in case of loss of all AC power supplies including the SBO generators.
S6	Uncertainty in the margins with respect to airplane crash could be reduced by performing a more extensive study of the impact on the safety functions of different airplane crashes.
S7	In previous periodic safety reviews an extensive set of formal analyses has been performed to address the threats of hydrogen to the containment. In 10EVA13 these studies will be reviewed and where necessary renewed and extended.

Regulatory position statements on topics for further study

- Flooding (S2): the regulatory body has the opinion that the impact of floods with a very long return period (e.g. ten thousand, one hundred thousand or one million years) is not known in much detail yet and further assessment is recommended. Several governmental bodies are involved in the assessment of the adequacy of the protection of the Netherlands against flood risks. Models have been developed and continuously are being improved to aid this assessment. It is recommended that a reassessment tailored to the needs of the Borssele site be undertaken considering: (1) given a specified return period the maximum challenge a flood will pose to the NPP and its dykes, (2) the various failure mechanisms of the dykes, (3) the impact of the maximum challenge by floods on the safety of the NPP, and (4) the various options to protect the plant against this challenge like improving dykes and/or adding other engineered structures.
- Earthquakes (S3): the Licensee Report states that the licensee plans to perform a seismic PSA or a Seismic Margin Assessment, a SMA. The regulatory body endorses this proposal.

It is known that the Netherlands Royal Meteorological Institute (KNMI) will contribute data and knowledge to this project. In the seismic study attention should be given to among others characterization of the subsurface of the site and possible seismic impact of foreseeable future mining activities in the neighbourhood.

- Extreme weather: heavy rain does not pose extreme challenges to the plant. A special case is the accumulation of water resulting from fire-fighting activities if drain pipes are blocked. The possible consequences of this need to be studied. Further recommended topics for additional study are: the minimum depth of underground piping required for proper protection against freezing, possibility to operate diesel generators at extremely low temperatures and the potential effect of accumulation of wind-transported snow on roofs.
- Severe accident management (SAM): the Licensee Report presents a SAM-strategy which uses existing resources and equipment based on its availability, including those designed for the control of design base accidents or non-nuclear class components. This is a common and acceptable approach for accident management past the design basis and for the purpose of the stress test this is acceptable too. However further assessments are recommended to establish the validity of the assumptions made regarding the associated SSCs. It is recommended to evaluate whether an upgrade of the equipment and/or instrumentation especially tailored to SAM needs is necessary and/or feasible. The proper qualification and classification of the SSCs needed for SAM should be part of the study Related topic for study are:
    - The various alternative power sources should be re-evaluated. As an alternative to emergency grid 2 (NS2) licensee mentions the emergency diesel generator (EDG) of the coal-fired plant, however this is not a nuclear class component and not tested for the purpose. The mobile EDG (EY080) currently needs off-site support, which can not relied upon in all crisis situations. Also procurement arrangements for externally supplied EDGs are mentioned, which still have to be concluded. Some crisis conditions may make it impossible to transport an EDG to the site. Alternate and independent means to recharge batteries should also be part of the study.
    - Connection equipment and connection points for power sources, fuel resources etc. are available. Their ease of use and availability for all foreseeable circumstances should be analysed;
    - In the Licensee Report, the fire fighting pond of the nearby coal-fired unit (CCB) is mentioned as an alternative resource (1,600 m^3) for cooling water. Investigation into the actual usefulness and availability of this resource is recommended.

Appendix A KFD review of licensee's assessment

In the appendix, as a courtesy, the document drafted by the Kernfysische Dienst (KFD). The KFD's findings have been included in the body of the National report.

Inspectorate of Housing, Spatial
Planning and the Environment
*Ministry of Infrastructure and the
Environment*

Post Fukushima Stress Test of the EPZ Nuclear Power Plant in the Netherlands

KFD review of the licensee’s assessment

Colophon

Inspectorate of Housing, Planning and the Environment
Department for Nuclear Safety Security Safeguards (VI-KFD)

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Samenvatting

Het ongeval in Fukushima in maart 2011 heeft op Europees niveau geleid tot de beslissing om de bestaande kerncentrales in Europa te onderwerpen aan een stresstest. Een stresstest is een onderzoek waarin de veiligheidsmarges van de centrale opnieuw worden geëvalueerd; met name voor situaties als de centrale aan extreme (weers-) omstandigheden wordt blootgesteld.
Het onderzoek moet voldoen aan de vereisten en kwaliteitscriteria zoals die door ENSREG[1] zijn vastgesteld.(ref. Declaration of ENSREG 13 maart 2011).
Het uitvoeren van deze stresstest in Nederland is verplicht gesteld door de overheid, zie hiervoor de brief ETM/ED/11074538 van de minister van Economische Zaken, Landbouw en Innovatie gedateerd 01 juni 2011. In deze brief wordt naast de ENSREG stresstest ook gevraagd de invloed van andere externe factoren te onderzoeken die kunnen leiden tot het verlies van meerdere veiligheidsfuncties. Hierbij moet rekening worden gehouden met "man made", waaronder moedwillige, verstoringen
EPZ[2] heeft voor haar kerncentrale in Borssele een dergelijke stresstest uitgevoerd. De resultaten van de test zijn opgenomen in een document, getiteld "Complementary Safety margin Assessment", 31 oktober 2011.
De Kernfysische Dienst (KFD) als toezichthouder heeft het stresstestrapport van EPZ beoordeeld en komt tot de volgende bevindingen:

Het stresstestrapport

- De uitgevoerde beoordeling geeft geen indicaties dat de installatie en organisatie niet aan de eisen van de huidige vergunning voldoen.
- Het stresstestrapport is, met inachtneming van de verder in dit rapport vermelde opmerkingen en gelet op de strakke ENSREG tijdsplanning, in z'n algemeenheid van voldoende kwaliteit. Het rapport voldoet op hoofdlijnen aan de eisen van ENSREG en de door de minister van EL&I gestelde aanvullende eisen.
- Op het punt van severe accident maatregelen ontbreken een aantal belangrijke aspecten, waaronder lange termijn maatregelen en accident management maatregelen na het verlies van de integriteit van het containment.
- Het rapport geeft een vrij volledig en getrouw beeld van de huidige technische en organisatorische situatie in de centrale. Het rapport geeft een realistisch beeld van de omstandigheden waaraan de centrale in extreme situaties zou kunnen worden blootgesteld.
- Het rapport bevat een brede analyse van de wijze waarop de bedrijfsonderdelen van de centrale op die extreme omstandigheden reageren of daartegen bestand zijn.

Hierbij zijn de volgende algemene opmerkingen te maken, naast de specifieke opmerkingen per ENSREG vereiste die in hoofdstuk 2 zijn opgenomen:
- De effecten van een mogelijke brand die ontstaat als gevolg van andere extreme situaties zijn onvoldoende belicht.
- De beschrijving van 'cliff edges' (kleine verandering met grote, veelal onomkeerbare, effecten) is voor de meeste scenario's summier uitgevoerd.

[1] ENSREG: European Nuclear Safety Regulators
[2] EPZ: Elektriciteits Produktiemaatschappij Zuid-Nederland

- Door het hele rapport heen is geen duidelijk verband zichtbaar tussen de voorgestelde maatregelen en de geïdentificeerde cliff edges of geconstateerde marges.
- De KFD acht het nodig dat de analyse van de aanpak van zware ongevallen wordt 'verdiept' door ook de effectiviteit van de gekozen aanpak in de praktijk te toetsen.
- De door EPZ voorgestelde maatregelen lijken bij eerste lezing de veiligheid te vergroten. Nader onderzoek en analyse van deze maatregelen door EPZ is echter nodig om dit zeker te stellen. Hiernaast is vastgesteld dat verder onderzoek door EPZ op een aantal punten nodig is.

De robuustheid van de centrale in geval van extreme (weers-)omstandigheden

- In het rapport is aannemelijk beschreven dat de centrale van EPZ beschikt over veiligheidsmarges ten opzichte van de technische en organisatorische eisen waaraan de centrale op dit moment wettelijk moet voldoen. Toch acht de KFD, in het licht van Fukushima, kordate aanpak nodig bij het realiseren van enkele gebleken verbeterpunten door middel van het treffen van maatregelen en het uitvoeren van nadere studies. Hierbij wordt in het bijzonder gedoeld op
  - het vergroten van de mogelijkheid om in geïsoleerde moeilijke omstandigheden langer en beter de veiligheid van de centrale te kunnen waarborgen. Daarmee wordt gedoeld op maatregelen voor het verlengen van de autarkietijd (de tijd dat de centrale zelfvoorzienend is) van apparatuur en het uitbreiden van de (zelf-) redzaamheid van personeel,
  - het aardbevingsbestendig maken van de voorzieningen op het gebied van brandbestrijding en het ventileren van het containment (t.b.v. de afvoer van mogelijk vrijkomend waterstof),
  - het vergroten van de mogelijkheden om het splijtstofopslagbassin onder alle omstandigheden te (blijven) kunnen koelen en van water te kunnen voorzien,
  - het inrichten van onder alle omstandigheden beschikbaar magazijn met voorzieningen ten behoeve van de aanpak van calamiteiten.

De voorstellen voor verbetering van de robuustheid van de centrale

- Het rapport bevat aannemelijke voornemens voor verbetering van de robuustheid van de centrale. Op een aantal punten heeft EPZ nadere onderzoeken geformuleerd. Hieruit zouden aangepaste of nieuwe maatregelen noodzakelijk kunnen blijken.
- De verbeteracties zijn (nog) niet van een prioritering en tijdsplanning voorzien. KFD acht het nodig dat EPZ het verbeterplan met deze aspecten ter beoordeling aan de KFD voorlegt.

Concrete verbeteracties dienen bij voorkeur gescheiden van de komende 10-jaarlijkse veiligheidsevaluatie te worden ingepland en uitgevoerd, dit in tegenstelling tot het voorstel van EPZ.

De Nederlandse stesstest ten opzicht van centrales in omliggende landen
Het stresstestrapport van EPZ kan zich meten met de kwaliteit van vergelijkbare rapporten van centrales in de ons omringende landen.
Een helder en afrekenbaar tijdschema voor de realisatie van de aanpassings-voorstellen ontbreekt vooralsnog. In dat opzicht zou EPZ zich kunnen spiegelen aan sommige nucleaire bedrijven in de ons omringende landen, die concreet in de tijd geplande aanpassingsvoorstellen doen

De stresstest is nog niet ten einde. Een volgende stap zijn de peer-reviews in Europees verband. De KFD zal de komende tijd de beoordeling, inclusief het oordeel over noodzaak en planning van onderzoeken en maatregelen, voortzetten mede gebruik makend van de bevindingen uit andere landen, bijgestelde normenkaders en de resultaten van de internationale peer-review.

Summary [3]

The Fukushima accident in March 2011 has led to the decision at European level that the existing nuclear power plants in Europe are to be subjected to a stress test. A stress test is an assessment in which the safety margins of the plant will be re-evaluated, in particular for situations where the plant is exposed to extreme (weather) conditions.
The assessment must meet the requirements and quality criteria as established by ENSREG. (Ref. ENSREG Declaration of May 13, 2011).
Conducting the stress test in the Netherlands was made compulsory by the government; see letter ETM/ED/11074538 of the Ministry of Economic Affairs, Agriculture and Innovation dated June 1, 2011. In this letter it was requested that, in addition to the ENSREG stress test, also the influence of other external factors that may lead to the loss of several safety features were to be assessed. "Man made" disturbances, including those instigated wilfully, should be taken into account.

EPZ has performed such a stress test for their Borssele nuclear power plant. The results of the test are reported in a document entitled "Complementary Safety Margin Assessment", October 31, 2011. The Department of Nuclear and Radiological Safety, Security and Safeguards (KFD) as the supervision authority reviewed the stress test report of EPZ and makes the following findings:

The stress test report

- The assessment carried out shows no indications that the installation and organization do not meet the current license requirements.
- Given the comments mentioned in this report and the tight timetable of ENSREG, the stress test report is in general of sufficient quality. On the main points the report meets the ENSREG requirements and the additional requirements set by the Ministry of EL&I.
- Regarding severe accident measures some important issues are missing, including accident management measures after loss of containment integrity and long-term measures.
- The report provides a fairly complete and accurate picture of current technical and organizational conditions in the plant. The report provides a realistic picture of the conditions to which the plant could be exposed in extreme situations.
- The report contains a broad analysis of the response and resistance of the system components of the plant on extreme conditions.

In addition to specific comments on the ENSREG requirements, given in Chapter 2, the following general comments have to be made:
- The effects of a potential fire as a result of other extreme situations have been addressed insufficiently.
- The description of 'cliff edges' (small change with large, often irreversible, effects) is worded briefly for most scenarios.
- Throughout the report no clear relation has been identified between measures that are envisaged and the cliff edges or margins observed.
- KFD sees it as imperative to deepen the analysis of handling of major accidents by practical tests.

[3] In cases where any differences occur between the English version and the original Dutch version, the Dutch version will prevail

- At first reading the measures proposed by EPZ seem to enhance safety. Further study and analysis of these measures is needed by EPZ to ensure this. Additionally, it is determined that further study on a number of aspects by EPZ is required.

The robustness of the plant in case of extreme (weather) conditions

- The report gives a plausible description that supports the conclusion that the EPZ power plant has safety margins with respect to the current technical and organizational requirements and the Dutch law. Nevertheless, in view of the Fukushima accident, KFD stresses that a resolute approach is taken in realizing improvements by means of implementing measures and carrying out further studies.
  This especially applies to
  - increasing and improving the potential of the plant to ensure the safety during isolated and more difficult conditions. This refers to measures to extend the autarky time (the time the plant is self-sufficient) of equipment and to expand the self-reliance of the staff,
  - proofing the facilities in the area of fire fighting and ventilation of the containment (for the disposal of potentially released hydrogen), against earthquakes,
  - increasing the number of options for the (continuous) cooling of the fuel storage pool and the supply of water to this pool,
  - setting up a warehouse with materials for dealing with emergencies and that will be available under all conditions.

The proposals for improving the robustness of the power plant

- The report contains plausible intentions for improving the robustness of the plant. On some points, further studies are specified by EPZ. It might turn out that these investigations result in the adjustment of already formulated measures or the determination of new measures.
- Up to now the improvements have not been prioritized and scheduled. KFD stresses the need for an improvement plan addressing these aspects and to be submitted to KFD by EPZ for review.

It is preferred that the planning and implementation of concrete improvement actions should be separated from the next 10-yearly safety evaluation, this in contrast to the proposal by EPZ.

The Dutch stesstest in comparison to plants in neighbouring countries

The stress test report of EPZ matches the quality of reports of similar plants in neighbouring countries.
A clear and accountable timetable for the implementation of the measures is lacking. In that respect, EPZ can take example from some nuclear operators in neighbouring countries that propose a specific schedule for the implementation of the measures.

The stress test is not over yet. The next step will be the peer reviews in the European context. Meanwhile KFD will continue the review, including the judgement on necessity and planning of studies and measures, using the findings from other countries, adjusted standards, frameworks and the results of the international peer-review.

1. Introduction

After the accident in March 2011 at the Fukushima Nuclear Power Plant in Japan, the European Council concluded that the safety of all EU nuclear plants should be reviewed in the light of Fukushima on the basis of a comprehensive and transparent risk assessment (‘stress test’). The European Nuclear Safety Regulators Group (ENSREG) developed the scope and modalities of this test, which should be used as a framework for the test. The Dutch government endorsed the European stress test specifications and asked the licensee to pay attention as well to other (probably man-made) effects that can have an adverse effect on safety systems.

The test is defined as a targeted reassessment of the safety margins of the nuclear power plants.
The reassessment can be seen as an evaluation consisting of three elements:

- Provisions taken in the design basis and plant conformance to its design requirements.
- Evaluation of the available margins in the design basis.
- 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 agreed methodology consists of one track on safety and another track on security. The first track focuses on extreme natural events like earthquake and flooding.
This track will also look into the consequences of loss of safety functions as a consequence of any other initiating event including man-made and other accidental impacts, for instance large disturbances from the electrical power grid and air plane crash. The second track, which deals with risks related to security threats, is not covered in this report.

EPZ, as the operator and licensee of the only nuclear power plant in the Netherlands, situated in Borssele (Zeeland), performed such an assessment. The results of the assessment are laid down in the report ‘Complementary Safety margin Assessment’ (October 31, 2011, final report). The report was sent to the Minister of Economic Affairs, Agriculture and Innovation (EL&I), who is responsible for all regulatory affairs under the Nuclear Energy Act.
The Department of Nuclear and Radiological Safety, Security and Safeguards (in Dutch Kernfysische Dienst, KFD), residing under the Inspectorate of the Ministry of Infrastructure and Environment as the independent nuclear supervision authority, reporting to the Minister of EL&I, reviewed the report of EPZ.

The KFD review has been performed by a group of KFD experts with specific knowledge in the areas of the stress test and people with broad experience in nuclear safety. The review has been carried out in a period of about three weeks with the assistance of several external national and international experts of governmental related organisations. National experts were consulted from specific institutes like KNMI (Royal National Meteorological Institute), Rijkswaterstaat (National Watermanagement Institute) and the Labor Inspectorate. KFD arranged for the input of additional international expertise by using experts from GRS (Gesellschaft für Anlagen – und Reaktorsicherheit) – a Technical Support Organisation on Nuclear Safety to the Ministry of Environment, Nature Conservation and Reactorsafety, Germany.

GRS prepared a first draft of its report in one week and sent this to KFD on 10th of November. The final report was provided on the 16th of November.
Furthermore, GRS did a plausibility and completeness check, and a comparison with the German stress test reports.
In its judgement KFD also took into account the available information about lessons learned from the Fukushima accident and partly also from public information that was provided by regulatory organisations or their technical safety organisations in other countries (e.g. FANC, IRSN). The scope of the stress test and the way to review the measures has be discussed with the other regulators

The timeframe set by the ENSREG stress test specification has implied that not all aspects could be analyzed thoroughly. KFD will continue with developing new insights from the Fukushima accident and using information from other nuclear power plants in other countries and the results of the peer reviews of all European stress test reports. These peer reviews will be performed in the first half of next year.

The measures as indicated by the licensee must therefore according to KFD be seen as a first initiative. Additional measures are necessary as this report shows. Although all measures given by the licensee seem to contribute in principle to the improvement of nuclear safety, more detailed analyses have to be performed in order to ensure that there are no adverse effects on nuclear safety.

One significant general measure showed by the licensee is the development of a set Extensive Damage Mitigation Guidelines (EDMG). This is a model developed by industry, specific in the US, largely put into effect after the September 11, 2001 events.
These EDMGs cannot be reviewed by KFD yet because they are under development by EPZ.

In the current report, Chapter 2 forms the main body of this report consisting of the overall summary of judgements on the main chapters of the EPZ assessment:

- General data of the site/plant
- Earthquake
- Flooding
- Extreme weather conditions
- Loss of electrical power and loss of ultimate heat sink
- Severe accident management
- Other extreme hazards

The detailed further actions that according to KFD should be taken by EPZ are collected in the ANNEX A. This annex is following the structure of the ENSREG requirements extended with the additional Dutch requirements.

2. Assessment of the ENSREG requirements

Introduction
This chapter should be read in conjunction with the EPZ report "Complementary Safety margin Assessment" (October 31, 2011, final report) and letter ETM/ED/11074538 of the Ministry of Economic Affairs, Agriculture and Innovation dated June 01, 2011. This information can be found on http://www.rijksoverheid.nl/onderwerpen/kernenergie/europese-stresstest-kerncentrales

2.1 Requirement 1: General data about the site/plant

ENSREG requires the licensee to provide general information about the plant and the site. This information should cover the general characteristics of the site and plant, the available systems to support the main safety functions and the scope and main results of the Probabilistic Safety Assessment (PSA). This chapter provides no results from analyses but forms the basis on which the following requirements have been build. Further details about the ENSREG requirements on general data about the site/plant can be found in appendix A.

In this review KFD has performed a check of the main issues of the supplied information related to requirement 1

KFD concludes that EPZ has provided an extensive overview related to this requirement that is beneficial for the assessment of the other requirements.
KFD has the following remarks:

- Before taking credit for the feeding of NS2 via fixed connection from the coal-fired plant (CCB), this supply-configuration should be demonstrated.
- EPZ mentions that in case of the deployment of MOX-fuel in the core the safety situation of the Borssele NPP is comparable to the current situation where only naturally enriched uranium is deployed. This does not automatically imply that the analyses presented in this report also cover the situation with a mixed core containing partly Mixed-Oxide fuel. Formally this is not an ENSREG requirement since up to now MOX fuel has not been deployed in the plant. However the use of MOX has to be taken in account in the future.

2.2 Requirement 2: Earthquake

ENSREG requires the licensee to provide information about the design basis of the plant in order to establish the magnitude of the earthquake against which the plant is designed. In addition to this the licensee has to evaluate the margins that exist on top of the design basis above which loss of fundamental safety functions or severe damage to the fuel (in vessel or in fuel storage) becomes unavoidable.
Further details about the ENSREG requirements on earthquake conditions can be found in appendix A.

KFD concludes that

- The information provided by EPZ on the vulnerability of the plant for earthquakes is plausible.

- Thorough safety margins for earthquake have not yet been given by EPZ.
- Additional surveys and studies have to be carried out, mainly on the seismic aspects of both the NPP and area where the NPP is situated.
- Technical and / or organizational improvements are initiated by the licensee, mainly on the fire-fighting system in buildings 01, 02 and 35 and on the filtered venting system.

2.3 Requirement 3: Flooding

ENSREG requires the licensee to provide information about the design basis of the plant in order to establish the magnitude of external flooding against which the plant is designed. In addition to this the licensee has to assess the differences 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.
Further details about the ENSREG requirements on flooding conditions can be found in appendix A.

KFD concludes that
- The information provided by EPZ on the vulnerability of the plant for flooding is sufficient and plausible given the model used by the licensee. However within the Netherlands other models with respect to flooding are known and in use. Some of these models give different values of water heights, water speeds and wave heights. The models have to be compared to each other and with existing data in order to draw a final conclusion.
- Safety margins on top of the design basis for flooding do exist.

2.4 Requirement 4: Extreme weather conditions

ENSREG asks for verification of the weather conditions that were used as design basis for various plants systems, structures and components: maximum temperature, minimum temperature, various type of storms, heavy rainfall, high winds, etc. by the licensee. Also the postulations of proper specifications for extreme weather conditions if not included in the original design basis are required.
Further details about the ENSREG requirements on extreme weather conditions can be found in appendix A.

The licensee evaluated the following weather conditions;
- maximum and minimum water temperatures of the River Westerschelde;
- extremely high and low air temperatures;
- extremely high wind (including storm and tornado);
- wind missiles and hail;
- formation of ice;
- heavy rainfall;
- heavy snowfall;
- lightning;
- credible combinations of the conditions mentioned above.

KFD concludes that
- The information provided by EPZ on the vulnerability of the plant for extreme weather conditions is sufficient and plausible.

- Safety margins for extreme weather conditions exist.
- However, additional surveys and studies have to be carried out on maximum roof-load of one building, the minimum depth of underground piping in order to be properly protected against freezing, and the possibilities to operate the diesel generators with extreme low temperatures.
- Technical and or organizational improvements have to be organised in order to cope with some of the extreme weather situations.

2.5 Requirement 5: Loss of electrical power and loss of ultimate heat sink

ENSREG asks for analyses of the situation of loss of electrical power and loss of ultimate heat sink. The two situations have to be analyzed using several different assumptions;

- Loss of off-site power
- Loss of off-site power and loss of the ordinary back-up AC power source (SBO-1)
- 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 (SBO-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)
- Loss of the primary ultimate heat sink and the alternate heat sink

In the case 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. Further details about the ENSREG requirements on loss of electrical power and loss of ultimate heat sink conditions can be found in appendix A.

KFD concludes that

Within the scope of ENSREG the following information is missing:

- In the case of SBO-2, credits for non-qualified systems, that need some necessary handling, are made within the first 24 hours.

Within the scope of ENSREG the following information is insufficient.

- The amount of diesel and lubricating oil should be considered as potential cliff-edge effects;
- Guides and procedures need to be developed such as deciding under which conditions the turbine oil pump must be switched off;
- Status of the core for the different solutions need to be considered too;
- In the case of SBO-2, the scenario without secondary feed and bleed should be analyzed.

Within the scope of ENSREG additional improvements shall be made by the Licensee EPZ:

- Qualification and classification of systems, structures and components important to handle severe accidents shall be defined.
- The quality of systems, structures and components important to handle severe accidents shall be enhanced.
- The systems, structures and components important to handle severe accidents shall be tested and the relevant procedures should be trained on a regular basis.

Finally, within the scope of ENSREG KFD concludes:

- No margins can be defined, but alternatives for loss of electrical (emergency) power and loss of ultimate heat sinks exist. However the alternative systems, structures and components are not qualified nor sufficiently proven;
- The potential modifications proposed by EPZ are necessary.

2.6 Requirement 6: Severe accident management

ENSREG requires the licensee to cover organization and arrangements for managing all type of accidents, starting from design basis accidents where the plants can be brought to safe shutdown without any significant nuclear fuel damage and up to severe accidents involving core meltdown or damage of the spent nuclear fuel in the storage pool.
Further details about the ENSREG requirements on severe accident management can be found in appendix A.

KFD concludes the following.

In general the ENSREG specification is well followed, but according to KFD the following issues or subjects are not (sufficiently) addressed with respect to severe accident management:

- Dose management
- Description of the radioactive release filters available in the installation
- Long term post-accident activities (beyond SAEG-1)
- Accident Management measures after the loss of containment (other than the use of existing equipment of the installation) such as:
  - Repair
  - Mitigating actions from outside
- Systematic analysis of access to rooms, equipment or reserve equipment in extreme situations for example:
  - access to the workshops and or warehouses for reserve equipment when the site is flooded
  - fires could start as indirect effect after aircraft crashes, explosions, earthquakes and lightning and could seriously impede access
- Potential hydrogen accumulations in other buildings than the containment

Although it is stated that the (Severe) Accident Management procedures are in place for all operational states it is unclear from the report what the typical differences are or which special actions are necessary for instance during shutdown. EPZ should provide additional information on this subject.
In a number of cases more information is needed to judge the issue.
The measures proposed are supported by KFD, but according to KFD more should be done. The list of measures that can be envisaged to enhance accident management capabilities should be extended by the additional measures from this review and the peer review process.

The most important additional issues according to KFD are:
The Severe Accident Management as it is currently organized makes use as far as possible from the existing resources and equipments that are available.
Equipment/instrumentation needed for the control of design basis accidents is in principle designed and classified for that purpose. For severe accidents it is assumed that equipment/instrumentation is used that is not designed or classified for that

purpose. The procedures are aimed at more or less checking what is available first and if so it will be used. KFD concludes that EPZ should start an action to improve the robustness of the plant by analyzing whether the equipment/instrumentation that is necessary or at least very useful to mitigate severe accidents can be improved (e.g. higher safety class, or from non-safety related to safety related, or introduce separate or additional equipment/instrumentation dedicated to the mitigation).

- Based on the lessons of Fukushima and R&D worldwide the SAMG's should be evaluated and improved when necessary.
- Before final measures are in place, temporary measures should be considered by EPZ.
- Analysis of the required additional staff and the organization for severe accidents has to be made.
- Evaluation of H-combustion vulnerability and H-management in several cases has to be improved.
- Severe accident mitigation strategies using other equipment than from the existing installation in cases like loss of containment and loss of water in the fuel pool have to be developed.
- The contract for external deliveries/support taking care of harsh circumstances has to be improved.
- The alternatives to provide electricity (AC and DC) like, batteries, battery loading, number of connections have to be improved.
- Procedures and solutions for the handling of large amounts of contaminated water have to be developed.
- The expected dose rates on the site and in the buildings have to be reevaluated.

2.7 Requirement 7: Other extreme hazards (additional requirement by the Dutch government)

The chapter of the licensee report covering this requirement handles other extreme hazards that may have a adverse effect on the nuclear power plant. This requirement is not included in the ENSREG requirements but is an additional requirement from the Dutch government. EPZ has evaluated the following potential threads;

- Internal explosion
- External explosion
- Internal fire
- External fire
- Airplane crash
- Toxic gases
- Large grid disturbances
- Failure of systems by introducing computer malware
- Internal flooding
- Blockage of the cooling water inlet

KFD concludes that

- The information provided by EPZ on the vulnerability of the plant for other extreme hazards is sufficient and plausible.
- Safety margins for external hazards are not clearly given.
- An evaluation of internal fire caused by electric powered equipment is missing.
- An evaluation of explosions on the NPP site outside of the buildings is missing

- Measures to account for the emergency control room not being protected against toxic gases are missing.
- As indicated by the licensee a more extensive study of the impact on the safety functions of different aircraft crashes has to be performed.

Appendix A: The ENSREG requirements for the national report including the detailed KFD review remarks

The ENSREG requirements are printed normal

The KFD remarks are printed italic

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 units • reactor type; • thermal power; • date of first criticality • existing spent fuel storage (or shared storage)
In this chapter in general minimum guaranteed quantities are reported. An exception seems to be the volume of UA demineralised water storage tanks of which the maximum volume is listed. KFD recommends that all quantities given in the report are checked to determine whether they are realistic, especially the UA, UJ storage tank and -pool.
1.3 Systems for providing or supporting main safety functions In this section, 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.
The information given in tables 1.8, 1.11 and 1.14 is not consistent. For example the RA main steam pressure relief valves in table 1.8: it is unlikely that the valves can not be operated with the power supply from emergency grid 1 while on the other hand they can be supplied by UPC1.
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).
Page 27, second bullet. Assumed is RS is the back-up of RL instead of RA.
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, as well as arrangements for accessing these, including where they are stored (both in relation to the site and protection from potential hazards), and whether they are shared between units or sites.
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.
In paragraph 1.3.6.4.1, it is stated that the emergency power system of the coal-fired plant (CCB) can feed emergency grid 2by means of a fixed connection. KFD recommends that this supply-configuration shall be demonstrated before credit is taken.
1.4 Significant differences between units This section 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.
During the latest IPSART mission at the Borssele NPP it was concluded that the "living" PSA of the plant requires updating.
The source term data presented in tables 1.23to 1.25 should be re-evaluated using the findings of the analysis of the Fukushima accident and the subsequent R&D activities in this field.
1.6 Future use of Mixed Oxide fuel
Since the Borssele NPP did not operate on a partially MOX fuelled core on the reference date for the analysis, 30th of June 2011, the impact of MOX on the analyses is not mandatory according to ENSREG. EPZ argues that during the licensing procedure it has been shown that the safety of the Borssele NPP with the use of MOX fuel is comparable with the current situation in which Enriched Natural Uranium (ENU) is used as fuel. However, for long term situations (>48 hours) the decay-heat of MOX-fuel is significantly larger than for ENU-fuel. Furthermore, the analyses presented in this report cover a much larger range of accident scenario's than analysed for the license application. Therefore the deployment of MOX fuel is not covered by the analyses provided in the report.

2. Earthquakes

Both the reactor and the spent fuel pools, as well as spent fuel storages at site, are to be considered.

2.1 Design basis
2.1.1 Earthquake against which the plants are 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.
Possible effects by human induced earthquakes should be mentioned; e.g. earthquakes as a result of gas drilling in the Northern part of The Netherlands and shale gas drilling in Noord-Brabant.
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.
The median return period associated with the intensity of the DBE corresponds to around 30,000 years. This has been adopted in the seismic hazard curve of the PSA, based on the original determination of the DBE. Information and systematics used to define this DBE might be dated. A re-evaluation should be carried out in order to update the DBE and the return period. In this re-evaluation the most recent insights in earthquake systematics and data must be taken into account.
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.
In the original definition of the DBE the site-specific local conditions are not taken into account. Although it is mentioned that in the update of the DBE (part of the second 10 yearly evaluation) the site conditions have been taken into account, specific data on the local geology are missing. So far, no seismic measurements have been carried out at the site. These measurements should be carried out in order to obtain more insight in the local site conditions (e.g. local geological features can cause local vibrations) and possible measures that must be taken to improve the resistance against earthquakes.
2.1.2 Provisions to protect the plants 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.
The evaluation of the seismic capacities of mechanical, electrical and instrumentation and control (I&C) systems is done by comparison with earthquake experiences at other NPP's. Although this might give some insight in the behaviour of systems, this doesn't provide a solid proof. Differences in design and local conditions influence the margins. These margins should be assessed.
In the analysis of the robustness of the systems required in order to achieve the safe shutdown state, insufficient attention is given to instrumentation and the availability of staff.
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, for instance
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.
In the report it is noted that the fire-fighting systems in buildings 01, 02 and 35 are not designed for operability after a DBE. This is listed as a weakness and therefore as a possible modification. In chapter 2.2.4 the weakness is again mentioned, however, a specific modification isn't mentioned. To enhance the protection against internal fire these systems should be qualified for the DBE.
2.1.2.3.2 Loss of external power supply that could impair the impact of seismically induced internal damage at the plants.
2.1.2.3.3 Situation outside the plants, 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).
The possibility of an explosion as an indirect effect of an earthquake has not been assessed. Only a possible fire is addressed
It is mentioned that liquefaction of certain ground layers cannot be ruled out. However, it is stated that there is sufficient margin against failure of the plant's foundations and loss of pile-bearing capacity. It is not made clear that liquefaction doesn't have negative influences on systems. This should be made clear.
2.1.3. Compliance of the plants with its current licensing basis
2.1.3.1 Licensee's processes to ensure that plants systems, structures, and components that are needed for achieving safe shutdown after earthquake or that might cause indirect effects discussed under the previous section remain in operable conditions.
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.
It must be made clear which internal and external mobile equipment is needed to control the DBE.
2.1.3.3 Potential deviations from licensing basis and actions to address those deviations.
Improvements on the availability and preparedness of auxiliary mobile equipment as made clear in the check performed as a result of the WANO Significant Operating Experience Report (SOER) are mentioned. These improvements should be further elaborated.
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 above which loss of fundamental safety functions or severe damage to the fuel (in vessel or in fuel storage) becomes unavoidable.
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 plants and consequent flooding exceeding design basis flood Possibility of external floods caused by an earthquake and potential impacts on the safety of the plants. 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.
Paragraph 2.2.3 mentions that the systems required for a safe shutdown in buildings 01, 02, 33 and 35 will remain available after a beyond design basis

earthquake of 0.3 g. In paragraph 2.2.1, however, it is mentioned that buildings 01 and 02 have a HCLPF capacity of 0.15 g. A more thorough seismic margin assessment is needed in order to determine the specific margins.
2.2.4. Measures which can be envisaged to increase robustness of the plants against earthquakes Consideration of measures, which could be envisaged to increase plants robustness against seismic phenomena and would enhance plants safety.
A cliff-edge is identified in the case that no personnel is available to replenish back-up feedwater system storage tanks within 10 hours.
The identified cliff-edge on the structural failure of the missile shield inside the containment for PGAs > 0,3 requires further clarification and analysis.
The measures proposed do not seem to be related to the identified cliff-edges. Although all measures proposed are worthwhile to consider, it should be made clear whether the identified cliff-edges give rise to further potential measures
The report gives an estimation of the seismic margins. PGA's that would result in a loss of fundamental safety functions or loss of integrity of the reactor containment are not estimated. The described measure of reducing the uncertainties in the seismic margins by carrying out a Seismic Margin Assessment or a seismic probabilistic Safety Assessment is supported by the KFD.

3. Flooding **Both the reactor and spent fuel pools, as well as spent fuel storages at site, are to be considered.**
3.1. Design basis
3.1.1. Flooding against which the plants are designed
3.1.1.1 Characteristics of the design basis flood (DBF) Maximum height of flood postulated in design of the plants 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.
In the case of flooding with a mostly intact dyke the wave height on-site is determined based on the diminishing effect of the dyke on the wave height and the enhancement of the wave height due to reflections against buildings. The latter effect is estimated to lead to a doubling of the wave height. In the case of flooding with a completely failed dyke only wave height reduction due to the shallowness of the site is assumed. No doubling due to reflections against buildings is taken into account. In order to perform a consistent analysis however, either the doubling of wave height due to reflections against buildings must be taken into account or the complete destruction of all buildings must be assumed. This contradiction should be clarified by the licensee
3.1.1.3 Conclusion on the adequacy of protection against external flooding
The NPP Borssele will be informed about expected water heights by external sources. There are no water level criteria given for preventive shut down the NPP. A preventive shut down level should be investigated by the licensee
In the analysis a possible weak point is identified: the stability of the end pylon of the overhead line connecting the step-up transformer to the grid during flooding situations. It is argued that in case of flooding this connection is not required as the normal procedure is to switch to house-load operation. In this argumentation the possible impact of a short circuit in the overhead line as a result of the failing end pylon on a flooded site is not analyzed. The licensee should analyze the effects of shirt circuit in the overhead lines.
According to some experts, extreme floods, may be violent and may have the potential to flood most of the province of Zeeland and even major parts of the Netherlands. According to aforementioned experts, this type of flood might feature a fast increase of water level. It also may feature large flow rates and speeds. As a consequence its waves may have the potential to entirely destroy dikes (like dikes A and B) and non-bunker type buildings. Above a certain level of damage to the dikes, they may not be able to limit the height and strength of the waves. It is unclear what impact this type of flood will have on the safety of the NPP. The licensee should make a comparison of the model used by the licensee and external experts.
3.1.2. 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. Provisions to protect the plants 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.
The NPP Borssele final report states that building 4 can withstand floods until 5 m +NAP but will be unavailable at 7.3 m +NAP. What is the status of the instrumentation (including wiring and power supplies) necessary to monitor and control the power plant during flooding conditions? The licensee should clarify this.
3.1.2.2 Main design and construction provisions to prevent flood impact to the plants.
3.1.2.3 Main operating provisions to prevent flood impact to the plants.
3.1.2.4 Situation outside the plants, including preventing or delaying access of personnel and equipment to the site.
3.1.3. Plants 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. Measures which can be envisaged to increase robustness of the plants against flooding. Consideration of measures, which could be envisaged to increase plants robustness against flooding and would enhance plants safety.
In the analysis of the effects of a rupture of the VC pipe flooding is only considered as a result of the event. It is however not unlikely that flooding will be the initiating event of VC pipe damages due to for example the static pressure of the water on site. The potential complications from the failure of the VC pipe on a flooding scenario have not been taken into account. In case of a rupture of the VC pipe without causing a flooding of the site, the plant will be more vulnerable to high tides and storm surges until the damage has been repaired. The report misses a clear set of measures to protect the plant in case this situation occurs. The licensee should analyze the effects of flooding on the VC system.
In the analysis of the flooding of buildings 33 and 35 it is suggested to install a parapet in order to prevent the splashing of waves against the air intakes. This measure is not mentioned in 3.2.2 where the measures to increase the robustness of the plant against flooding is discussed. This should be clarified by the licensee.

4. Extreme weather conditions
All relevant weather conditions have been assessed
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 plants systems, structures and components: maximum temperature, minimum temperature, various type of storms, heavy rainfall, high winds, etc.
Most minima and maxima are not designed based but result from engineering and/or national codes.
The plant will be informed by the local authorities in case of extreme water-level forecast.
4.1.1.2 Postulation of proper specifications for extreme weather conditions if not included in the original design basis.
A cliff-edge is found for the minimum air temperature (-18°C).
Concerning frozen fire lines: it is not clear what method has been used to determine the minimum depth of the lines in the soil that safeguards them from freezing. There is only a limited difference between the actual pipe depth (0,8m) and the mentioned frost zone (0,7m) in the soil.
It is not clear whether sufficient engineering features are in place to keep the VE-piping in the upper part of the wells free of frost.
The soil temperature at the level of the hydrant piping is not monitored. KFD suggest EPZ to consider the possibility to monitor the soil temperature around the hydrant piping.
A check should be made whether the emergency diesel generators and their auxiliary systems can operate for a longer period with extreme low temperatures caused by the combustion air.
The potential effect of build-up of snow dunes (accumulation of wind-transported snow) on the roofs of the buildings has no be taken into account.
Attention should be paid for the roof-load in combination with heavy rain or snow, especially building 33 which has such a high roof edge that the maximum roof-load will be exceeded in combinations with blocked drains and heavy (extreme) rain or other source of much water like fire fighting actions.
The effects of lightning are difficult to asses. In June 2008 several annunciators were spuriously activated and one of the two main feedwaterpumps tripped. Analysis of the incident showed that disturbance of the electronics was caused by lightning and damaged lightning-shielding on building 15 due to construction work.
LOOP caused by failures of pylons or salt on isolators forces the plant to fall back on emergency grid 1. House-load operation will not be possible as the lines will be short circuited and the plant has no generator switch.
4.1.1.3 Assessment of the expected frequency of the originally postulated or the redefined design basis conditions.
The figures used for the frequency will be assessed by the national Metrological institute.
4.1.1.4 Consideration of potential combination of weather conditions.
4.1.1.5 Conclusion on the adequacy of protection against extreme 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. Measures which can be envisaged to increase robustness of the plants against extreme weather conditions Consideration of measures, which could be envisaged to increase plants robustness against extreme weather conditions and would enhance plants safety.
The licensee has the intention to develop a check-list for plant walk-downs en required actions after exposure to various levels of the foreseeable hazards. Where applicable hardware provisions should be taken to increase the robustness of the backup emergency cooling water systems VE and UJ to extreme low temperatures

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

For writing chapter 5, it is suggested that 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.

In general the worst case is LPUHS-SBO2. For qualified systems 3 hours of secondary cooling is available using 185 m3 water of the feed water tank (RL). In this case the water is pumped into the steam generator using the steam driven pump (RL023) or forced using the steam coming from the outlet of the steam generator (RA), the so called secondary feed and bleed. All other options as suggested in the report are non qualified systems like using fire-hoses (UJ). This system is important in most scenarios.

KFD recommends that the UJ system shall be modified to a nuclear safety class 3 or higher safety class as proposed by EPZ. In addition, KFD recommends that the UJ system should be resistant to external events to a higher level than the design based external events. This includes both pumps. It should be investigated if the UJ system can be brought to a operational system for 24 hours without any physical human interaction.

In addition, the diesel generators of the coil fire plant are in this report for the first time mentioned as the back-up system in case of SBO-2. However, all back-up actions involving CCB are not qualified. KFD recommends testing this possibility and improving the connection according to the SAHARA principle.

The degradation of the core (in time) is not presented in the analyses. KFD recommends to have this information available to determine the highest priority during a severe accident.

Measure M9 should be accompanied by a measure to improve the cooling capacity of the well pump system in order to reduce the at least 13h before the pump system can be put in operation.

5.1 Nuclear power reactors

5.1.1 Loss of electrical power

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.

5.1.1.1 Loss of off-site power

Dependence on the functions of other reactors on the same site. Robustness of the provisions in connection with seism and flooding.
Autonomy of the on-site power sources and provisions taken to prolong the service time of on-site AC power supply

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

In addition to the mentioned diesel fuel storage, the lubricating oil

consumption and the amount of lubricated oil in stock should be considered as well.
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
Normally, the battery discharge time is two hours. However by switching of the turbine oil pump this time can be increased to 5.7 hours, but it will destroy the turbine. Because of the economical consequences it is unlikely that the turbine oil pumps will be switched off immediately and therefore the maximum battery time will be less than 5.7 hours. KFD recommends EPZ to establish a clear set of criteria to determine when to switch off the turbine oil pump in order to reduce the loss of battery time.
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 and spent fuel pool 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).
5.1.1.4. Conclusion on the adequacy of protection against loss of electrical power.
The first alternative power for emergency grid 2 is using the emergency diesel generators of the coal-fired plant. This is not a proven practice or a qualified system. The second alternative power for emergency grid 2 is using the mobile diesel generator EY080 (1.0 MW) located at the site. However, this diesel generator is only mobile when (heavy) external equipment can be brought to the site. It cannot be classified as portable light equipment that can arrive to the site from other locations after the first 24 hours. A third alternative power for emergency grid 2 is using an external diesel generator (1.0 MW) from Rotterdam. However, this diesel is not available. No procurement arrangements with suppliers are arranged. In addition, it can't be classified as portable light equipment that can arrive to the site from other locations after the first 24 hours. Furthermore, there are no statements regarding protection against external events for the above three mentioned alternatives and it is not mentioned consistently whether procedures for these alternatives exist. KFD recommends an alternative power system in the case of SBO-2 for the first 24 hours without any physical human interaction. This system should be resistant to external events to a higher level than the design based external events level.
5.1.1.5. Measures which can be envisaged to increase robustness of the plants in case of loss of electrical power
In the case of SBO2 and exhausted batteries the primary pressure vessel relief valve (YP) cannot be operated neither by hand. Primary feed and bleed is not

possible. KFD recommends that this should be clear before developing Extensive Damaging Management Guides.
5.1.2 Loss of the decay heat removal capability/ultimate heat sink 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.
5.1.2.1. 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. Robustness of the provisions in connection with seism and flooding.
5.1.2.2. Loss of the primary ultimate heat sink (e.g., loss of access to cooling water from the river, lake or sea, or loss of the main cooling tower)
5.1.2.2.1 Availability of an alternate heat sink, dependence on the functions of other reactors on the same site.
5.1.2.2.2 Possible time constraints for availability of alternate heat sink and possibilities to increase the available time.
The quality of the water of the fire fighting pond of the CCB 1.600 m3 is poor. It is doubtful whether it can be used as cooling water. The risk of blocking the equipment of the fire-fighting is unknown. KFD recommends further investigation of the actual usefulness of this cooling water.
5.1.2.3. Loss of the primary ultimate heat sink and the alternate heat sink
5.1.2.3.1 External actions foreseen to prevent fuel degradation.
5.1.2.3.2 Time available to recover one of the lost heat sinks or to initiate external actions and to restore core and spent fuel pool cooling before fuel damage: consideration of various examples of time delay from reactor shutdown to loss of normal reactor core and spent fuel pool cooling condition (e.g., start of water loss from the primary circuit).
Strategy in priority measurements are dealt with in chapter 6.2.2. KFD recommends that in paragraph 5.1.2.3.2 information should be presented about the possibilities and required time to implement alternatives or recover lost heat-sinks. No (time) information is given about the degradation process of the core in the several scenarios.
5.1.2.4. Conclusion on the adequacy of protection against loss of ultimate heat sink
5.1.2.5. Measures which can be envisaged to increase robustness of the plants in case of loss of ultimate heat sink
Running out of diesel is an extra potential cliff-edge effect in the case of using the fire brigade pumps (UJ).
Pag 5-59, a procedure for direct injection of VE by UJ is addressed as a potential action to increase the robustness of the installation. In addition, alternative supplies for UJ are addressed. In the latter, KFD suggests that VE/UJ/RL should be a part of this assessment.
5.1.3. Loss of the primary ultimate heat sink, combined with station black out (see stress tests specifications).
5.1.3.1. Time of autonomy of the site before loss of normal cooling condition of the reactor core and spent fuel pool (e.g., start of water loss from the primary circuit).
5.1.3.2. External actions foreseen to prevent fuel degradation.
5.1.3.3. Measures, which can be envisaged to increase robustness of the plants in case of loss of primary ultimate heat sink, combined with station black out
Running out of diesel is an extra potential cliff-edge effect in the case of using the fire brigade pumps (UJ).
5.2. Spent fuel storage pools

Where relevant, equivalent information is provided for the spent fuel storage pools as explained in Section 5.1 for nuclear power reactors.
5.2.1. Loss of electrical power
5.2.1.1 Measures which can be envisaged to increase robustness of the plant in case of loss of electrical power
5.2.2. Loss of the ultimate heat sink
5.2.2.1 Measures which can be envisaged to increase robustness of the plant in case of loss of ultimate heat sink
5.2.3. 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).
5.2.3.1. Measures, which can be envisaged to increase robustness of the plant in case of loss of primary ultimate heat sink, combined with station black out
See remark 5.1.2.3.2
No cliff-edge effects are specified.

6. Severe accident management
6.1. Organization and arrangements of the licensee to manage accidents Section 6.1 should cover organization and arrangements for managing all type of accidents, starting from design basis accidents where the plants can be brought to safe shutdown without any significant nuclear fuel damage and up to severe accidents involving core meltdown or damage of the spent nuclear fuel in the storage pool.
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
Calling of 2 additional shifts on site when the site is still intact: *• EPZ should develop criteria, based on the (predicted) evolution of the external hazard, when the additional shifts should be called.* *• Also the number of additional people needed for different hazards should be determined.* *• It is unclear how it can be guaranteed that two additional shifts are available during all the year; therefore EPZ should make arrangements to realize this* *• In case the site is flooded or after a (beyond) “design” earthquake, the current ACC could be available; therefore until the new shelter will be realized, a temporary solution should be developed/implemented.* *• It is unclear what measures are foreseen if the emergency organization is not complete for some reason (e.g. lines on communication and decision authorities). EPZ should provide additional information.*
6.1.1.3 Measures taken to enable optimum intervention by personnel
Tasks of ERO: *• EPZ should develop a strategy for the organization for a long duration of a complex emergency situation containing at least the following elements:* *o Provisions for communication with the public and the media.* *o Provisions for dealing with the long term, while the ERO is dealing with the actual situation.* *o Provisions for independent review of the ERO during its operation.* *• The incident of Fukushima has made clear that it is important to have communications with the public and liaison with the different authorities right from the beginning; EPZ should re-evaluate its current policy on this issue.*
6.1.1.4 Use of off-site technical support for accident management
EPZ should describe in more in detail the ROT en NPK organization and the corresponding responsibilities.
6.1.1.5 Procedures, training and exercises.
Procedures, training and exercises: *• In the light of Fukushima, EPZ should evaluate the contents and frequency of the SAMG’es training program, taking into account the EDMG’s and harsh circumstances, including amongst others:* *o Reduced accessibility of the site.* *o Reduced number of ERO staff.* *o Reduced availability of instrumentation.* *o Long duration of the accident.* *• Periodic exercises of SAMG’es are necessary to ensure maintenance of the*

capability and guidance usability.
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)
Improvement of off-site and on-site mobile diesel: *• The on-site mobile diesel generator shall be independent from external support.* *• The connection equipment should be simple and easy to handle.* *• The number of connection points should be analyzed/improved (not under all circumstances available). It should be analyzed what measures are necessary to use them in case of flooding.*
Reload of batteries: *• EPZ shall evaluate alternative means to reload batteries.*
6.1.2.2 Provisions for and management of supplies (fuel for diesel generators, water, etc.)
Contracts for chemicals or fuel supply : *• EPZ should evaluate/periodically control the contracts for fuel, chemical and boron supply to cope with extreme emergency situations (e.g. flooding).*
Repair of equipment: *• EPZ should evaluate the access to the workshops and or warehouses of reserve equipment in case the site is flooded or otherwise limited (e.g. by earthquake or fire) and if necessary improve the protection of those locations/buildings and the equipment.*
6.1.2.3 Management of radioactive releases, provisions to limit them
Contaminated water: *• EPZ should analyze/evaluate the capacity of the storage tanks and the processing of contaminated water for long term severe emergency situations, taking into account that the water could be heavily contaminated, and develop a written strategy/procedure to deal with it.* *• EPZ should analyze for severe accident situations the locations where contaminated water could appear (in- or outside the controlled area) and determine the way to deal with it.*
Radioactive release filtering: *• EPZ should evaluate the strategies for severe emergency situations to limit the potential radioactive release through other buildings.*
6.1.2.4 Communication and information systems (internal and external).
Voice and data communication: *• EPZ should determine the minimum time that the equipment, listed on page 6-12, should function in case of total loss of power and improve, if it is necessary.* *• For the independent voice and data communication EPZ should determine the criteria when to take it into operation.*
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
New Emergency Response Centre: *• EPZ should consider for the emergency response centre design to include*

amongst others: *o The storage of mobile equipment.* *o The storage important spare parts.* *o A workshop.* *o The storage of sufficient protective equipment for personnel, taking into account a long term radiological emergency situation.* *o The emergency control room function (backup).* *o The monitoring of important parameters.*
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
Dose management: *• EPZ should provide a dose management strategy for a long term emergency situation*
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
Accessibility of local control and sampling points: *• EPZ should re-evaluate accessibility of local control and sampling points in case of prolonged flooding (new emergency response centre will not solve this problem).*
Calculated dose rates (table 6.2): *• Based on the lessons of Fukushima and R&D, EPZ should (re)evaluate the dose rate calculations for several representative scenario's, including the experience of Fukushima-Daiichi taking in to account amongst others:* *o Different time evolution of releases.* *o Severity of the incident (e.g. lost containment).*
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)
Protection of equipment against severe external hazards: *• EPZ should identify all safety-critical structures, systems and components essential to providing defence in depth in a severe accident situation, and to propose measures to strengthen such SSC's to withstand events beyond the basis of the plant's design. Amongst others one could think of:* *o Upgrading of the safety classification of those SSC's.* *o Increased automation.* *o Increased protection and defence against the external hazard.*
6.1.3.7 Unavailability of power supply
6.1.3.8 Potential failure of instrumentation
Batteries and instrumentation *• EPZ should consider to create alternative/independent ways to reload the batteries* *• EPZ should (re)evaluate which (additional) instrumentation is necessary to monitor the situation in a severe accident case and if necessary improve the availability for the harsh circumstances* *• EPZ should show/analyze the procedure in case of complete loss of hydrogen measurement*
6.1.3.9 Potential effects from the other neighbouring installations at site, including

considerations of restricted availability of trained staff to deal with multi-unit, extended accidents.
Possible impact of Coal Fired Plant (CCB) • *EPZ should provide a comprehensive list of shared installations or services with CCB and the potential negative effects on the KCB-installation/organization during a severe accident*
6.1.4. Conclusion on the adequacy of organisational issues for accident management
6.1.5. Measures which can be envisaged to enhance accident management capabilities
The list of measures that can be envisaged to enhance accident management capabilities should be extended by the additional measures from this review and the ENSREG peer review process. *List of EDMG's:* • *EPZ should improve/provide a complete list.*
6.2. Accident management measures in place at the various stages of a scenario of loss of the core cooling function
6.2.1. Before occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes (including last resorts to prevent fuel damage)
6.2.2. After occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes
6.2.3. After failure of the reactor pressure vessel/a number of pressure tubes
6.3. Maintaining the containment integrity after occurrence of significant fuel damage (up to core meltdown) in the reactor core
6.3.1. Elimination of fuel damage / meltdown in high pressure
6.3.1.1 Design provisions
6.3.1.2 Operational provisions
6.3.2. Management of hydrogen risks inside the containment
6.3.2.1 Design provisions, including consideration of adequacy in view of hydrogen production rate and amount
H2-production from spent fuel pool: • *EPZ should provide additional information to prove that the H2-production of the fuel pool was considered in the design and allocation of the PAR's.*
Spray and Hydrogen production: • *EPZ should provide additional information on the use of the spray system in a severe accident if H2 is already present in the containment*
6.3.2.2 Operational provisions
Inertization of the containment: • *EPZ should provide additional information about the usefulness and effectiveness of Nitrogen injection from the accumulators.*
6.3.3. Prevention of overpressure of the containment
Venting system: • *EPZ should evaluate the necessity and possibility (including accessibility: e.g. radiation level too high) to replace the filtering material in case the system is used a long term during a severe accident.* • *EPZ should evaluate the vulnerability to combustion in the venting system and provide more information about the design (e.g. capacity of the filters, off-gas piping, yes or no separate pipe to the stack exit).*
6.3.3.1 Design provisions, including means to restrict radioactive releases if prevention of overpressure requires steam / gas relief from containment
6.3.3.2 Operational and organisational provisions
6.3.4. Prevention of re-criticality

Prevention of re-criticality: *• EPZ should present results of evaluation of the prevention of re-criticality during severe accident mitigation.*
6.3.4.1 Design provisions
6.3.4.2 Operational provisions
6.3.5. Prevention of base mat melting through
6.3.5.1 Potential design arrangements for retention of the corium in the pressure vessel
6.3.5.2 Potential arrangements to cool the corium inside the containment after reactor pressure vessel rupture
6.3.5.3 Cliff edge effects related to time delay between reactor shutdown and core meltdown
According to KFD the EPZ statement: "EOP's and SAMG's provide strategies to mitigate the accident for all possible scenario's" is questionable.
6.3.6. Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity
6.3.6.1 Design provisions
6.3.6.2 Operational provisions
6.3.7. Measuring and control instrumentation needed for protecting containment integrity
Batteries and instrumentation *• EPZ should consider to create alternative/independent ways to reload the batteries* *• EPZ should (re)evaluate which (additional) instrumentation is necessary to monitor the situation in a severe accident case and if necessary improve the availability for the harsh circumstances* *• EPZ should show/analyze the procedure in case of complete loss of hydrogen measurement*
6.3.8. Capability for severe accident management in case of simultaneous core melt/fuel damage accidents at different units on the same site
6.3.9. Conclusion on the adequacy of severe accident management systems for protection of containment integrity
6.3.10. Measures which can be envisaged to enhance capability to maintain containment integrity after occurrence of severe fuel damage
6.4. Accident management measures to restrict the radioactive releases
6.4.1. Radioactive releases after loss of containment integrity
Actions if the containment is lost: *• EPZ should develop severe accident management strategy/actions for the situation if the containment is lost (this paragraph deals mainly with the prevention). Additional equipment than the existing systems could be required in this case.*
6.4.1.1 Design provisions
6.4.1.2 Operational provisions
6.4.2. Accident management after uncovering of the top of fuel in the fuel pool
6.4.2.1 Hydrogen management
H2-production from spent fuel pool: *• EPZ should provide additional information to prove that the H2-production of the fuel pool was considered in the design and allocation of the PAR's.*
Spray and Hydrogen production: *• EPZ should provide additional information on the use of the spray system in*

a severe accident if H2 is already present in the containment
6.4.2.2 Providing adequate shielding against radiation
• *EPZ should develop severe accident management action for the situation of the complete loss of cooling water from the fuel pool (this paragraph deals with the prevention).*
6.4.2.3 Restricting releases after severe damage of spent fuel in the fuel storage pools
6.4.2.4 Instrumentation needed to monitor the spent fuel state and to manage the accident
6.4.2.5 Availability and habitability of the control room
6.4.3. Conclusion on the adequacy of measures to restrict the radioactive releases
See 6.4.1
Ad Annex 6.1 page 6-38: EPZ should evaluate the relocation or protection of the TL003 system.

7. Other extreme hazards
7.1. Introduction
7.2. Internal explosion
No specific attention is given to explosions on the NPP site but outside of the buildings
7.2.1 General description of the event
7.2.2 Potential consequences for the plant safety systems
7.3. External explosion
On page 7-8 is assumed that the buildings 33 and 35 could be destroyed by an external explosion. Nothing is said about the buildings 01 and 02. However in chapter 4 can be found that the resistances against blasts for the four buildings 01/02/33 and 35 are about equal
7.3.1 General description of the event
7.3.2 Potential consequences for the plant safety systems
7.4 Internal fire
No specific attention is given to internal fire cause by electric powered equipment. In general these kind of fire will lead to specific threads because of the nature of these fire. Effects as fire in cable ducts, short-circuits and availability of instrumentation should be taken into account
7.4.1 General description of the event
7.4.2 Potential consequences for the plant safety systems
7.5 External fire
7.5.1 General description of the event
7.5.2Potential consequences for the plant safety systems
7.6 Airplane crash
7.6.1 General description of the event
7.6.2 Potential consequences for the plant safety systems
7.7 Toxic gases
No measures seems to be considered give the fact that the emergency control room is not protected against toxic gases
7.7.1 General description of the event
7.7.2 Potential consequences for the plant safety systems
7.8 Large grid disturbances
7.8.1 General description of the event
7.8.2 Potential consequences for the plant safety systems
7.9 Failure of systems by introducing computer malware
7.9.1 General description of the event
7.9.2 Potential consequences for the plant safety systems
7.10 Internal flooding
7.10.1General description of the event
7.10.2Potential consequences for the plant safety systems
7.11 Blockage of cooling water inlet
7.11.1 General description of the event
7.11.2 Potential consequences for the plant safety systems

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