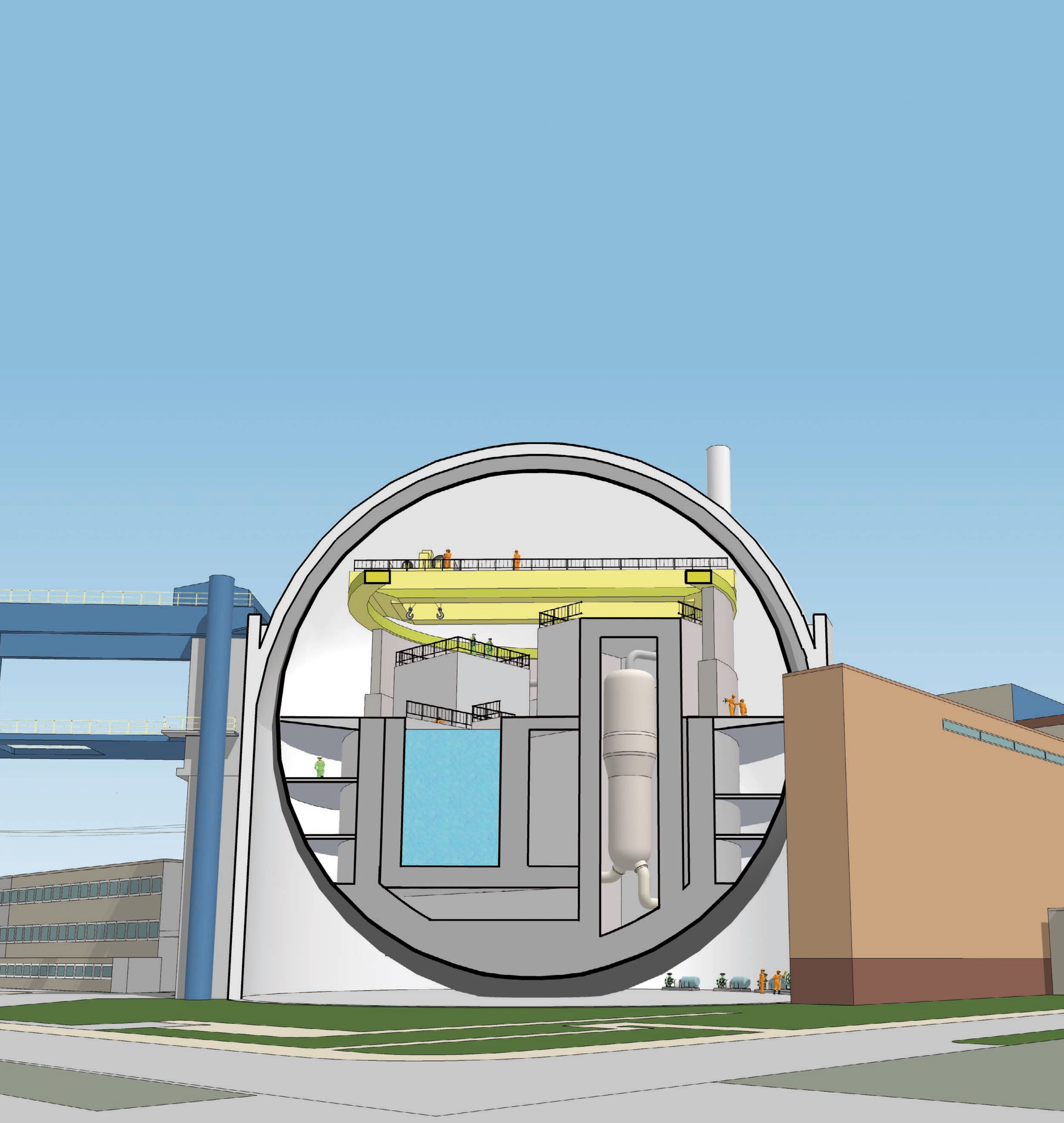
Final Report

Complementary Safety margin Assessment

October 31, 2011



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Prologue

On March 11, 2011, a large part of the Japanese eastern coastal area was devastated by an earthquake, followed by an immense tsunami. As a result, thousands of people were killed, injured or made homeless. In the days that followed, the situation was further complicated because of the failing nuclear reactors on the Fukushima coast. The local environment suffered from radioactive releases, requiring evacuation zones, and generating international concerns about nuclear safety.

In the wake of this disaster the European Union decided to assess safety on all operating nuclear reactors in its member states.

This safety evaluation initiated by the European Union focusses on extreme natural hazards, beyond the standard safety evaluations which regularly have to be performed to demonstrate the safety of a nuclear power plant.

Consequences of these extreme hazards for the Borssele NPP have been evaluated based on available safety analyses, supplemented by engineering judgement. In this way, the robustness of the existing plant has been assessed and possible measures to further increase the safety margins have been identified.

This document presents the results of the Complementary Safety margin Assessment (CSA) performed for the NPP Borssele.

The distinct difference between this report and former risk analysis reports in general and the existing Safety Report of the NPP Borssele is that the maximum resistance of the plant against redefined and more challenging events has been investigated, whereas traditionally the plant design is investigated against certain events that are determined on a historical basis. This different approach requires different analyses and studies, which in turn presents new insights into the robustness of the plant.

This document has been prepared in the short time period between June 1 and October 31, 2011. If more time had been granted for this study, some of the subjects could have been pursued in greater depth. The EPZ project team has been supported by several external experts. Apart from EPZ internal review, review has been performed by a dedicated steering committee, including independent outside experts.

The main purpose of this report is to answer the questions posed on EPZ by the Ministry of Economic Affairs, Agriculture and Innovation. It was decided to write at the same time a report in Dutch in order to to communicate the results of the CSA to the general public.

Executive summary

Complementary Safety Margin Assessment (CSA)

Following the accident at the Fukushima nuclear power plant in Japan, the European Council declared that “the safety of all EU nuclear power plants should be reviewed on the basis of a comprehensive and transparent risk assessment (‘stress test’)”. Based on this, the Ministry of Economic Affairs, Agriculture and Innovation (EL&I) requested the Elektriciteits Produktiemaatschappij Zuid-Nederland EPZ (EPZ) on June 1, 2011 to perform a targeted assessment of the safety margins of the NPP Borssele, based on the ENSREG specifications. In addition the Ministry indicated that in the assessment ‘delibirate disturbances’ should be taken into account. This request was implemented by EPZ as the ‘Complementary Safety margin Assessment’ (CSA). The results are presented in this Final Report.

Methodology

The methodology of the assessment consists on the one hand of an evaluation of the plant’s response when facing a set of extreme situations, and on the other hand of a verification of the preventive and mitigating measures that have to ensure the safety of the plant. In this assessment, the possibility of cliff-edge effects beyond the level of protection is identified (a cliff-edge effect is a small change in a parameter that leads to a disproportional increase in consequences).

The assessment considers three elements:

- provisions incorporated in the design basis and the plant’s conformance to its design requirements;
- evaluation of the design basis;
- assessment of the margins ‘beyond design’; how far can the design envelope be stretched until accident management provisions (design and operational) can no longer prevent fuel damage and/or a radioactive release to the environment.

The assessment of the margins ‘beyond design’ might require information about the plant that is not yet available. In those cases engineering judgement has been used to conservatively determine the margins.

The assessment leads to insights into severe accident conditions and how the NPP Borssele reacts, even if the emergency measures provided for that situation would fail. As a result, the assessment delivers the following insights:

- how the installation and its safety management systems react in ever more serious accidents in which protective measures are progressively defeated;
- the robustness of the installation and its safety management system;
- the potential for modifications to improve the robustness.

Within the scope the following extreme hazards are assessed:

- earthquake;
- flooding;
- extreme weather conditions;
- loss of electrical power supply and loss of ultimate heat sink;
- other extreme hazards that could be possible at the plant site, caused by various means, such as external and internal events;
- combinations of realistic hazards.

Combinations of hazards

The following combinations of hazards have been evaluated:

- earthquake and consequent flooding;
- high air temperature + high water temperature;
- low air temperature + low water temperature;
- snow + extreme wind;
- extreme wind + extreme rainfall + lightning.

In case of flooding extreme high level will not be attained without a storm (i.e. extreme wind and rainfall).

CSA in relation to continuous improvement at the NPP Borssele

The policy of EPZ is based on the principle that nuclear safety has an overriding priority. This is expressed in the pursuit of excellence through continuous improvement. The strive for improvement is executed by regular self-assessments and reviews by external organisations. The evaluation of international and internal experiences leads to the implementation of improvements.

Every ten years, a Periodic Safety Review (PSR) is conducted, which leads to more comprehensive improvements to adjust the installation and its operation to the current state-of-the-art. With these modifications, the design basis of the NPP Borssele, which originated in 1973, is strengthened. In 1994, a Design-Basis Reconstruction was carried out, which led to major improvements in 1997 (Project Modifications). Due to the regular update of the design basis in connection with the periodic safety reviews, the KCB design basis is, in many respects, consistent with the design basis of newer plants. For example situations caused by extended station blackout (SBO) and loss of off-site power (LOOP) can be envisaged for longer periods. The fourth PSR (10EVA13) is in progress at the moment and has to be finished by January 1, 2014.

Future use of Mixed Oxide fuel

Concerning the future use of Mixed Oxide fuel (MOX) in the NPP Borssele it is shown by profound analyses during the licensing procedure, that the safety of the NPP when using MOX fuel (as licensed to EPZ) is comparable with the safety in case Uranium Oxide is used as fuel. The consequences for man and environment also turned out to be comparable for both kinds of nuclear fuel. Therefore no separate assessment has been performed for MOX fuel.

Main conclusions of the assessment

The main conclusions on the resistance of the NPP Borssele against the assessed extreme hazards are given below. For each extreme hazard the validity of the design basis, the conformity of KCB with the design basis and the margins are briefly discussed.

Earthquake

The definition of the intensity of the design-basis earthquake (DBE) is adequate, both on deterministic and probabilistic grounds. The conformity of the plant with its design basis is ensured within the existing surveillance programme. The DBE corresponds with a peak ground acceleration of 0.075 g and it has been shown that there is significant seismic margin with respect to the fundamental safety functions. The lowest seismic capacity of all considered systems, structures and components has been assessed to be 0.15 g, based on an engineering judgement.

Flooding

The current nuclear base level for external flooding is adequate, both on deterministic and probabilistic grounds. Tsunamis have been taken into consideration in this re-evaluation. The conformity of the plant with its design basis is ensured within the existing surveillance programme. The safety-related systems, structures and components are sufficiently protected with the current design basis of the NPP Borssele to cover flooding up to a level of 7.3 m + NAP. Margins exist up to a water level of 8.55 m + NAP, and in some areas higher.

Extreme weather conditions

The current design basis for extreme weather conditions is adequate, both on deterministic and probabilistic grounds. Extreme weather conditions (e.g. high cooling water temperature, extreme wind, formation of ice and lightning) have been considered for their influence on the safety of the NPP Borssele. It is concluded that extreme weather conditions cannot lead to core damage.

Loss of electrical power supply and loss of ultimate heat sink

The NPP Borssele is amply protected against a loss of electrical power supply. Apart from the emergency power system with three redundant emergency diesel generators, a diverse and bunkered station-blackout power system is available with two redundant diesel generators and batteries with a minimum discharge time of 7.3 hours.

This system is protected against external events like flooding, earthquake and explosion. As a defence in depth measure, the emergency power system of the neighbouring coal-fired power plant and a mobile diesel generator are available. All this equipment is adequate to provide electric power for safe shutdown, cooling and prevention of radiological release with a very high probability.

In case of a loss of ultimate heat sink, the plant is able to cool the reactor and the spent fuel pool by the systems that are available on-site, without the need for external actions. As an additional safety feature, the plant has a reserve ultimate heat sink using groundwater pumps.

Since a loss of electrical power supply finally results in a loss of ultimate heat sink, the combination of these two events does not lead to any other situation than the one described in loss of ultimate heat sink.

Severe Accident Management

The current organisation and arrangements of the KCB to manage accidents is adequate. With respect to protecting the containment integrity, KCB is well equipped with accident management systems. The automatic catalytic recombiners and the filtered venting system are effective design provisions that mitigate high hydrogen concentrations and over-pressurisation of the containment. The SAMGs give necessary guidance to protect the containment and give additional strategies using operational systems. The accident management capabilities can be enhanced by development and training of a set of Extensive Damage Management Guides (EDMG).

Other extreme hazards

In addition to the above-mentioned initiating events, some other extreme hazards (explosions, fire, airplane crash, toxic gases, large grid disturbances, computer malware, internal flooding and blockage of cooling water inlet) are assessed. In general it is concluded that the NPP Borssele is well equipped to handle these events safely due to the spatial separation and redundancies of safety-related systems and, their installation in protected buildings. Moreover, non-computerised, i.e. not vulnerable to computer malware, electronic hardware is used for safety-related systems.

Uncertainty in the margins with respect to airplane crash could be reduced by performing a more extensive study on the impact on the safety functions of different airplane crashes.

Measures which can be envisaged to increase robustness of the plant

Combinations of extreme situations show that there are some areas at NPP Borssele where possibilities exist to enlarge the margins. These are included in this report. These are summarised in the following table.

Implementation of the measures mentioned in this table will most probably require hardware modifications in the plant.	
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.

In the framework of the CSA, the maximum resistance of the plant against external events has been investigated, whereas traditionally the plant design is investigated against certain pre-defined external events. These different approaches require different analyses and studies. Within the timeframe of the CSA, it was not possible to perform extensive studies and in some cases engineering judgement has been applied for establishing the margins. In general a conservative approach is chosen when applying engineering judgement. In some areas extensive studies could therefore reveal that the actual margins are larger than those presented in the CSA report. In other cases extensive studies would formally validate the margins presented in the CSA report. Furthermore, in some cases, extensive studies could reveal measures for further increasing the margins.	
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.

The CSA showed that the robustness of the plant against external hazards can be increased by implementation of a number of procedures.	
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

Execution of the procedures mentioned above requires mobile equipment. Below equipment is identified that is not sufficiently available at present.
List of equipment to be available for the Alarm Response Organisation. This list is not exhaustive and will be extended when writing the EDMG procedures.
• Mobile high-volume pump • Mobile high-pressure pump • Various flexible hoses • Leak repair materials • Mobile diesel generators • Grinding machines, drilling machines • Electronic personal dosimeters • Legal personal dosimeters • Clothing • Flashlights • Hand tools (hammers, screwdrivers, …)

Introduction

Following the accident at the Fukushima nuclear power plant in Japan, the European Council, meeting on 24 and 25 March 2011, declared that "the safety of all EU nuclear plants should be reviewed*,* on the basis of a comprehensive and transparent risk assessment ('stress test')".

On the basis of the proposals made by the Western European Nuclear Regulators Association (WENRA), the European Commission and members of the European Nuclear Safety Regulatory Group (ENSREG), it was decided to agree upon "an initial independent regulatory technical definition of a 'stress test' and how it should be applied to nuclear facilities across Europe". ENSREG finally provided the "EU 'stress test' specifications".

On June 1, 2011, the Ministry of Economic Affairs, Agriculture and Innovation (EL&I) requested the Elektriciteits Produktiemaatschappij Zuid-Nederland (EPZ) to perform a targeted assessment of the safety margins of NPP Borssele, based on the ENSREG specifications (see Annex 0.1 and 0.2).

In addition the Ministry indicated that in the assessment 'delibirate disturbances' should be taken into account. This request was implemented by EPZ as the Complementary Safety margin Assessment (CSA), of which the results are presented in this Final Report. Earlier, in August 2011, a Progress Report was released to inform the authorities about the status of the CSA. Beside this Final Report, which is meant for the regulatory bodies, a Dutch version will be released, which will make the results of the assessment accessible to the general public.

The CSA is defined as a targeted assessment of the safety margins of all the European nuclear power plants. This assessment consists on the one hand of an evaluation of the response of a nuclear power plant when facing a set of extreme situations, and on the other hand of a verification of the preventive and mitigating measures that ensure the safety of the plant.

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

The CSA focuses on extreme natural events like earthquake and flooding. It will also look for the consequences of loss of safety functions if the situation is provoked by indirect initiating events, for instance a large disturbance from the electrical power grid impacting AC power distribution systems, external fire or aeroplane crash. Furthermore, disturbances caused by deliberate human actions are taken into consideration.

The assessment leads to insights into severe accident conditions and how the NPP Borssele reacts, even if the emergency measures provided for that situation will fail. As a result, the assessment delivers the following insights:

- how the NPP Borssele and the safety management systems react in ever more serious accidents in which protective measures are supposed to be progressively defeated;
- the robustness of the installation and its safety management system;
- the potential for modifications to improve the robustness.

This Final Report gives the results of the stress test, with conclusions on the robustness of the NPP Borssele and the potential for modifications to further increase this robustness.

General safety policy

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

Periodic safety review

Every ten years, an extensive safety evaluation is performed on nuclear safety and radiation protection. Four main areas are evaluated: technical, organisational, personnel and administrative.

The evaluation focuses on nuclear safety and radiation protection. The objective of a ten-yearly safety evaluation is to make an evaluation compared to the state-of-the-art using a comprehensive assessment as to wether:

- the design basis and the safety documentation are still valid;
- the arrangements in place to ensure the plant's safety are still valid and effective;
- the plant conforms to current national and international safety standards and practices

The goal of the evaluation is to improve the design of the plant and its operation, so that the nuclear safety and radiation protection performance will increase. This means that the plant's design is, as far as possible, in accordance with the highest technical design levels for modern nuclear power plants and is operated in line with the latest safety guidelines and best practices.

The successive periodic safety reviews lead to improvements to adjust the installation and its operation to the current state-of-the-art. Three large periodic safety reviews have been conducted to date. These have led to improving resistance against the following:

- internal accidents (e.g. improved redundancy and separation of safety systems);
- internal hazards (e.g. protection from internal flooding and fire);
- loss of electrical power (e.g. separated emergency power systems and additional diesel generators);
- loss of ultimate heat sink (e.g. additional cooling by groundwater pumps);
- external events (e.g. additional safety systems in bunkered buildings, earthquake-resistant reinforcements and extended autonomy);
- severe accidents (e.g. accident management procedures, passive hydrogen recombiners and filtered containment venting).

With these modifications the design of the NPP Borssele has been strengthened and considerable margins were gained in the resistance against accidents and hazards. The fourth PSR (10EVA13) is in progress at the moment and has to be finished by 1-1-2014.

World Association of Nuclear Operators (WANO)

In addition to the 10 yearly safety evaluation, a separate safety review is initiated by EPZ's membership of the World Association of Nuclear Operators (WANO). As part of this membership, EPZ participates actively in a peer review programme, which means that WANO peer members are invited every four to six years to undertake a full-scope peer review of the nuclear safety of EPZ's NPP. One to two years after the peer review, WANO is invited for a peer review follow-up to check the progress in implementing the recommendations that were defined during the peer review. Other useful information from WANO is the operating experience reports produced, which are based on reported incidents in other NPPs. These reports are used by EPZ as an important knowledge source, and useful lessons learned by others are implemented.

ENSREG EU CSA specifications (See Annex 0.2)

The technical scope of the stress tests has been defined after considering the issues that were highlighted by the events that occurred at Fukushima, including combinations of initiating events and failures. The focus is on the following issues:

a) Initiating events

- Earthquake;
- Flooding;
- Extreme weather conditions;

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

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

c) Severe accident management issues

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

b) and c) are not limited to earthquake and tsunami as in Fukushima; flooding is included, regardless of its origin. Furthermore, extreme weather conditions have been included. Also, the assessment of consequences of loss of safety functions is relevant if the situation is provoked by indirect initiating events, for instance a large disturbance from the electrical power grid impacting AC power distribution systems, an external fire or aeroplane crash. In the Netherlands, malicious acts have been included in this list.

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

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

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

The reactor and spent fuel storage are presumed to be affected at the same time. The possibilities of degraded conditions in the surrounding areas of the site are taken into account. Consideration is given to:

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

Three main aspects are reported:

- Provisions incorporated in the design basis of the plant and the plant's conformance to its design requirements.
- Robustness of the plant beyond its design basis. For this purpose, the robustness (available design margins, diversity, redundancy, structural protection, physical separation, etc.) of the safety-relevant systems, structures and components and the effectiveness of the defence-in-depth concept have been assessed. Regarding the robustness of the installations and measures, one focus of the review is on identifying cliff-edge effects (a cliff-edge effect is a small change in a parameter that leads to a disproportionate increase in consequences) and, if necessary, considering measures for their avoidance.
- Potential for modifications that are likely to increase the robustness of the plant, in terms of strengthening the resistance of components or of strengthening their independence with other levels of defence.

In addition, the protective measures aimed at avoiding the extreme scenarios that are envisaged in the stress tests are described, in order to provide context for the stress test. The analysis is complemented, where necessary, by results of dedicated plant walkdowns.

With this aim in mind, the following are identified:

- the means to maintain the three fundamental safety functions (control of reactivity, fuel cooling, confinement of radioactivity) and support functions (power supply, availability of ultimate heat sink), taking into account the probable damage done by the initiating event;
- possibility for the use of mobile external means and the conditions for their use;
- existing procedures identified in one reactor that can be used to help another reactor.

Where severe accident management is identified, the the length of time before damage to the fuel becomes unavoidable becomes relevant:

- for fuel in the reactor vessel, the time before the water level reaches the top of the core, and the time before fuel degradation (fast cladding oxidation with hydrogen production);
- for fuel is in the spent fuel pool, the time before pool boiling is indicated, how long can adequate shielding against radiation be maintained, the time before the water level reaches the top of the fuel elements, the time before fuel degradation starts.

Assessment methodology

The methodology of the assessment consists of evaluating the plant's response when facing a set of extreme situations and verifying the preventive and mitigative measures necessary to ensure the safety of the plant. In this assessment, the possibility of cliff-edge effects beyond the level of protection is identified.

The assessment considers three elements:

- provisions incorporated in the design basis and the plant's conformance to its design requirements;
- evaluation of the design basis;
- assessment of the margins 'beyond design'; how far can the design envelope be stretched until accident management provisions (design and operational) can no longer prevent fuel damage and/or a radioactive release to the environment.

The assessment of the margins 'beyond design' might require information about the plant that is not yet available. In those cases engineering judgement has been used to conservatively determine the margins.

The assessment leads to insights into severe accident conditions and how the NPP Borssele reacts, even if the emergency measures provided for that situation fail. This means that for the determination of the safety margins, a deterministic approach is chosen. The intention is that an ever more serious threat (for example, an increasingly higher tidal wave or more serious earthquake) is assumed, so as to determine how the NPP Borssele and its safety management system respond and to what level of threat the safety systems work adequately.

Following a deterministic approach in the assessment of safety margins for the different issues, it is, important to know how likely it is that such an event can occur so as to provide further evaluation and improved measures. This information is also presented in this report.

Concerning the future use of Mixed Oxide fuel (MOX) in the NPP Borssele it is shown by profound analyses during the licensing procedure, that the safety of the NPP when using MOX fuel (as licensed to EPZ) is comparable with the safety in case Uranium Oxide is used as fuel. The consequences for man and environment also turned out to be comparable for both kinds of nuclear fuel. Therefore no separate assessment has been performed for MOX fuel.

The main aspects of the assessment for the different issues are described below.

Earthquake

The design-basis earthquake (DBE) for the NPP Borssele is specified and the conformity of the plant with this design basis is evaluated. The evaluation of the conformity of the plant with its design basis is carried out using the existing surveillance programme. The judgment on the adequacy of the design basis makes reference to the German KTA. This is justified in view of the fact that the seismic conditions at NPP Borssele are generally comparable to the German seismic conditions, in particular those in the seismically calm region of Northern Germany. The provisions to protect the plant against the DBE are discussed.

The evaluation assesses the seismic margins regarding the fundamental safety functions. Neither a seismic PSA nor an explicit Seismic Margin Assessment has been performed in the past. Thus the available seismic margins are elaborated as follows: first the concept of seismic margin is introduced, then the sources of seismic margin applicable to KCB are derived and finally an estimate of the seismic margin is given.

Flooding

The design-basis flood (DBF) for the NPP Borssele is specified and the conformity of the plant with this design basis is evaluated. The evaluation of the plant's conformity with its design basis is carried out using the existing surveillance programme. The protection against flooding of the plant systems, structures, and components that are needed for achieving and maintaining the safe shutdown state is determined.

The available margins with respect to flooding are evaluated by a step-wise increase in the flooding level. The condition of the different structures (buildings), systems and components are described for flooding levels up to 7.3 m +NAP (design level) and beyond.

Possible problems on the site and of access to the site are discussed.

Extreme weather conditions

The design basis of the NPP Borssele with respect to extreme weather conditions is evaluated. The following weather conditions are taken into account:

- high and low-water temperature of the River Westerschelde;
- extremely high and low air temperature;
- extremely high wind (including storm and tornado);
- wind-blown debris and hail;
- formation of ice;
- heavy rainfall;
- heavy snowfall;
- lightning.

The available margins with respect to extreme weather conditions are evaluated.

The credible combinations of the events above are also evaluated.

Loss of electrical power supply and loss of ultimate heat sink

The design provisions to prevent loss of off-site power and station blackout (LOOP/SBO), loss of ultimate heat sink (LUHS) and the combination of loss of ultimate heat sink and station blackout are described. The main aspects are redundancy, capacity, diversity and the autonomy period.

Increasingly severe situations are considered and it is indicated which options will or will not be available due to additional system failures. Descriptions of the ultimate means and evaluation of the time available to prevent severe damage of the reactor core and of the spent fuel in the pool in various circumstances are included.

Severe accident management

With regard to severe accident management (SAM) the following items, as mentioned in the ENSREG Safety Annex I EU CSA specification, are discussed:

a) Accident management measures currently in place at the various stages of a scenario of a loss of core cooling function;
b) Accident management measures and plant design features for protecting the containment function after an occurrence of fuel damage;
c) Accident management measures currently in place to mitigate the consequences of a loss of containment function;
d) Accident management measures currently in place at the various stages of a scenario of a loss of cooling function in the fuel storage.

For these items, the cliff-edge effects, the adequacy of the measures in place and the potential for new measures are discussed at each stage.

Other extreme hazards conceivable at the plant site

In addition to the above-mentioned initiating events (earthquake, flooding and extreme weather conditions), the following extreme hazards conceivable at the plant site are assessed:

- internal and external explosion;
- internal and external fire;
- aeroplane crash;
- toxic gases;
- large grid disturbance;
- millennium-bug-like failure of systems;
- internal flooding;
- blockage of the cooling water inlet.

The assessment of these hazards is limited to the consequences of the loss of safety functions provoked by these events and severe accident management issues.

At the same time that ENSREG envisioned this CSA, a so-called 'track no. 2' was included in the request, covering security issues. This analysis, which is conducted by the government, will not be made public. Consequently, this will not be reported in this assessment. Therefore, manmade violence against the NPP Borssele of any form is not mentioned explicitly in this report.

Project organisation

By order of the CEO EPZ established an experienced project team, led by a project manager and advised by a Steering Committee. In the Steering Committee, members from outside the nuclear environment and from outside EPZ (Prof. dr. W.C. Turkenburg and Prof. dr. ir. A.H.M. Verkooijen) were included.

The EPZ technical support department is responsible for the analyses, the reviews and the results, and in general for the technical quality of the report. The Head of Nuclear Power Plant Borssele (HKCB) has, in his responsibility for nuclear safety, executed an independent review of the report.

To ensure the necessary expertise and resources to generate the CSA report, experienced external parties have taken part in the project from the beginning.

Planning

The following planning is used for the CSA:

- June 1, 2011 Request to EPZ from the regulatory body to perform a "CSA"
- August 15, 2011 Licensee Progress Report released by EPZ
- September 15, 2011 National Progress Report of the Netherlands released to the EC
- October 31, 2011 Licensee Final Report (this report) released by EPZ
- December 9, 2011 National Progress Reports considered in the European Council
- December 31, 2011 National Final Report of the Netherlands released to the EC
- April 30, 2012 Finalisation of the peer reviews of the National Reports
- Middle of 2012 National Reports including peer reviews and conclusions considered in the European Council.

Transparency

Public information has a prominent place in the safety margin assessment. Both the National Report and the EC report will be made available to the public. EPZ has also decided to publish its own report in the Netherlands, with the restriction that the information should not be made public for security reasons. Furthermore, meetings will be organised by EPZ where other people will be invited to discuss the results.

Structure of the report

The report consists of seven chapters (apart from the introduction). The first chapter is for information purposes and provides some general data on the plant and its site. The site characteristics and the main characteristics of the unit are described, followed by a description of the systems that are important for safety. Finally this chapter describes the Probabilistic Safety Assessment and its main results. Chapters two to seven provide the result of the assessment. Chapters two, three and four report the results of the evaluation of the initiating events Earthquake, Flooding and Extreme weather conditions. Chapter five provides the consequences of the loss of safety functions (electrical power and ultimate heat sink), and chapter six reports the severe accident management provisions. The final chapter seven gives the results of the assessment of the other extreme hazards conceivable at the plant site.

The chapters assessing the initiating events (2, 3 and 4) describe firstly the design basis event against which the plant is protected, the available provisions to protect the plant and the compliance of the plant with its current licensing basis. This is followed by an evaluation of the vailidity of the current design basis. Subsequently, an assessment has been performed of the margins 'beyond design' that are available before fuel damage or radioactive releases can no longer be prevented. Finally the measures that can be envisaged to increase the robustness of the plant are described. Where appropriate, combinations of events are assessed.

Chapter 5, assessing the loss of electrical power and ultimate heat sink, describes the different scenarios that are relevant. This is followed by the evaluation of the adequacy of the available protection measure and the measures that can be envisaged to increase the robustness of the plant. Finally the combination of a loss of electrical power and the loss of of ultimate heat sink are described. The chapter describes this firstly for the reactor and secondly for the spent fuel pool.

Chapter 6 describes the organisation and arrangements of the NPP Borssele to manage accidents. The available accident management measures in case of loss of core cooling and to maintain the containment integrity after occurrence of fuel damage are provided. This is followed by accident management measures to restrict radioactive releases. The adequacy of the provisions is assessed and measures to enhance accident management capabilities are given.

Chapter 7 assesses other extreme hazards conceivable at the plant site. For each hazard the event is described followed by the potential consequences for the plant safety systems..

Annex 1.1. Letter to EPZ from the Ministery of Economic Affairs, Agriculture and Innovation, Identification ETM/ED/11074538, juni 1, 2011

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Ons kenmerk
ETM/ED / 11074538

Uw kenmerk

Bijlage(n)
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Datum 1 juni 2011

Betreft Uitvoering stresstest

Geachte heer Bongers,

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

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

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

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

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

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Directoraat-generaal voor Energie, Telecom en Markten
Directie Energie en Duurzaamheid

Ons kenmerk
ETM/ED / 11071538

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

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

Als resultaat zal de stresstest het volgende moeten opleveren:

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

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

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

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

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

Directoraat-generaal voor Energie, Telecom en Markten
Directie Energie en Duurzaamheid

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

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

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

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

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

Hoogachtend,

Mr. Anneke van Limburgh
MT-lid directie Energie en Duurzaamheid

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Annex 1.2. ENSREG Declaration and Safety annex I EU "Stress test" specifications

Declaration of ENSREG

ENSREG and the European Commission have worked intensively to provide a response to the request of the European Council on 25 March 2011.

Notably, they have developed the scope and modalities for comprehensive risk and safety assessments of EU nuclear power plants. On 13 May 2011, ENSREG and the Commission have agreed the following:

1. In the light of the Fukushima accident, comprehensive risk and safety assessments undertaken by the operators under the supervision of the national regulatory authorities of nuclear power plants will start at the latest by 1 June 2011. These assessments will be based on the specifications in annex 1 largely prepared by WENRA and will cover extraordinary triggering events like earthquakes and flooding, and the consequences of any other initiating events potentially leading to multiple loss of safety functions requiring severe accident management. The methodology of these assessments is covered by annex 1. Human and organisational factors should be part of these assessments;

2. Risks due to security threats are not part of the mandate of ENSREG and the prevention and response to incidents due to malevolent or terrorists acts (including aircraft crashes) involve different competent authorities, hence it is proposed that the Council establishes a specific working group composed of Member States and associating the European Commission, within their respective competences, to deal with that issues. The mandate and modalities of work of this group would be defined through Council Conclusions[1].

3. Paragraphs 1 and 2 above contribute to a comprehensive risk and safety assessment.

Annex I

EU "Stress tests" specifications

Introduction

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

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

Definition of the "stress tests"

For now we define a "stress test" as a targeted reassessment of the safety margins of nuclear power plants in the light of the events which occurred at Fukushima: extreme natural events challenging the plant safety functions and leading to a severe accident.

This reassessment will consist:

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

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

For a given plant, the reassessment will report on the response of the plant and on the effectiveness of the preventive measures, noting any potential weak point and cliff-edge

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

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

Process to perform the "stress tests" and their dissemination

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

The timeframe is as follows:

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

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

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

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

During the regulatory reviews, interactions between European regulators will be necessary and could be managed through ENSREG. Regulatory reviews should be peer reviewed by other regulators. ENSREG will put at the disposal of all peer reviews

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the expertise necessary to ensure consistency of peer reviews across the EU and its neighbours.

Peer review process

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

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

Transparency

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

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

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

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

Full transparency but also an opportunity for public involvement will contribute to the EU "stress tests" being acknowledged by European citizens.

<u>Technical scope of the "stress tests"</u>

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

<u>a) Initiating events</u>
- Earthquake
- Flooding

<u>b) Consequence of loss of safety functions from any initiating event conceivable at the plant site</u>
- Loss of electrical power, including station black out (SBO)
- Loss of the ultimate heat sink (UHS)
- Combination of both

<u>c) Severe accident management issues</u>
- Means to protect from and to manage loss of core cooling function
- Means to protect from and to manage loss of cooling function in the fuel storage pool
- Means to protect from and to manage loss of containment integrity

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

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

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

The next sections of this document set out:
- general information required from the licensees;
- issues to be considered by the licensees for each considered extreme situation.

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General aspects

Format of the report

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

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

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

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

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

Hypothesis

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

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

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

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

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

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

Information to be included

Three main aspects need to be reported:
- Provisions taken in the design basis of the plant and plant conformance to its design requirements;

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

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

To this aim, the licensee shall identify:

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

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

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

Supporting documentation

Documents referenced by the licensee shall be characterised either as:
- validated in the licensing process;
- not validated in the licensing process but gone through licensee's quality assurance program;
- not one of the above.

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

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Earthquake

I. Design basis

a) Earthquake against which the plant is designed :

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

b) Provisions to protect the plant against the DBE

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

c) Plant compliance with its current licensing basis:

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

II. Evaluation of the margins

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

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

e) Based on available information (which could include seismic PSA, seismic margin assessment or other seismic engineering studies to support engineering judgement), what is the range of earthquake severity the plant can withstand

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without losing confinement integrity.

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

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Flooding

I. Design basis

a) Flooding against which the plant is designed :

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

b) Provisions to protect the plant against the DBF

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

c) Plant compliance with its current licensing basis:

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

II. Evaluation of the margins

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

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

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Loss of electrical power and loss of the ultimate heat sink

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

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

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

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

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

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

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

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

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

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

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

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

Two situations have to be considered:

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

For each of these situations:

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

d) Loss of the primary UHS with SBO

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

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

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

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Severe accident management

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

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

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

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

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

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

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

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

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

- identify any cliff edge effect and evaluate the time before it;
- assess the adequacy of the existing management measures, including the procedural guidance to cope with a severe accident, and evaluate the potential for additional measures. In particular, the licensee is asked to consider:
    - the suitability and availability of the required instrumentation;
    - the habitability and accessibility of the vital areas ot the plant (the control room, emergency response facilities, local control and sampling points, repair possibilities);
    - potential H_2 accumulations in other buildings than containment ;

The following aspects have to be addressed:

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

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

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

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

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Chapter 1 General data about the site/plant

1.1 Brief description of the site characteristics

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

In Figure 1.1 the map of the Province Zeeland is shown in which the position of the KCB is indicated.

Figure 1.1 Map of the Province Zeeland in which the position of the KCB is indicated

Figure 1.2 shows an aerial photograph of the Borssele site.

Figure 1.2 An aerial photograph of the Borssele site

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

Borssele Nuclear Power Plant (unit BS30)

- Construction started in 1969 and it first produced in 1973;
- Gross capacity 512 MW, net capacity 485 MW.

Borssele Coal-fired Power Plant (unit BS12)

- Built as oil-fired station in 1972, it was converted to coal-firing in 1987 and is also able to use natural gas;
- Gross capacity 427 MW, net 404 MW;
- Modified to be also fuelled by phosphorus gas (by-product from a neighbouring industry) and biomass.

Wind powered turbines

- Five wind turbines in the near vicinity of the site, have been in operation since early 2005;
- Installed capacity 11.75 MW.
- The total net capacity installed at the site is 898 MW.

Figure 1.3 shows a plan of the area and the position of the buildings.

Surrounding area

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

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

There is an intensive shipping on the River Westerschelde; the number of ship movements amounts to over 40,000 per year. Their origin or destination is, in many cases, the port of Antwerp (Belgium). Included among these ships are transporters of dangerous materials, including LPG, flammable liquids and liquefied ammonia.

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

The NPP Borssele is located approximately 500 m from the nearest railway line. A local yard and sideline from the main line provide a service to the local ports and industries.

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

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

The closest military airbase is Woensdrecht in Noord Brabant, which is 40 km away in a northeasterly direction. Nuclear power plants in The Netherlands must have restricted airspace for military air traffic; the ground dimensions are 3.6 km square with a vertical height of 0.5 km.

There is one military facility in the area. It is the ammunition depot Ritthem at a distance of 5.5 km.

The village of Borssele (number of inhabitants: 1,500) is located about 1.4 km northwest of the site. The cities of Vlissingen, Middelburg, Goes and Terneuzen lie 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 1.3 Plan of the Borssele site buildings

1.2 Main characteristics of the unit

1.2.1 Technical description of Borssele NPP

Figure 1.4 Borssele Nuclear Power Plant

The Borssele Nuclear Power Plant was designed and built by Kraftwerk Union (KWU) and started commercial operation in October 1973. An overview of the plant is given in Figure 1.4.

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

The turbine generator installation consists of one high-pressure and three condensing dual-flow steam turbines, a generator and an exciter on a single shaft. The condensers have titanium tubes and are cooled with salt water from the Westerschelde. As 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.

The main control room was back-fitted during the second 10 yearly safety evaluation (Modification Project, 1997) and is based on an ergonomically optimisation of plant operation procedures, including emergency procedures. A redundant bunkered control room is available for controlled shutdown, core cooling and spent fuel pool cooling after external hazards and in beyond-design conditions.

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

The containment is a 46-meter spherical steel shell, which is in turn encapsulated by the concrete reactor building. The spherical shell not only contains the reactor and steam generators, but also the spent fuel pool. There is no separate fuel storage facility outside the containment. The water in the spent fuel pool contains boron at 2,300 ppm. Boron is, however, not required to guarantee a sub criticality of $K_{eff} < 0.95$.

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

The reactor is fuelled with 121 fuel assemblies 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 exists of 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 5.41% (w/w) fissionable plutonium. The maximum allowed number of MOX fuel elements in the reactor will be 48 (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 fuel elements and is the contractor for specialised maintenance and inspection jobs.

Figure 1.5 shows the plot plan of the unit.

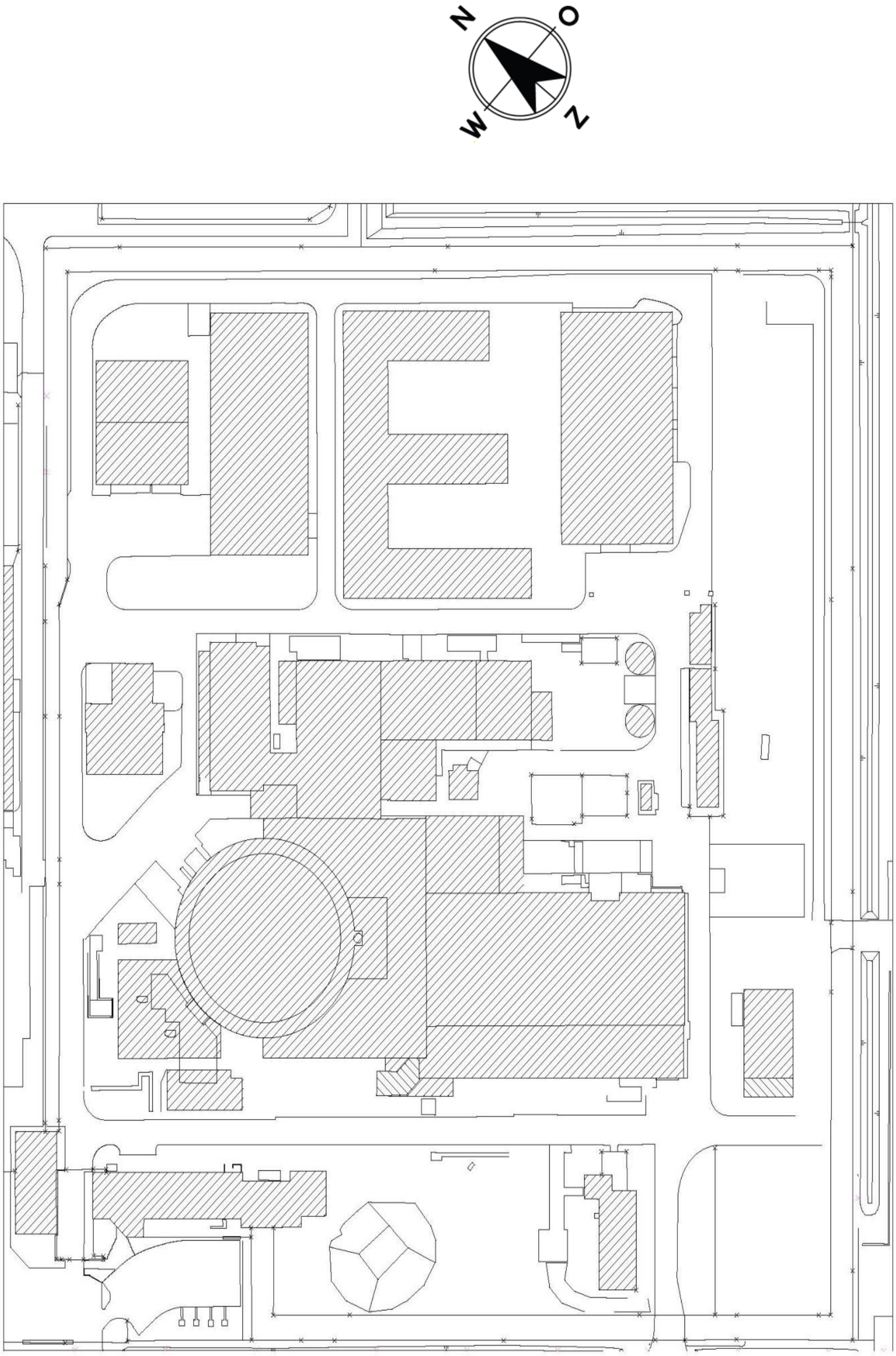


Figure 1.5 Map of NPP Borssele buildings

1.2.2 Key plant parameters and system characteristics

Table 1.1 and Table 1.2 present the key plant parameters and the main safety systems characteristics for Borssele NPP.

Characteristics and measuring units	Value
Primary coolant pressure at the core outlet, bar	155 bar
Coolant temperature at the reactor inlet, °C	292.5 °C
Coolant temperature at the reactor outlet, °C	317.5 °C
PRZ temperature, °C	362 °C
Coolant flow rate through the reactor, m^3/h	46,260 m^3/h
Steam flow rate at nominal parameters, kg/s	743 kg/s
SG pressure at nominal load, bar	58 bar
Steam temperature in SG at nominal load, °C	272.2 °C
Feedwater temperature in nominal mode, °C	214.2 °C
PRZ capacity (full volume), m^3	40.54 m^3
SFP capacity, m3	730 m^3
Reactor pool capacity, m^3	680 m^3
Boron concentration SFP, ppm	2,300 ppm

Table 1.1 Key plant parameters of NPP Borssele

System	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 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 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 m^3, 25 bar) each 2 trains of 2 storage tanks (178 m^3 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)
Key Safety Support Systems	Self testing reactor protection system Emergency control room Fire protection systems: Inergen, CO_2, fine water spray and Sprinkler systems, crash tender

System	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

Table 1.2 Main safety systems characteristics of NPP Borssele (con’t)

1.3 Systems for providing or supporting main safety functions

1.3.1 System descriptions

This section describes the systems in alphabetical order of system code.

For the system code description of the relevant systems see Annex 1.1.

For the building code description of the relevant buildings see Annex 1.2.

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 200% 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).

If 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 start up, 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 consists of 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 generator, EY040 and EY050 respectively. In addition, each bus bar is equipped with an auxiliary input for connecting a 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.

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.

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 shut-down 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.

Main condensate system (RM)

The Main condensate system 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 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.

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.

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:

- 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;
- 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;
- 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.

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 m^3/h. These pumps draw reactor coolant from the volume control tank, occasionally supplemented by the TB system, and discharge it into the 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 m^3, 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.

Chemical control 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 RCS, the 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.

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 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 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 VC to the Westerschelde.

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.

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 TF system. 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 UHS, either TG020 or TG025 can be switched to the TG080 cooler, using the VE cooling system (well water cooling system).

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:

Provide a means for normal decay heat removal from the reactor vessel during normal plant cooldown: the residual heat removal mode (TJR-RHR)
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.

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.

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.

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).

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.

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:

1. 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.

1.3.2 Reactivity control

Reactivity control of the reactor is achieved by control rods and boron concentration in the RCS water. The boron concentration is controlled with the Chemical control system (TB) combined with the Volume control system (TA). In case of an Anticipated Transient Without Scram (ATWS), the Backup coolant makeup system (TW) can shut the reactor down by injecting borated water (2,300 ppm).

See 1.3.1 for a description of TB, TA and TW.

1.3.2.1 Reactivity control for the spent fuel pool

Spent fuel is stored in specific racks containing neutron absorbing material. These racks with spent fuel are located in the spent fuel pool (SFP) in building 01. The SFP contains water with 2,300 ppm boron. The boron concentration in the TG system is controlled and adjusted by the Chemical control system (TB) combined with the Safety injection system & residual heat removal system (TJ). By design, with un-borated water in the spent fuel pool, the fuel in the storage racks remains subcritical ($K_{eff} < 0.95$).

See 1.3.1 for a description of TG, TB and TJ.

1.3.3 Heat transfer from reactor to the ultimate heat sink

1.3.3.1 Heat transfer chains for the reactor and operation states

The heat in the reactor core is transferred to the water in the RCS. The RCS can be cooled by the steam generators (its heat sink is secondary water) or by heat exchangers using water from the Westerschelde (this heat sink is Westerschelde water) or deep well water (its heat sink is deep well water). Each cooling option has its own specifications and limitations with respect to operation and process conditions. Deviation is made between the following operating modes:

- Hot standby ($T_{prim} \geq 180$ ºC and $K_{eff} < 0.99$);
- Hot subcritical (180 ºC $> T_{prim} > 80$ ºC and $K_{eff} < 0.99$);
- Cold shutdown with closed primary system ($T_{prim} < 80$ ºC and $K_{eff} < 0.99$);
- Cold shutdown with open primary system ($T_{prim} < 80$ ºC and $K_{eff} < 0.99$).

Hot standby

The heat of the RCS is transferred to the steam generators. Normally the steam generators are provided with water by the Main and auxiliary feedwater system (RL). If this system is not available the Backup feed water system (RS) can also provide water to the steam generators. In both cases the heat is released to the environment through the main condenser and VC to the Westerschelde. If the main condenser is not available the heat is released to the environment by the safety and relief valves of the Main steam system (RA).

Hot subcritical

During the subcritical phase the heat can be removed by two different heat sinks, being the Westerschelde and the deep well water. The heat from the RCS to the Westerschelde can be transferred by:

- Main steam system (RA)-condenser-VC- Westerschelde, like in hot stand by mode) till a temperature of about 120°C;
- Backup feed water system (RS) in case of unavailability of RA;
- Safety injection system & residual heat removal system (TJ) - Component cooling water system (TF) - Conventional emergency cooling water system (VF).

The transfer of heat from the RCS to deep well water released to the environment is via:

- Backup residual heat removal (TE);
- Backup cooling water system (VE).

The first cooling chain can transfer the heat on a higher primary pressure and temperature compared to the latter. The limiting starting conditions for the systems are:

- TJ/TF/VF → $T_{prim} \leq 180$ ºC and $P_{Prim} \leq 30$ bar;
- TE/VE → $T_{prim} \leq 120$ ºC and $P_{Prim} \leq 13$ bar and more than 13 hours after shut down.

Cold shut down with closed or open primary system

During cold shutdown with closed or open RCS, the heat from the reactor can be removed by:

- TJ/TF/VF (i.e. water from the Westerschelde);
- TE/VE (i.e. water from deep well pumps).

See for a description of RL, RS, TE, TF, TJ, VE and VF.

1.3.3.2 Layout of heat transfer chains and protection against internal/external events

The available heat transfer chains might differ for the different operational modes. With respect to lay-out and the protection to internal and/or external events the following can be summarized.

For all systems the physical separation between redundant pumps is achieved by fire bariers, except for TE. For the VE system the physical separation is achieved by separation of the 8 deep wells.

Protection against internal events is achieved by appropriate measures like fire barriers, flood protection and warning, separation of cable routings.

Protection against external events is assured for RS, TE and VE. Additionally, some (parts) of systems are protected against earthquake or flooding, like RA from the steam generators to the steam isolation valves or some TJ-valves that are designed for earthquake in order to enable operation of TE. The robustness of these systems is elaborated upon in the succeeding chapters.

Further general information on these systems is presented in the respective system descriptions in 1.3.1.

1.3.3.3 Water supply of heat transfer chains and possibilities for extension

The time it takes for a heat transfer chain to remove heat from the reactor depends on the amount of water available, the capacity of the system to provide enough water flow and the amount of heat to be removed. The first two items are known and the water supply alternatives can be identified. The amount of heat to be removed is also known, but this changes with time (residual heat of the core). For example, during the first 5 minutes after SCRAM the heat can be removed with about 10 m^3 water, while 60 hours after the same amount of water is sufficient to remove the heat for about 1 hour. This means that the runtimes depend on when the heat has to be removed. For this reason, no runtimes have been determined, but instead information is presented for a specific accident scenario for which the runtime can be determined. This information includes alternative water supplies as well as the supply rate (water flow). At the end of 1.3.3.3, information is provided on the systems which are directly driven by diesel generators. These systems are related to fire-fighting.

Heat transfer chain RL/RA

The water source for the Main and auxiliary feedwater system (RL) is the feed water tank which contains 185 m^3 water. In Table 1.3 the systems that can supply water to RL are presented. The flow rate of the feed water pump (RL) is related to the steam generator level. See for more details on RM, RZ and UA 1.3.1.

System	System code	Minimum water volume (m^3)	Supply rate (m^3/h)
Feed water tank	RL	185	86.4 at 10.2 bara
Hot well	RM	41	936 at 20 bar
Demineralised water supply tank	RZ	268	12 at 12 bar
Demineralised water preparation tank	UA	814	45 at 4 bar

Table 1.3 Systems that can supply water to the Main and auxiliary feed water system (RL)

Heat transfer chain RS/RA

The Backup feed water system (RS) consists of two trains, both with their own water supply. In the case of failure of one RS train , these water supplies can be coupled together in order to make full use of the available water supply. The minimum available water source is 900 m^3 water (2 x 450 m^3). The operation of the RS pump is related to the steam generator level. The water supply is enough for 72 hours of heat removal.

Alternative water sources are the Low-pressure fire extinguishing system (UJ) and the fire-fighting pond at CCB (see Table 1.4). The UJ system has a fixed connection to the RS-system to refill the RS tank. The UJ tank is refilled by the public water supply with a capacity of 48.5 m^3/h. In emergency situations this capacity can be enlarged to 180 - 200 m^3 water. To achieve this, the following actions have to be taken.

The RS system uses a free filler opening with a fire hose connection to the fire pond to refill the RS tank. The fire pond can also use the UJ-tank as water source. See for more details on UJ 1.3.1.

System	System code	Minimum water volume (m^3)	Supply rate (m^3/h)
Basins of the Backup feed water system	RS	900	144 at 90 bar
Water tank of the fire extinguishing system	UJ	1,200	360 at 10 bar 48.5 / 180 - 200 refill
Water of the firefighting pond of CCB	UJ	1,600	192 at 10 bar 119 / 107 at 4 / 10 bar 223 / 85 at 5 / 9 bar

Table 1.4 Available water supplies for the heat transfer chain RS/RA

Heat transfer chain TJ/TF/VF

In case the Conventional emergency cooling water system (VF) is not available alternative water supply can be delivered by the Low-pressure fire extinguishing system (UJ) and from the fire-fighting pond at CCB (see Table 1.5). The VF system normally uses water from the Westerschelde. Therefore the fire-fighting pond can also use the Westerschelde as a water source. For the latter option, support from external fire brigades is necessary. Connections for fire hoses are available at the VF system. See for more details on UJ 1.3.1.

System	System code	Minimum water volume (m^3)	Supply rate (m^3/h)
Conventional emergency cooling water system	VF	unlimited	2,100 at 2.1 bar
Water tank of the fire extinguishing system	UJ	1,200	360 at 10 bar 48.5 / 180 - 200 refill
Water from the fire-fighting pond of CCB	UJ	1,600	192 at 10 bar 119 / 107 at 4 / 10 bar 223 / 85 at 5 / 9 bar
Water from the Westerschelde for fire-fighting pond	UJ	unlimited	192 at 10 bar 119 / 107 at 4 / 10 bar 223 / 85 at 5 / 9 bar

Table 1.5 Alternative water supplies in case the Conventional emergency cooling water system (VF) is not available

Heat transfer chain TE/VE

In case the Backup cooling water system (VE) is not available, an alternative water supply can be delivered by the Low-pressure fire extinguishing system (UJ) and the fire-fighting pond at CCB (see Table 1.6). The VE system normally uses water from its deep water wells, which provide brackish water. This means that the fire-fighting pond can also be resupplied from the Westerschelde. For this latter option, support from external fire brigades is necessary. A connection for a fire hose is available at the VE system in the bunkered building 33. See for more details on UJ 1.3.1.

System	System code	Minimum water volume (m^3)	Supply rate (m^3/h)
Backup cooling water system	VE	unlimited	220 at 12 bar
Water tank of the fire extinguishing system	UJ	1,200	360 at 10 bar 48.5 / 180 - 200 refill
Water from the fire-fighting pond of CCB	UJ	1,600	192 at 10 bar 119 / 107 at 4 / 10 bar 223 / 85 at 5 / 9 bar
Water from the Westerschelde for fire-fighting pond	UJ	unlimited	192 at 10 bar 119 / 107 at 4 / 10 bar 223 / 85 at 5 / 9 bar

Table 1.6 Alternative water supplies in case the Backup cooling water system (VE) is not available

Diesel generator driven systems

The Low-pressure fire extinguishing system (UJ) and the fire-fighting trucks are diesel generator driven. Table 1.7 presents the data as well as the minimum runtimes. The runtimes are based on minimum available fuel in the dedicated tank. These minimum amounts are 75% and 40% of the tank capacities of the Low pressure fire extinguishing system and firefighting trucks respectively.

System	System code	Minimum fuel (m^3)	Fuel consumption rate (m^3/h)	Runtime (h)
Low- pressure fire extinguishing system	UJ	0.45	0.075	6
Fire-fighting truck 4942	UJ	0.12	0.028	4
Fire-fighting truck 4940	UJ	0.032	0.024	1
Fire-fighting truck 4941	UJ	0.053	0.024	2

Table 1.7 Minimum available fuel, fuel consumption and the minimum runtimes of the Low-pressure fire extinguishing system (UJ) and the fire-fighting trucks

1.3.3.4 Electrical power supply to heat transfer chains

The available AC power is off-site power, house load power, NS 1 and NS 2. NS 1 consists of EY010/020/030. NS 2 has its own diesel generators (EY040/050) but it can get additional electrical power from the emergency power system at CCB and from the mobile diesel generator (EY080). The electrical power supply to the components of the heat transfer chains is presented in Table 1.8.

System	System code	Off-site power	NS 1	UPS 1	NS 2	UPS 2
Pressure relief valves	RA	x		x	x	x
Emergency feedwater pumps	RL	x	x[1]	x		
Main condensate system	RM	x				
Backup feed water system	RS	x			x	x
Demin water supply system	RZ	x	x	x		
Backup residual heat removal system	TE	x			x	x
Component cooling water system	TF	x	x	x		
Safety injection & residual heat removal system	TJ	x	x	x		
Demineralised water plant	UA	x				
Low-pressure fire extinguishing system	UJ	x[2]				
Backup cooling water system	VE	x			x	x
Conventional emergency cooling water system	VF	x	x	x		
Pressure control system	YP	x	x	x	x	x

Table 1.8 Electrical power supply to the components of the heat transfer chains

[1] Of the 3 RL-pumps 2 are electrically driven and one is steam-turbine driven
[2] Of the 2 UJ pumps, one is diesel-generator driven and one is electrically driven and powered by emergency power from CCB

1.3.3.5 Cooling of heat transfer chain components

The components of a heat transfer chain might need cooling. It is important to know whether the failure of such a component cooling will result in a failure of the component followed by a failure of the system. In Table 1.9, the cooling requirement for each component is identified and if cooling is necessary, the means of cooling are indicated.

System	System code	Air	TF	VF	VE	Other
Pressure relief valves	RA					
Main and auxiliary feed water system	RL			x[3]		UK
Main condensate system	RM					
Backup feed water system	RS					
Demin water supply system	RZ					
Backup residual heat removal system	TE				x	
Component cooling water system	TF					
Safety injection system (H is high pressure)	TJH		x			
Safety injection & residual heat removal system	TJL		x			
Demineralised water plant	UA					
Low-pressure fire extinguishing system	UJ					
Backup cooling water system	VE					
Conventional emergency cooling water system	VF					

Table 1.9 Identification of component cooling in a heat transfer chain and, if applicable, the means of cooling

The vulnerability of the heat transfer chains with respect to loss of component cooling is presented in Table 1.10.

Heat transfer chain	Air	TF	VF	VE	Other
RL/RA			x		UK
RS/RA					
TJ/TF/VF		x			
TE/VE					

Table 1.10 Vulnerability of the heat transfer chains with respect to loss of component cooling

[3] If VF is lost the cooling of the emergency feedwater pumps is automatically taken over by the normal service water system (UK). The UK system retrieves it water from the public water supply. If the public water supply is lost too the UK system can provide cooling for 18 hours, except for the case that UK water is used elsewhere e.g. fire fighting

1.3.4 Heat transfer from spent fuel pools to the ultimate sink

1.3.4.1 Heat transfer chains and/or means for the spent fuel pool

The spent fuel pool normally contains fresh fuel elements and used or spent fuel elements. Only the used and spent fuel elements produce decay heat. This heat, which is produced by the fuel, is transferred to the water in the spent fuel pool. The cooling of pool water is carried out by the Spent fuel pool cooling system (TG). This system is split into one part for normal operation (whose heat sink is the Westerschelde) and another part for specific accident conditions (whose heat sink is deep well water). The heat transfer chains for spent fuel pool cooling are called TG/TF/VF and TG080/VE. If the Spent fuel pool cooling system (TG) fails, the produced heat of the spent fuel results in heating up the pool water until the boiling point is reached. The next step is the evaporation of pool water, which results in a loss of pool water. In principle the heat is released to the containment. In order to maintain cooling and radiation protection from the spent fuel elements, it is necessary to refill the pool.

1.3.4.2 Lay out of heat transfer chains and protection against internal/external events

The available heat transfer chains for spent fuel pool cooling are TG/TF/VF and TG080/VE.

With respect to lay-out and the protection to internal and/or external events the following can be summarized: for all systems the physical separation between redundant pumps is achieved by fire bariers, except for TG. All three TG-pumps are located in a one room, however spatially separated and with a very limited fire-load (only greasing oil for one pump) in the room. For the VE system the physical separation is achieved by separation of the 8 deep wells.

Protection against internal events is achieved by appropriate measures like fire barriers, flood protection and warning, separation of cable routings.

Protection against external events is assured for TG080 and VE. Additionally, some (parts) of systems are protected against external events, like TG in order to enable operation of TG080. The robustness of these systems is elaborated upon in the succeeding chapters.

Further general information on these systems is presented in the respective system descriptions in 1.3.1.

1.3.4.3 Water supply for the heat transfer chains and possibilities for extension

The ultimate heat sink for the heat transfer chains of the spent fuel pool cooling are is the Westerschelde via VF and VE. The information on water supply, additional water supply and the possibilities for extension are already described in section 1.3.3.2. With regard to heat transfer by the heating up and evaporation of pool water, the water content in the spent fuel pool is approx. 730 m^3 and the total water content in the spent fuel pool above the active part of the fuel is approx. 565 m^3.

1.3.4.4 Electrical power supply to heat transfer chains

The components of the heat transfer chains of the spent fuel pool are presented in Table 1.11.

System	System code	Off-site power	NS 1	UPS 1	NS 2	UPS 2
Spent fuel pool cooling system	TG	x	x	x	x	
Component cooling water system	TF	x	x	x		
Conventional emergency cooling water system	VF	x	x	x		
Spent fuel pool cooling system	TG080	x			x	x
Backup cooling water system	VE	x			x	x

Table 1.11 Electrical power supply of the components of the heat transfer chains of the spent fuel pool

1.3.4.5 Cooling of heat transfer chain components

The components of a heat transfer chain might need cooling. It is important to know whether the failure of such a component cooling will result in a failure of the component followed by a failure of the system. In Table 1.12 the cooling requirement for each component is identified and, if cooling is necessary, the means of cooling are indicated.

System	System code	Air	TF	VF	VE	Other
Spent fuel pool cooling system	TG		x[4]			
Component cooling water system	TF					
Conventional emergency cooling water system	VF					
Spent fuel pool cooling system	TG080					
Backup cooling water system	VE					

Table 1.12 Identification of component cooling in the heat transfer chain of the spent fuel pool and, if applicable, the means of cooling

The vulnerability of a heat transfer chain with respect to loss of component cooling is presented in Table 1.13.

Heat transfer chain	Air	TF	VF	VE	Other
TG/TF/VF					
TG080/VE					

Table 1.13 Vulnerability of the heat transfer chains with respect to loss of component cooling

[4] TG030 has a seal cooling by TF/VF; TG020/025 are cooled by own TG water

1.3.5 Heat transfer from the reactor containment to the ultimate heat sink

1.3.5.1 Heat transfer chains for the containment

Heat removal from the containment during design-basis accidents is not necessary because of the large air volume in the containment. In the case of severe accidents, the spray system can be used for this purpose. The spray pumps get water from the inundation tanks of the Safety injection system (TJ).

If these tanks are empty, the TJ system is operated in the recirculation mode (TJR): the water from the sump is cooled by heat exchangers (TJ) and the heat is transferred to the Component cooling water system (TF) and the Conventional emergency cooling water system (VF).

1.3.5.2 Lay- out of the heat transfer chain and protection against internal/external events

The available heat transfer chain from the containment to the Westerschelde is TJ spray/TF/VF.

With respect to lay-out and the protection to internal and/or external events the following can be summarized.

For all systems the physical separation between redundant pumps is achieved by fire bariers.

Protection against internal events is achieved by appropriate measures like fire barriers, flood protection and warning and separation of cable routings.

Protection against external events is not assured, although physical separation can result in increased robustness. The robustness of these systems is elaborated upon in the succeeding chapters.

Further general information on these systems is presented in the respective system descriptions in 1.3.1.

1.3.5.3 Water supply for the heat transfer chain and possibilities for increasing the supply

The ultimate heat sink for the heat transfer chain of the containment cooling is the Westerschelde via VF. The information on water supply, additional water supply and possibilities for increasing this supply has already been described in 1.3.3.3. The water supply for the spray system is initially the Safety injection stocks (TJ). When these stocks are empty the system can be refilled.

1.3.5.4 Electrical power supply to the heat transfer chain

The components of the heat transfer chain for the containment are presented in Table 1.14.

System	System code	Off-site power	NS 1	UPS 1	NS 2	UPS 2
Containment spray system	TJ-spray	x	x	x		
	TJL	x	x	x		
Component cooling water system	TF	x	x	x		
Conventional emergency cooling water system	VF	x	x	x		

Table 1.14 Electrical power supply to the components of the heat transfer chain of the containment

1.3.5.5 Cooling of heat transfer chain components

The components of a heat transfer chain might need cooling. It is important to know whether the failure of such a component cooling will result in the failure of the component followed by a failure of the system. In Table 1.15, the cooling requirement for each component is identified and if cooling is necessary, the means of cooling indicated.

System	System code	Air	TF	VF	VE	Other
Containment spray system	TJ-spray					
	TJL		x			
Component cooling water system	TF					
Conventional emergency cooling water system	VF					

Table 1.15 Identification of the need for cooling and, if applicable, the means of cooling for the heat transfer chain of the containment

1.3.6 AC power supply

1.3.6.1 Off-site power supply

1.3.6.1.1 Reliability of off-site power supply

The off-site power supply in the Netherlands is reliable. The historical data for NPP Borssele show there was 1 event in 15 years when a switching error during maintenance in the Borssele switchyard resulted in the loss of both 150 kV AC buses. Borssele's specific data have been combined with general off-site power statistics in the Netherlands and results in a failure rate of 1.5 . 10^{-2} per year for loss of off-site power.

The recovery time after loss of off-site power versus probability is based on data in the Netherlands and is presented in Figure 1.6. This shows that if loss of off-site power occurs, a recovery time of 1 hour has a probability of 0.83.

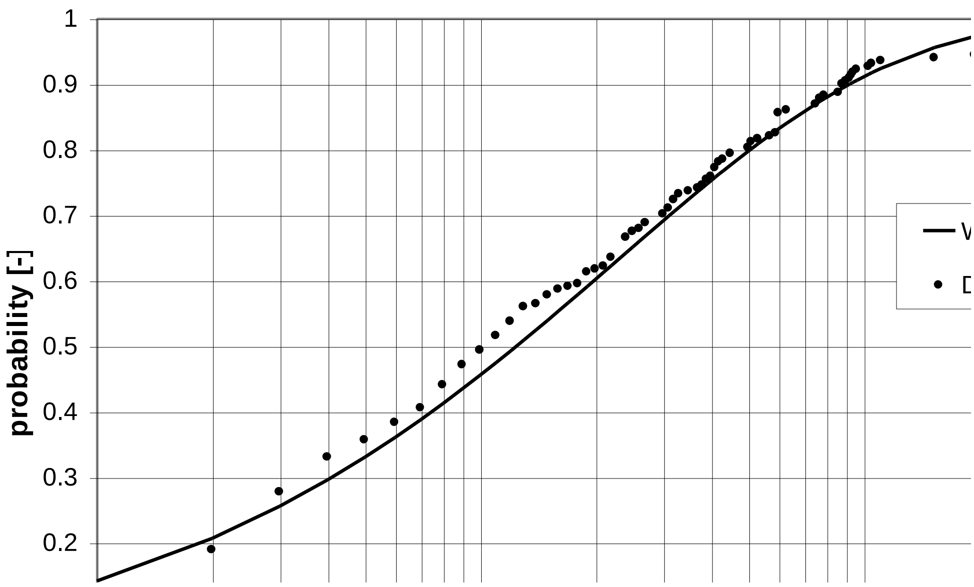


Figure 1.6 Probability of recovery after loss of off-site power as function of time

1.3.6.1.2 Connections from the plant to external power grids

The electrical energy at a voltage-level of 21 kV generated by the turbine generator is transformed to a voltage level of 150 kV and transported to the 150 kV grid using a step-up transformer. The electrical energy required for feeding the plant's systems is supplied by a house-load transformer. The house grid is 6 kV. During start-up and shut-down the energy to this grid is supplied by two start-up transformers, which are connected to the 150 kV off-site grid.

The same grid (6 kV) can be fed by the neighboring conventional power plant by two separate lines (also 6 kV). In addition Emergency Grid 2 is fed by 10 kV lines, that are connected to the 150 kV grid.

A mobile diesel generator can also be connected to this section of the power system.

1.3.6.2 Power distribution inside the plant

1.3.6.2.1 Main cable routings and power distribution switchboards.

The 6 kV house grid consists of two redundant rails, which are operated in parallel and independent of each other. The house grid is divided into three parts, which are the normal grid, Emergency Grid 1 (NS 1) and the bunkered Emergency Grid 2 (NS 2). NS 1 and NS 2 have their own backup power supply (diesel generators) and each system is capable of safely shutting down the reactor in the event of Loss of Off-site Power and removing the heat for 72 hours. Via transformers, the 6 kV is converted into 380 V to feed the various systems. NS 1 and NS 2 each have their own DC power grid fed by AC/DC converters. Both systems have batteries to ensure an uninterrupted power supply to feed safety-relevant systems and components (i.e. I&C equipment, reactor protection system, etc.) in case AC power is lost (see Figure 1.7).

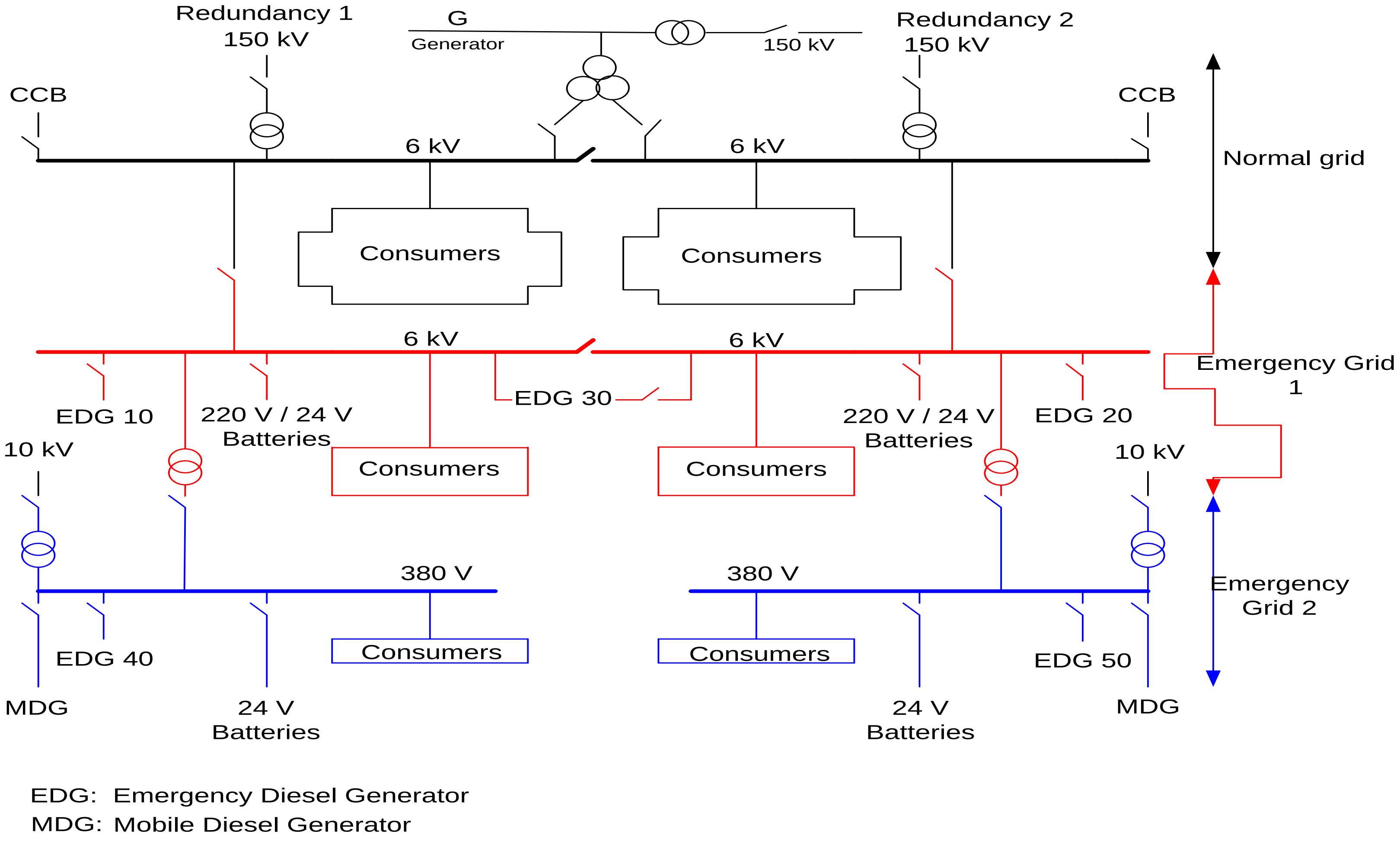


Figure 1.7 Borssele NPP electrical power system

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

The normal grid supplies power to the systems and components that are needed for both normal operation and the safety systems. However the systems and components for normal operation are usually not powered by the emergency power systems. Except for redundancy and physical separation no specific protection against internal- or external hazards is specified.

1.3.6.3 Main ordinary on-site source for back-up power supply

1.3.6.3.1 First back-up for Loss of Off-site Power: house load operation

In case of loss of the 150 kV grid, the main transformer is automatiquely isolated from the grid and the turbine regulation runs back to 'house power load'

1.3.6.3.2 Second back-up for Loss of Off-site Power: Emergency Grid 1 (NS 1)

In case of Loss of Off-site Power the next electrical power supply backup is Emergency Grid 1 (NS 1). NS 1 consists of 2 redundant power supply chains, each with a diesel generator. The third diesel generator is a spare; in case of failure of one diesel, the spare is started and coupled to the failed train.

1.3.6.3.3 Lay-out of Emergency Grid 1 (NS 1) and protection against internal/external events

The diesel generator system has two redundancies, which are housed in one building, physically separated and protected (e.g. double fire resistant doors) that ensures protection against internal events. They are cooled by air coolers. The third diesel generator system is housed in a separate building (distance ~ 50 m) and is cooled by the Conventional emergency cooling water system (VF).

1.3.6.3.4 Runtime and backup measures for NS 1

After loss of off-site power NS 1 is available for about 3 days (72 hours).

The capacity of the fuel tanks for the first 2 diesel generators is 95 m^3 which gives a runtime up to 79 hours (see Table 1.16). The spare diesel generator (EY030) has a runtime of 25 hours. The most important external action is to replenish the diesel fuel, thus resulting in unlimited electrical power supply. Arrangements are in place with a fuel supplier to replenish the fuel tanks within 8 hour after a request is received. No specific arrangements are in place in case of infrastructural problems resulting from for example, earthquake or flooding.

System code	Fuel tanks (m^3)	Fuel consumption rate (m^3/h)	Runtime (h)
EY010	95	~1.2[(5)]	79
EY020	95	~1.2	79
EY030	30	~1.2	25

Table 1.16 Amounts of fuel, consumption rates and runtimes of the diesel generators of NS 1

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

1.3.6.4.1 Diverse backup power supplies: Emergency Grid 2 (NS 2)

Following a loss of off-site power and failure of NS 1 the bunkered Emergency Grid 2 (NS 2) will provide power to the systems and components required to safely shut down the reactor and remove residual heat. NS 2 is powered by 2 redundant chains, each with a diesel generator.

The first alternative power supply to NS 2 is the emergency power system of the coal-fired power plant (CCB). This emergency power system consists of 2 redundant diesel generators which can feed NS 2 via the fixed CCB connection.

The second alternative power supply to NS 2 is the mobile diesel generator located on-site. This system consists of 1 diesel generator on a truck which can feed NS 2 directly via the 380 kV rail. Currently the mobile diesel generator needs external support to transport the system to the location of the NS 2 connection point.

A third alternative is the connection to NS 1 via transformers to the 6 kV BU/BV rails.

[5] The presented value is the maximum fuel consumption rate

1.3.6.4.2 Lay-out and protection against internal/external events

The principal power supply of NS 2 is housed in one building with two seperate entrances and physical separation / protection that ensures protection against internal- and external events. The CCB's emergency power system is located more than 200 m from the reactor building on the EPZ site. The mobile diesel generator is located about 400 m away from the reactor building. The redundancy, the separation of redundant sources and their physical separation ensures protection against internal and external hazards of NS 2

1.3.6.4.3 Runtime and back-up measures

After loss of off-site power and NS 1, the bunkered NS 2 is available for 72 hours. This period is guaranteed by redundant means. The runtime could be extended if, after the loss of off-site power and NS 1, NS 2 was powered by just one diesel generator and the other diesel generator was shut off in order to save diesel fuel. This might extend the runtime by 1 or 2 days.

The two emergency diesel generators at the CCB have a common fuel tank. When off-site power is lost, both diesel generators start up and keep running. This means that the runtime is valid for the simultaneous operation of both diesel generators. This runtime could be extended by switching off one diesel generator. However no procedure is in place for such actions.

The mobile diesel generator has its own diesel tank (3 m^3).

The most important external action is replenishment of the diesel fuel resulting in unlimited electrical power supply. Arrangements with a fuel supplier for replenishment of fuel tanks within 8 hours after a request are in place. No specific arrangements are in place in case of infrastructural problems as result of for example earthquake or flooding. The amounts of fuel, consumption rates and runtimes of the diesel generators of NS 2, CCB's emergency diesels generators and the mobile diesel generator are presented in Table 1.17.

System code	Minimum fuel (m^3)	Fuel consumption (m^3/h)	Runtime (h)
EY040	2.7	0.1[6]	
EY050	2.7	0.1	
Additional tank for EY040/050	8.8		
Total runtime EY040/050			72
Emerg. diesel generators (2) CCB	4.0	0.43	3
Mobile diesel generator EY080	3	0.3	10

Table 1.17 Amounts of fuel, consumption rates and runtimes of the diesel generators of NS 2, CCB's emergency diesels and the mobile diesel generator

[6] The fuel consumption given is an average value for a 72-hours operation

1.3.6.5 External power sources

1.3.6.5.1 Potential dedicated connections to other power plants

There are no connections to other power plants.

1.3.6.5.2 Transportable power sources

Arrangements have been made for the transport and installation of 1 or 2 external diesel generators for the Emergency Grid 2 (NS 2). Connection points inside the bunkered building are in place.

1.3.6.5.3 Information on the transportable diesel generator

Technical information on the transportable diesel generator(s) is presented in Table 1.18.

	Unit	Value
Number of units	-	1 or 2
Delivered power	kVA	1000
Voltage	V	400
Cos ϕ	-	0.8
Frequency	Hz	50
Requirements on voltage	%	+/- 4
Requirements on frequency	%	+/- 4

Table 1.18 Technical information on the transportable diesel generator EY080

1.3.6.5.4 Preparedness to utilise the transportable diesel generator

The transportable diesel generators are owned by an international company, which has its head office in The Netherlands and subsidiaries in many countries in Europe. The delivery of 1 or 2 of their diesel generators is possible within 4 hours from a subsidiary in The Netherlands. On-site staff needs about 4 hours to connect a diesel generator to the Emergency Grid 2 (NS 2).

Procedures for connection of an external power supply to NS 2 are in place. However no specific arrangements are in place in case of infrastructural or connecting problems resulting from, for example, earthquake, flooding or bad weather.

1.3.7 Batteries for DC power supply

During normal operation the DC power grid is fed by rectifiers powered by AC power. If there is of loss of off-site power, the emergency power grids NS 1 and NS 2 will adapt this task. In order to ensure uninterrupted power supply, both emergency power grids have their own backup DC power supplies in the form of batteries. Table 1.19 summarises the discharge times of the various battery banks.

Battery system	Discharge time (h)
UPS 1 (NS 1)	
220V	2.8
+ 24V	2.3
- 24 V	2.6
UPS 2 (NS 2)	
+ 24V	7.3
- 24 V	10.5

Table 1.19 Discharge times of the various battery banks

1.3.7.1 Consumers of battery banks

NS 1 and NS 2 each have their own redundant battery backup to provide an uninterrupted power supply for. The UPS of NS 1 provides 24 V DC to safety related instrumentation & control, the control room, the reactor protection system, etc. The 220 V grid of NS 1 can supply uninterrupted power to containment isolation valves, primary system isolation valves and other valves of safety-related systems as well as to the turbine’s emergency oil pump and lighting on the escape routes. The UPS of NS 2 provides 24 V to safety related instrumentation and control, the emergency control room and the reactor protection system.

1.3.7.2 Lay out and protection against internal/external events

The battery configuration of the emergency grids NS 1 and NS 2 each consists of two redundancies. The battery configuration of NS 1 is housed in normal buildings, while that of NS 2 is housed in a building designed for external events.

1.3.7.3 Recharging of battery banks

Currently there are no possibilities available on-site for recharging a battery bank. When there is of total loss of AC-power, the batteries will be lost after about 7 hours. If an external diesel generator is connected to NS 2, the DC power grid would be functional again and the batteries of NS 2 would be recharged as well.

1.4 Significant differences between units

Not applicable.

1.5 Scope and main results of Probabilistic Safety assessment

1.5.1 Overall scope of the Probabilistic Safety Assessment (PSA)

NPP Borssele has implemented a full scope "living" PSA. This therefore includes the following:

1. A complete Level-1 PSA for power and non-power operations;

A Level-2 PSA to identify the likelihood and mechanism of potential releases from the containment;

A Level-3 PSA to assess the dose consequences and the associated risk to the population.

1.5.1.1 Scope of the Level-1 PSA

The scope of Level-1 PSA includes power and non-power operations. The Level-1 analysis evaluates the core damage frequency and plant damage state frequencies, and identifies the main weak points in the plant safety features.

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

The Level-1 analysis of the external events has been conducted based on a successive screening process. First, the external event scenarios were identified, the initiating event frequency quantified, and the impact on the plant determined. If the frequency was low, then the scenario has been screened out. If the frequency was above the truncation frequency, then the plant response and other factors were considered.

1.5.1.2 Scope of the level-2 PSA

The total core damage frequency obtained in the Level-1 PSA was further developed into a release frequency in the Level-2 analysis. The severe accident progression analysis was performed using MAAP4, with ex-vessel phenomena being analysed with more detailed mechanistic codes to obtain details concerning core/concrete interaction, hydrogen distribution and containment loads.

The Level-2 analysis is based on 12 accident sequences representing the major physical processes during accident progression. The source terms resulting from the Level-2 analysis were then used as the input for the Level-3 analysis.

1.5.1.3 Scope of the Level-3 PSA

Using the source terms that form the output of the Level-2 analyses, the COSYMA computer program has been used to calculate the radiological consequences to human life, wild-life and vegetation.

1.5.2 Results of the Probabilistic Safety Assessment

1.5.2.1 Results of the Level-1 PSA

The results of full-power Level-1 PSA are presented in Table 1.20

TCDF-P			
Initiating event group	**Subgroup**	**Core Damage Frequency (per year)**	**Percentage of total**
Internal	LOCA (Loss of cooling accidents)	1.95 .10^{-7}	13.3%
	ISLOCA (Interfacing systems LOCA)	5.78 . 10^{-9}	0.4%
	SGTR (Steam generators tube rupture)	2.85 .10^{-8}	1.9%
	SBO (Station blackout)	2.05 . 10^{-9}	0.1%
	ATWS (Anticipated transients without scram)	7.99 .10^{-9}	0.5%
	TRANS (Transients)	4.77. 10^{-9}	0.3%
	SSIE (Special system initiators)	2.02 . 10^{-7}	13.8%
Internal hazards		9.38 . 10^{-7}	64.1%
External hazards		7.92 . 10^{-8}	5.4%
Total		**1.46 . 10^{-6}**	**100.0%**

Table 1.20 Full power Level-1 PSA results

This plant operational state (POS) is the largest contributor to the total core damage frequency (TCDF) due to the fact that the plant is usually in this POS for about 95% of the year. In Table 1.20 it can be seen that internal hazards provide the highest contribution to the TCDF for power operation conditions.

The results of low-power and shutdown Level-1 PSA are presented in Table 1.21.

Total CDF contributions of all POSs			
	Fraction of year	Core Damage Frequency (per year)	Core damage frequency fraction
POS-HE/HL[7]	5.92E-3	$9.44 \cdot 10^{-9}$	1.28%
POS-RE/RL	5.80E-3	$1.34 \cdot 10^{-7}$	18.17%
POS-ME/ML	8.93E-3	$5.56 \cdot 10^{-7}$	75.40%
POS-CU/CL	9.78E-3	$6.20 \cdot 10^{-9}$	0.84%
POS-FE	2.28E-2	$3.18 \cdot 10^{-8}$	4.31%
Total		**$7.37 \cdot 10^{-7}$**	**100%**

Table 1.21 Low power and shutdown Level-1 PSA results

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

Although there is only a short time interval in this POS, the POS is important because of the reduced inventory and the fact that there is no redundancy for the low-pressure TJ system.

Similar to power operation, internal hazards are the highest contributors to the core damage frequency for low-power and shutdown states.

The contribution of low-power and shutdown POSs to the TCDF is about 35%.

1.5.2.2 Results of the Level-2 PSA

For scenarios with occurrence of core melt there are different time frames associated with the releases from the containment:

- Early release: release between 0 and 12 hours following reactor trip or shutdown;
- Late release: release between 12 and 72 hours following reactor trip or shutdown;
- Very late release: release greater than 72 hours after reactor trip or shutdown.

A summary of the source term frequencies is presented in Table 1.22.

Summaries for the fission product release fractions for the three release phases are presented in Table 1.23 to Table 1.25. For the inventories of the nuclides in the reactor core immediately after shut-down is referred to Annex 1.4.

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

Time phase	STC	Frequency	Percentage of total	Percentage of time phase total	Containment release mode
Early releases	1	$4.41 \cdot 10^{-9}$	0.2%	18.8%	Dry SGTR without isolation
	2	$1.66 \cdot 10^{-10}$	0.0%	0.7%	Dry SGTR with isolation
	3	$1.65 \cdot 10^{-8}$	0.8%	70.4%	Induced SGTR with secondary water
	4	$2.18 \cdot 10^{-9}$	0.1%	9.3%	Containment rupture
	5	$1.81 \cdot 10^{-10}$	0.0%	0.8%	Containment leak
Total Early releases		$2.34 \cdot 10^{-8}$	**1.1%**		
Late Releases	6	$6.68 \cdot 10^{-9}$	0.3%	21.2%	Interfacing system LOCA
	7	$1.15 \cdot 10^{-10}$	0.0%	0.4%	SGTR without secondary water
	8	$1.14 \cdot 10^{-8}$	0.5%	36.2%	SGTR with secondary water
	9	$4.99 \cdot 10^{-9}$	0.2%	15.9%	Containment rupture
	10	$8.28 \cdot 10^{-9}$	0.4%	26.3%	Containment leak + Isolation failure
Total Late releases		$3.14 \cdot 10^{-8}$	**1.5%**		
Very Late Releases	11	$1.34 \cdot 10^{-12}$	0.0%	0.0%	ISLOCA + Isolation failure
	12	$1.55 \cdot 10^{-10}$	0.0%	0.0%	SGTR with and without secondary water
	13	$2.18 \cdot 10^{-9}$	0.0%	0.1%	Containment rupture and leak
	14	$2.32 \cdot 10^{-9}$	0.1%	0.1%	Basemat penetration
	15	$1.59 \cdot 10^{-6}$	74.9%	99.8%	Filter vented release
Total Very late releases		$1.59 \cdot 10^{-6}$	**75.0%**		
No release	16	$4.75 \cdot 10^{-7}$	22.4%	100%	No Containment failure
Total No release		$4.75 \cdot 10^{-7}$	**22.4%**		
All		$\mathbf{2.21 \cdot 10^{-6}}$			

Table 1.22 Summary of source term grouping

GROUP	Early Releases				
	STC-1	STC-2	STC-3	STC-4	STC-5
NOBL,IN	$9.60 \cdot 10^{-1}$	$9.13 \cdot 10^{-1}$	$7.96 \cdot 10^{-1}$	$9.90 \cdot 10^{-1}$	$9.99 \cdot 10^{-1}$
CSI	$1.90 \cdot 10^{-1}$	$1.60 \cdot 10^{-1}$	$3.00 \cdot 10^{-3}$	$3.70 \cdot 10^{-2}$	$3.05 \cdot 10^{-2}$
TEO2	$4.63 \cdot 10^{-8}$	N.A.[8]	N.A.	$3.33 \cdot 10^{-5}$	N.A.
SRO	$5.50 \cdot 10^{-2}$	$8.10 \cdot 10^{-4}$	$4.30 \cdot 10^{-5}$	$1.50 \cdot 10^{-3}$	$2.75 \cdot 10^{-3}$
MOO2	$5.00 \cdot 10^{-2}$	$1.59 \cdot 10^{-2}$	$1.31 \cdot 10^{-3}$	$7.50 \cdot 10^{-3}$	$1.85 \cdot 10^{-3}$
CSOH	$1.90 \cdot 10^{-1}$	$1.50 \cdot 10^{-1}$	$1.99 \cdot 10^{-3}$	$2.60 \cdot 10^{-2}$	$2.38 \cdot 10^{-2}$
BAO	$7.00 \cdot 10^{-2}$	$6.30 \cdot 10^{-3}$	$5.50 \cdot 10^{-4}$	$5.70 \cdot 10^{-3}$	$2.09 \cdot 10^{-3}$
LA2O3	$3.00 \cdot 10^{-2}$	$1.36 \cdot 10^{-5}$	$7.60 \cdot 10^{-6}$	$3.90 \cdot 10^{-4}$	$5.13 \cdot 10^{-4}$
CEO2	$5.70 \cdot 10^{-2}$	$2.46 \cdot 10^{-5}$	$7.30 \cdot 10^{-5}$	$2.30 \cdot 10^{-3}$	$2.37 \cdot 10^{-3}$
SB	$2.70 \cdot 10^{-1}$	$7.05 \cdot 10^{-2}$	$4.90 \cdot 10^{-3}$	$9.20 \cdot 10^{-2}$	$1.38 \cdot 10^{-1}$
TE2	$1.80 \cdot 10^{-2}$	$0.00 \cdot 10^{0}$	$0.00 \cdot 10^{0}$	$2.30 \cdot 10^{-2}$	$5.52 \cdot 10^{-2}$
UO2,ACT	$3.70 \cdot 10^{-6}$	$0.00 \cdot 10^{0}$	$0.00 \cdot 10^{0}$	$1.80 \cdot 10^{-5}$	$2.49 \cdot 10^{-5}$

Table 1.23 Summary of Fission Product Release Fractions (Early release)

[8] N.A. is not applicable

GROUP	Late Releases				
	STC-6	STC-7	STC-8	STC-9	STC-10
NOBL,IN	$9.60 . 10^{-1}$	$9.90 . 10^{-1}$	$1.00 . 10^{0}$	$9.90 . 10^{-1}$	$8.00 . 10^{-1}$
CSI	$2.00 . 10^{-2}$	$5.40 . 10^{-1}$	$4.91 . 10^{-2}$	$4.50 . 10^{-2}$	$1.56 . 10^{-2}$
TEO2	$6.22 . 10^{-7}$	N.A.	N.A.	$0.00 . 10^{0}$	N.A.
SRO	$3.35 . 10^{-4}$	$1.40 . 10^{-4}$	$2.90 . 10^{-2}$	$2.70 . 10^{-3}$	$1.09 . 10^{-3}$
MOO2	$1.34 . 10^{-3}$	$9.70 . 10^{-4}$	$1.70 . 10^{-2}$	$1.20 . 10^{-2}$	$4.96 . 10^{-4}$
CSOH	$1.03 . 10^{-2}$	$5.90 . 10^{-1}$	$4.91 . 10^{-2}$	$2.40 . 10^{-2}$	$6.09 . 10^{-3}$
BAO	$1.36 . 10^{-3}$	$3.80 . 10^{-4}$	$1.20 . 10^{-2}$	$1.00 . 10^{-2}$	$9.52 . 10^{-4}$
LA2O3	$2.25 . 10^{-4}$	$2.60 . 10^{-5}$	$3.10 . 10^{-3}$	$5.90 . 10^{-4}$	$1.74 . 10^{-4}$
CEO2	$4.71 . 10^{-4}$	$1.90 . 10^{-4}$	$5.70 . 10^{-3}$	$4.80 . 10^{-3}$	$6.35 . 10^{-4}$
SB	$4.30 . 10^{-2}$	$4.80 . 10^{-2}$	$2.60 . 10^{-2}$	$1.40 . 10^{-1}$	$5.35 . 10^{-2}$
TE2	$8.35 . 10^{-6}$	$1.90 . 10^{-1}$	$6.00 . 10^{-3}$	$3.50 . 10^{-2}$	$1.42 . 10^{-2}$
UO2,ACT	$2.19 . 10^{-9}$	$2.20 . 10^{-6}$	$1.90 . 10^{-7}$	$2.20 . 10^{-5}$	$4.29 . 10^{-6}$

Table 1.24 Summary of Fission Product Release Fractions (Late release)

	Very Late Releases					No Releases
	STC-11	STC-12	STC-13	STC-14	STC-15	STC-16
NOBL,IN	$9.90 . 10^{-1}$	$4.50 . 10^{-1}$	$7.20 . 10^{-1}$	$7.6 . 10^{-1}$	$7.05 . 10^{-1}$	$9.00 . 10^{-3}$
CSI	$1.23 . 10^{-1}$	$5.50 . 10^{-2}$	$1.40 . 10^{-2}$	$1.56 . 10^{-2}$	$7.95 . 10^{-3}$	$1.40 . 10^{-8}$
TEO2	$5.65 . 10^{-3}$	$0.00 . 10^{0}$	$5.33 . 10^{-4}$	N.A.	N.A.	N.A.
SRO	$1.68 . 10^{-3}$	$4.50 . 10^{-6}$	$6.20 . 10^{-4}$	$1.09 . 10^{-3}$	$6.23 . 10^{-4}$	$4.60 . 10^{-12}$
MOO2	$1.39 . 10^{-2}$	$3.80 . 10^{-5}$	$2.10 . 10^{-6}$	$4.96 . 10^{-4}$	$1.95 . 10^{-4}$	$6.40 . 10^{-11}$
CSOH	$1.20 . 10^{-1}$	$5.60 . 10^{-2}$	$7.00 . 10^{-3}$	$6.09 . 10^{-3}$	$1.83 . 10^{-3}$	$1.40 . 10^{-8}$
BAO	$5.2 . 10^{-3}$	$2.20 . 10^{-5}$	$5.30 . 10^{-4}$	$9.52 . 10^{-4}$	$5.01 . 10^{-4}$	$1.80 . 10^{-10}$
LA2O3	$2.13 . 10^{-4}$	$1.30 . 10^{-8}$	$1.00 . 10^{-4}$	$1.74 . 10^{-4}$	$9.54 . 10^{-5}$	$3.20 . 10^{-12}$
CEO2	$6.60 . 10^{-4}$	$1.50 . 10^{-8}$	$2.60 . 10^{-4}$	$6.35 . 10^{-4}$	$2.61 . 10^{-4}$	$2.00 . 10^{-13}$
SB	$1.57 . 10^{-1}$	$4.20 . 10^{-3}$	$7.80 . 10^{-2}$	$5.35 . 10^{-2}$	$2.96 . 10^{-2}$	$8.80 . 10^{-9}$
TE2	$5.40 . 10^{-3}$	$0.00 . 10^{0}$	$5.50 . 10^{-4}$	$4.52 . 10^{-3}$	$1.22 . 10^{-3}$	$5.80 . 10^{-11}$
UO2,ACT	$1.00 . 10^{-6}$	$0.00 . 10^{0}$	$1.60 . 10^{-6}$	$4.29 . 10^{-6}$	$1.86 . 10^{-6}$	$3.20 . 10^{-9}$

Table 1.25 Summary of Fission Product Release Fractions (Very late release)

STC-1: Loss of offsite power transient with induced SGTR, containment does not fail, controlled release

STC-2: Similar as STC-1, but SG isolation is performed as an Accident Management measure

STC-3: SGTR and ATWS with feedwater available to keep SG filled (scrubbing) as an AM measure

STC-4: Small LOCA with no injection, early containment failure

STC-5: Small LOCA with containment isolation failure initially

STC-6: Interfacing system LOCA (event V) including extended release path

STC-7: SGTR with failed open secondary relief valve

STC-8: Similar to STC-7 but with auxiliary feedwater to keep SG filled (scrubbing) as an AM measure

STC-9: Small LOCA with injection, early containment failure

STC-10: Small LOCA, containment failure due to hydrogen deflagration-small leak

STC-11: Early midloop, vents closed, interfacing system LOCA, including extended release path

STC-12: SGTR at restart after core load

STC-13: Loss of RHR flow, early midloop, open vessel

STC-14 Basemat melt through, assumed identical to STC-10

STC-15: Small LOCA, containment vent, reduction due to filter not incorporated

STC-16: No containment failure, release design-leakage

1.5.2.3 Results of the Level-3 PSA

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

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

The societal risk is defined as the risk of 10 or more casualties, which are directly attributable to the accident, and this risk shall be lower than 10^{-5} per year for 10 deaths, 10^{-7} per year for 100 deaths, 10^{-9} per year for 1,000 deaths, etc.

Ad 1. Total lifetime individual risk

The total lifetime individual risk for all source terms is shown in Figure 1.7. Furthermore, the maximum individual risk limit (10^{-6} per year) is shown on this figure.

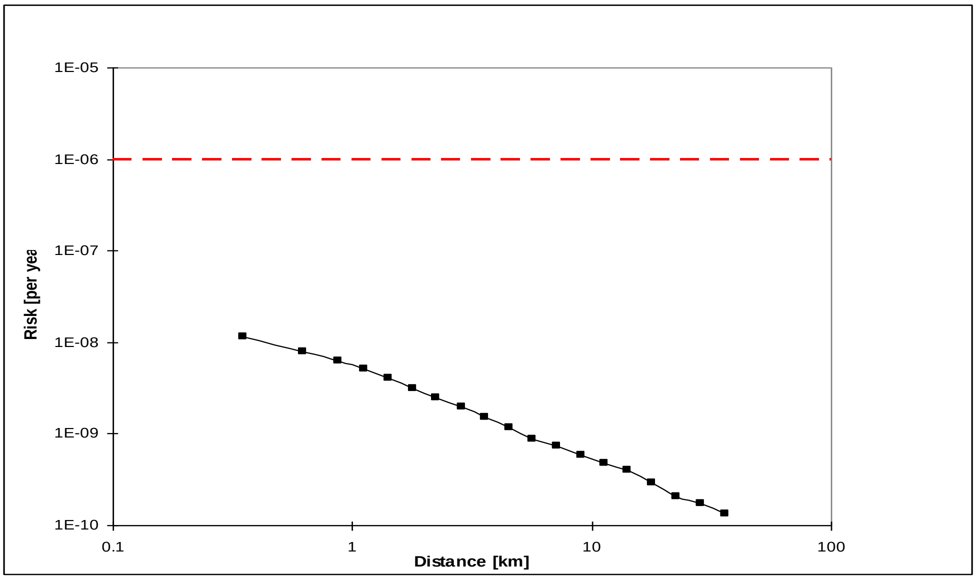


Figure 1.8 Total lifetime individual risk

It can be seen that the individual risk is well below the limit of 10^{-6} per year.

Societal risk

The societal risk criterion (acceptable level) is shown in red in Figure 1.9. The probability of exceeding certain early deaths as quantified in Level-3 PSA for Borssele NPP is also shown.

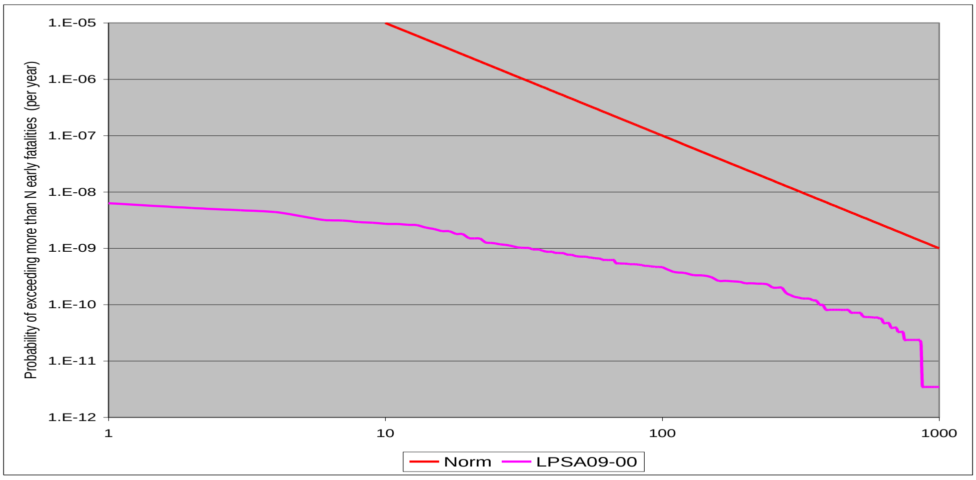


Figure 1.9 CCDF of early fatalities

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

1.6 Future use of Mixed Oxide fuel

EPZ has the intention to use Mixed Oxide fuel (MOX) in the NPP Borssele in the near future. For that reason EPZ has applied for a license to use of MOX fuel elements [9] with maximum 5.41% (w/w) fissionable Plutonium (Pu_{fiss}). The maximum allowed number of MOX fuel elements in the reactor applied for equals 48 (40%).

During the licensing procedure it is shown by profound analyses that the safety of the NPP with the use of MOX fuel is comparable with the current situation in which Enriched Natural Uranium oxide (ENU) is used as fuel. The consequences for man and environment also turn out to be comparable for both kinds of nuclear fuel. The main results of the performed safety analyses are presented In Table 1.26 to Table 1.29.

[9] The licensing process has not been completely finished

Design of the MOX fuel element

In each MOX fuel element the percentage Pu_{fiss} in the separate fuel rods will be: ≤ 2.6% (w/w) in 12 rods, ≤ 3.6% (w/w) in 56 rods and ≤ 6.4% (w/w) in 137 rods. The maximum allowed number of MOX fuel elements in the reactor will be 48 (40%). By analyses it is demonstrated that the design of the MOX fuels rods for future use by the NPP Borssele fulfills all required design criteria. This means that the occurring loads to the rods during normal operation and incidents are acceptable. The results of the above mentioned analyses are shown in Table 1.26.

Design criterium	Design parameter	unit	Design criterium	Calculated value
Fuel temperature	Melting temperature minus fuel temperature	(K)	≥ 0	336
Internal rod pressure	Maximum internal pressure cladding	(bar)		173
	Creap rate cladding	(10^{-4} %/h)	≤ 1.0	0.08
	Integral creap tension cladding	(%)	≤ 0,3	0.01
	Tangential tension cladding	N/mm^2	≤ 100	4
Deformation	Tangential deformation	(%)	≤ 1.0	0.75
	Plastic deformation	(%)	≤ 3.5	2.3
Corrosion and hydrogen uptake	Oxide layer thickness	(µm)	≤ 100	48
Stresses	Safety factor for stress in cladding and weld		≥ 1.00	1.03
	Safety factor for tension by dynamic loads		≥ 1.0	7.1
Deformation bij external overpressure	Safety factor for elastic bending		≥ 1.00	3.4
	Safety factor for plastic deformation		≥ 1.00	1.07
Enthalpy rise	Enthalpy rise by emission of control rod	(cal/g)	< 60 - 170	30

Table 1.26 Results of the analyses of the mechanical and thermodynamical behaviour of the MOX fuel rods

Feasibility study for the use of MOX in NPP Borssele

In a feasibilty study for the use of MOX in NPP Borssele it is shown that the reactor during normal operation and incidents, during the equilibrium cycle and all transition cycles, can be operated in a safe way. For the equilibrium cycle the calculated results for the main safety parameters are shown in Table 1.27 together with their design criteria.

Design criterium	Unit	Design criterium	Calculated value
Maximum hot channel factors			
Fxy			1.68
Fq		≤ 2.80	2.51
FH		≤ 1.80	1.69
FZ		≤ 1.44	1.28
$Q'_{max,\ xyz}$	(W/cm)	≤ 568	509
Control rod worth (zero power, 294 °C, 324 FPD[10], Xe-equilibrium)	(%)	≤ -5	-5.59
Moderator temperature coefficient (0 FPD, 294 °C, Xe-free)	(pcm/K)	< 0	Max -20.8
Departure from Nucleate Boiling Ratio (324 FPD, main steamline rupture)		≥ 1.45	2.03
Maximum fuel temperature after main steam line rupture	(°C)	≤ 2,650	2,340

Table 1.27 Calculated values for the main safety parameters for the use of MOX in NPP Borssele

[10] FPD is full power day

Design basis accidents

In general design basis accidents do not result in releases of radioactivity to the environment as the design of the plant is based on the control of these kind of accidents and therefore of the confinement of the radioactivity. However, some special design basis accidents can give rise to a radioactive release exceeding the releases due to normal operation. By means of a radiological analysis the consequences of these releases are determined and is shown that they are within required limits (see Table 1.28).

Postulated Initiating Events		ENU (mSv)		MOX (mSv)		Doselimit (mSv)	
		E[11]	Hth[12]	E	Hth	E	Hth
1.5.1	Long lasting loss of main heat sink during operational leakage of steamgenerator tubes	0.024	0.466	0.025	0.475	0.4	500
7.2.2	Unintended opening of primary pressure relief valve	0.00080	0.0143	0.00082	0.147	4	500
7.2.3	Rupture of reactor coolant line	0.684	4.50	0.695	4.58	40	500
7.3.2.2	Steam generator tube rupture combined with temporary emergency power supply situation	0.19	3.40	0.19	3.48	4	500
7.4.2	Leak in a measuring line containing primary coolant outside of containment	0.12	2.42	0.13	2.47	4	500
8.2	Leak in the radioactive off-gas system piping	0.0099	0.0111	0.0100	0.011	0.04	500
8.4.1	Fuel element damage during handling	0.0097	0.191	0.0099	0.196	0.4	500
9.1.2	Earthquake effect on reactor building	0.20	0.717	0.20	0.733	4	500

Table 1.28 Evaluation of the dose consequences of those design base accidents that give rise to a radioactive release. The results for 40% MOX have been compared with those for enriched natural uranium elements.

[11] E is Effective dose
[12] H_{th} is Thyroid dose

Beyond-design accidents

In the Level-1 analysis of the PSA the core damage frequency is evaluated. For the core damage frequencies of MOX (40%) and ENU, see Table 1.29.

The radiological consequences of the beyond-design accidents are calculated in level 3 of the probabilistic safety assessment (PSA level-3). The results of the PSA level-3 calculations are evaluated against the criteria provided by the Dutch regulations. The results show that large margins exist. Furthermore, a comparison between the results for MOX and those for ENU show that the introduction of MOX only slightly influences the radiological consequences of the beyond-design accidents (Table 1.29). For more detailed information about ENU one is referred to 1.5.2.3.

	ENU	MOX (40%)	Criterium
Core damage frequency (y^{-1})	$2.12 \cdot 10^{-6}$	$2.12 \cdot 10^{-6}$	n.a.
Maximum individual risk (y^{-1})	$1.9 \cdot 10^{-8}$	$2.0 \cdot 10^{-8}$	$1 \cdot 10^{-6}$
Group risk 10 fatalities (y^{-1})	$5.6 \cdot 10^{-9}$	$5.6 \cdot 10^{-9}$	$1 \cdot 10^{-5}$

Table 1.29 Evaluation of the dose consequences of the beyond-design accidents.The results for MOX have been compared with those for enriched natural uranium elements.

Annex 1.1. System code description for the 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 H_2-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 CO_2 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

Annex 1.2. Building code description for the relevant buildings

Building number	Name
01	Containment
02	Reactor Building Inside: Annulus
03	Nuclear Auxiliary Building
04	Turbine building
05	Electrical Building (Alt: Switchgear Building)
06	Access Building
09	Condensate Plant
10	Diesel generator building
11	Generator Transformer
12	Auxiliary Transformer
13	Ventilation Stack
21	Cooling Water Inlet Building
23 33	Cooling Water Outlet Building Backup Systems Bunker
34	Radioactive Waste Storage
35	Building
41	Remote Shutdown Building Auxiliary Transformer
48	Fire Station
72	Diesel generator building

Annex 1.3. Prediction of the Source Term for scenarios that lead to core melt

With respect to the source terms for scenarios that lead to core melt, KCB is using the source term category tree from the PSA level 2, see Figure 1.10.

The source term is determined based on the pre-calculated data for the selected accident sequences. The selection of the most appropriate sequence is made by the use of a decision tree on paper. The top events of this tree consist of phenomenological events or processes, and consequential system failures or functions resulting from physical phenomena, human actions or the accident environment. The endpoints of this tree represent the source term groups. The specific source term data are available (pre-calculated) for each source term group based on deterministic accident simulations (using e.g. the MAAP code).

For scenarios with occurrence of core melt there are different time frames associated with the releases from the containment:

- Early release: release between 0 and 12 hours following reactor trip or shutdown;
- Late release: release between 12 and 72 hours following reactor trip or shutdown;
- Very late release: release greater than 72 hours after reactor trip or shutdown.

Furthermore there is a source term group characterized as the design leakage of the containment, so no additional leakage beyond the normally allowed limit (source term 16).

The source terms within each of the time frames are specified below.

For the early releases the following source terms are recognized:

- a containment bypass due to a steam generator tube rupture (SGTR). Distinction is made between a SGTR without water in the secondary side (source term category 1 and 2 for a SGTR without isolation and with isolation respectively) and a SGTR with water in the secondary side (source term category 3);
- a containment rupture (source term category 4);
- a containment leak (source term category 5).

For the late releases the following source terms are recognized:

- a containment bypass due to an interfacing system LOCA sequence (source term category 6);
- a SGTR without water in the secondary side (source term category 7);
- a SGTR with water in the secondary side (source term category 8);
- a containment rupture (source term category 9);
- a containment leak (source term category 10).

For the very late releases the following source terms are recognized:

- a containment bypass due to an interfacing system LOCA sequence (source term category 11);
- a SGTR (source term category 12);
- a containment rupture (source term category 13);
- a containment leak (source term category 14);
- a release from the containment by use of the containment filtered vent system TL003 (source term category 15).

The calculated source term is provided in the format adjusted to the input requirements of the off-site simulation code COSYMA. In this format data include the information on the beginning, magnitude and duration of release for each nuclide group, elevation of the release point, temperature and energy of release. COSYMA calculates the off-site consequences.

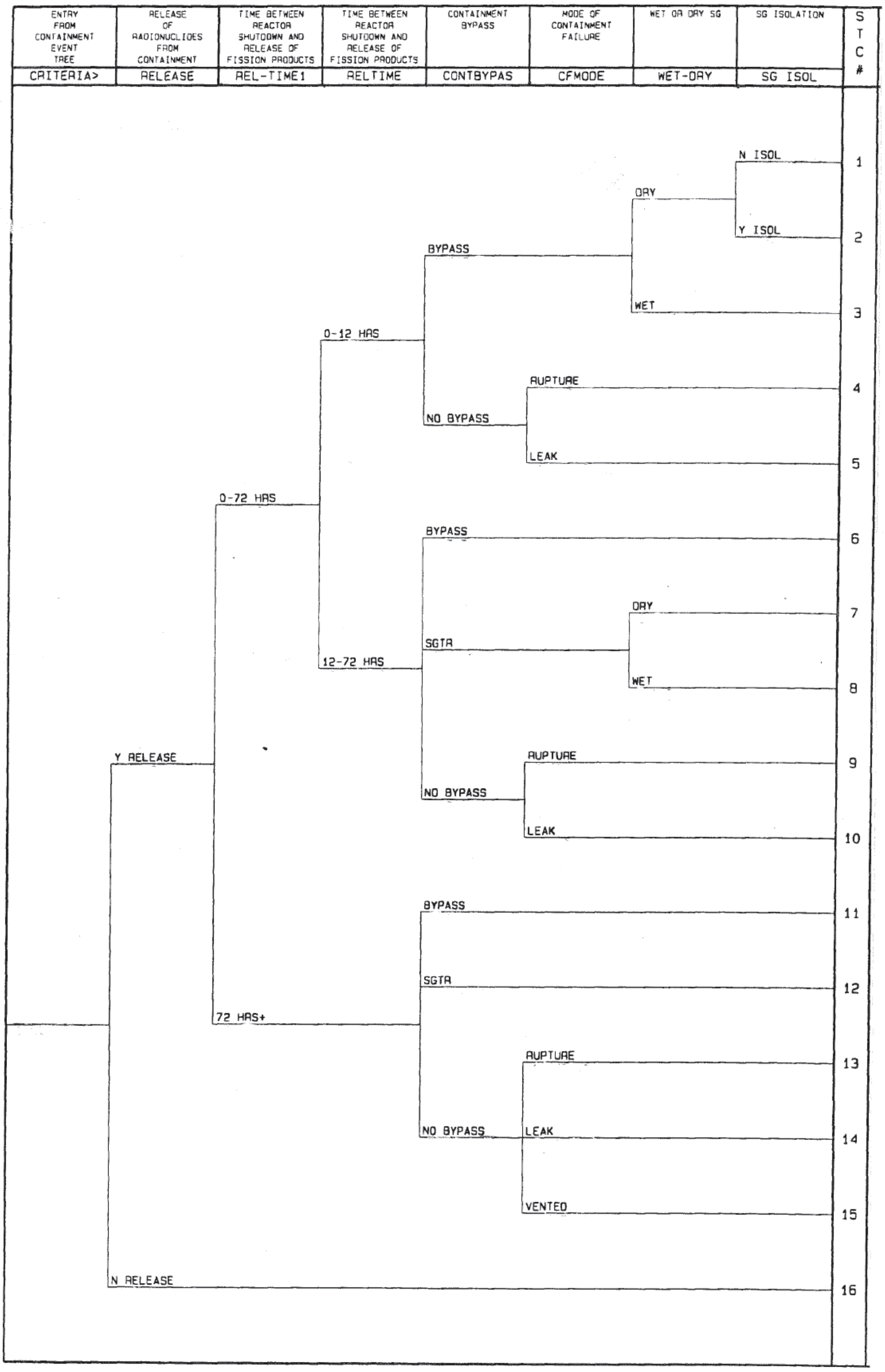


Figure 1.10 Source Term category tree

Annex 1.4. Inventories of the nuclides in the reactor core immediately after shut-down for ENU and 40% MOX

Nuclide	MOX-40% (Bq)	ENU (Bq)		Nuclide	MOX-40% (Bq)	ENU (Bq)
Kr-83m	1.36E+17	1.55E+17		I-131	1.28E+18	1.28E+18
Kr-85m	3.2E+17	3.74E+17		I-132	1.78E+18	1.81E+18
Kr-85	1.61E+16	1.81E+16		I-133	2.51E+18	2.57E+18
Kr-87	5.42E+17	6.41E+17		I-134	2.71E+18	2.79E+18
Kr-88	7.33E+17	8.70E+17		I-135	2.41E+18	2.46E+18
Rb-86	2.36E+15	2.68E+15		Xe-131m	1.38E+16	1.38E+16
Rb-88	7.56E+17	8.95E+17		Xe-133m	7.63E+16	7.77E+16
Rb-89	1.02E+18	1.22E+18		Xe-133	2.54E+18	2.60E+18
Sr-89	1.07E+18	1.29E+18		Xe-135m	5.39E+17	5.36E+17
Sr-90	1.16E+17	1.33E+17		Xe-135	7.68E+17	5.98E+17
Sr-91	1.35E+18	1.58E+18		Xe-138	2.16E+18	2.26E+18
Sr-92	1.47E+18	1.68E+18		Cs-134m	6.05E+16	6.39E+16
Sr-93	1.67E+18	1.86E+18		Cs-134	2.62E+17	2.60E+17
Y-90m	9.41E+13	1.12E+14		Cs-135	8.38E+11	6.46E+11
Y–90	1.21E+17	1.39E+17		Cs-136	7.04E+16	6.06E+16
Y-91m	6.92E+17	8.10E+17		Cs-137	1.86E+17	1.83E+17
Y–91	1.44E+18	1.70E+18		Cs-138	2.37E+18	2.47E+18
Y–92	1.48E+18	1.69E+18		Ba-139	2.18E+18	2.30E+18
Y–93	1.70E+18	1.89E+18		Ba-140	2.15E+18	2.26E+18
Zr-89	6.34E+10	7.68E+10		La-140	2.23E+18	2.34E+18
Zr-93	2.86E+12	3.06E+12		La-141	1.99E+18	2.08E+18
Zr-95	2.09E+18	2.27E+18		La-142	1.94E+18	2.03E+18
Zr-97	2.02E+18	2.12E+18		Ce-141	2.06E+18	2.16E+18
Nb-93m	2.22E+11	2.23E+11		Ce-143	1.87E+18	2.00E+18
Nb-94m	2.01E+12	1.95E+12		Ce-144	1.60E+18	1.73E+18
Nb-94	4.11E+08	3.71E+08		Pr-143	1.87E+18	2.01E+18
Nb-95m	2.32E+16	2.52E+16		Pr-145	1.29E+18	1.36E+18
Nb-95	2.11E+18	2.29E+18		Nd-147	8.22E+17	8.48E+17
Nb-97	2.03E+18	2.13E+18		Pm-147	2.79E+17	2.79E+17
Mo-99	2.33E+18	2.40E+18		Pm-148m	5.19E+16	4.58E+16
Mo-101	2.21E+18	2.22E+18		Pm-148	2.15E+17	2.23E+17
Tc-99m	2.04E+18	2.10E+18		Pm-149	6.58E+17	6.79E+17
Tc-99	2.33E+13	2.30E+13		Pm-151	2.54E+17	2.40E+17
Tc-101	2.21E+18	2.22E+18		Eu-152m	1.89E+14	1.22E+14
Ru-103	2.08E+18	1.92E+18		Eu-152	5.22E+12	1.63E+12

Nuclide	MOX-40% (Bq)	ENU (Bq)		Nuclide	MOX-40% (Bq)	ENU (Bq)
Ru-105	1.53E+18	1.32E+18		Eu-154	1.66E+16	1.43E+16
Ru-106	9.07E+17	6.88E+17		Eu-155	8.90E+15	6.54E+15
Rh-103m	2.08E+18	1.92E+18		Eu-156	3.04E+17	2.90E+17
Rh-105	1.47E+18	1.25E+18		Po-210	9.49E+02	1.00E+03
Ag-108m	5.59E+09	2.81E+09		Ra-226	2.99E+04	3.52E+04
Ag-110m	7.16E+15	4.70E+15		U-234	1.57E+12	2.06E+12
Ag-110	2.16E+17	1.53E+17		U-235	2.83E+10	3.95E+10
Ag-111	8.72E+16	7.00E+16		U-238	4.43E+11	4.50E+11
Sb-124	1.59E+15	1.25E+15		Np-237	3.59E+11	4.27E+11
Sb-125	2.64E+16	1.97E+16		Np-238	3.47E+17	4.33E+17
Sb-126	1.15E+15	9.95E+14		Np-239	2.17E+19	2.33E+19
Sb-127	1.31E+17	1.14E+17		Pu-236	3.18E+11	3.32E+11
Sb-128l	1.86E+16	170E+16		Pu-238	1.69E+16	3.74E+15
Sb-129	3.83E+17	3.66E+17		Pu-239	9.33E+14	4.53E+14
Sb-130l	4.41E+17	4.45E+17		Pu-240	2.13E+15	6.02E+14
Sb-131	1.04E+18	1.06E+18		Pu-241	4.50E+17	1.46E+17
Te-125m	5.66E+15	4.17E+15		Pu-242	1.36E+13	2.71E+12
Te-127m	1.2E+16	8.74E+15		Am-241	1.63E+15	1.67E+14
Te-127	1.19E+17	1.02E+17		Am-242m	7.93E+13	5.76E+12
Te-129m	6.98E+16	6.67E+16		Am-242	4.00E+17	9.14E+16
Te-129	4.13E+17	3.93E+17		Am-243	1.24E+14	2.71E+13
Te-131m	1.88E+17	1.83E+17		Cm-242	2.95E+17	5.49E+16
Te-131	1.13E+18	1.13E+18		Cm-243	1.50E+14	2.04E+13
Te-132	1.75E+18	1.78E+18		Cm-244	2.21E+16	4.14E+15
Te-133m	1.3E+18	1.33E+18		Cm-245	3.14E+12	3.68E+11
Te-133	1.23E+18	1.28E+18		Cm-246	8.23E+11	1.41E+11
Te-134	2.17E+18	2.28E+18		Cm-247	3.68E+06	4.93E+05
I-129	5.87E+10	5.19E+10		Cm-248	1.43E+07	1.98E+06
I-130	3.66E+16	3.82E+16				

Chapter 2 Earthquake

2.1 Design basis

2.1.1 Earthquake against which the plant is designed

The analysis of the design-basis earthquake (DBE) for the KCB site has been performed posterior to the plant's construction and commissioning in the framework of the second 10 yearly safety evaluation.

The KCB is located in a region with low seismic hazard; there appear not to be any significant source regions within a radius of 50 km. More distant source regions with the strongest contribution to the site's seismic hazard are the Belgian zone and the Lower Rhine Graben (see Figure 2.1).

Figure 2.1 Seismotectonic units relevant for KCB)

2.1.1.1 Characteristics of the design basis earthquake (DBE)

The highest earthquake intensity ever recorded in the area of the KCB was approximately V½ MMI (Modified Mercalli Intensity), 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 documented earthquakes in the period between 217 and 1990 AD. For the DBE this intensity level was increased with one unit, in accordance with IAEA (1979), so the DBE was established at VI½ MMI.

The peak ground acceleration (**PGA)** of this DBE was first specified at **0.98 m/s² (0.1 g)** This PGA was used in for defining the ground response spectrum, obtained by scaling the USAEC spectrum, i.e. an 84% fractile spectrum (see Figure 2.2). With this definition of the DBE, a first set of analyses was performed by Belgatom for EPZ to demonstrate the stability and integrity of the structures and equipment.

In the framework of the second 10 yearly safety evaluation, the definition of the DBE was updated and now derives the ground response spectrum directly from the intensity of the design-basis earthquake (VI½ MMI).

Two spectra (in the sequel called ‘Hosser spectra’) were defined in this process (see Figure2.2):

- one for alluvial ground conditions (on the left in Figure 2.2), to be used if the input ground motion is applied at ground level; the corresponding PGA is **0.6 m/s²**;
- one for medium ground conditions (on the right in Figure 2.2), to be used if the input ground motion is applied at the basis of the pile foundation; the corresponding PGA is **0.75 m/s²**.

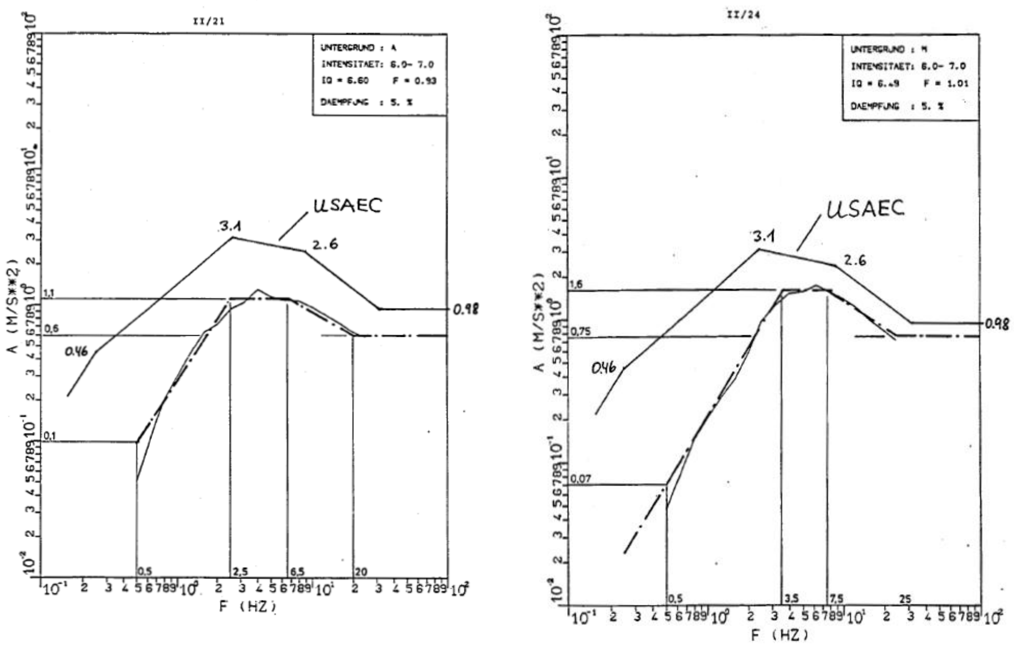


Figure2.2 USAEC spectrum scaled to 0.98 m/s² PGA and intensity-based site-specific spectra for alluvial (left) and medium (right) ground

The Hosser spectra were also modified to account for the effects of larger earthquakes at greater distances. This did **not lead to any change in the PGA**; however, it did lead to an extension of the plateau region of the spectra to the left (low-frequency region), as shown in Figure 2.3.

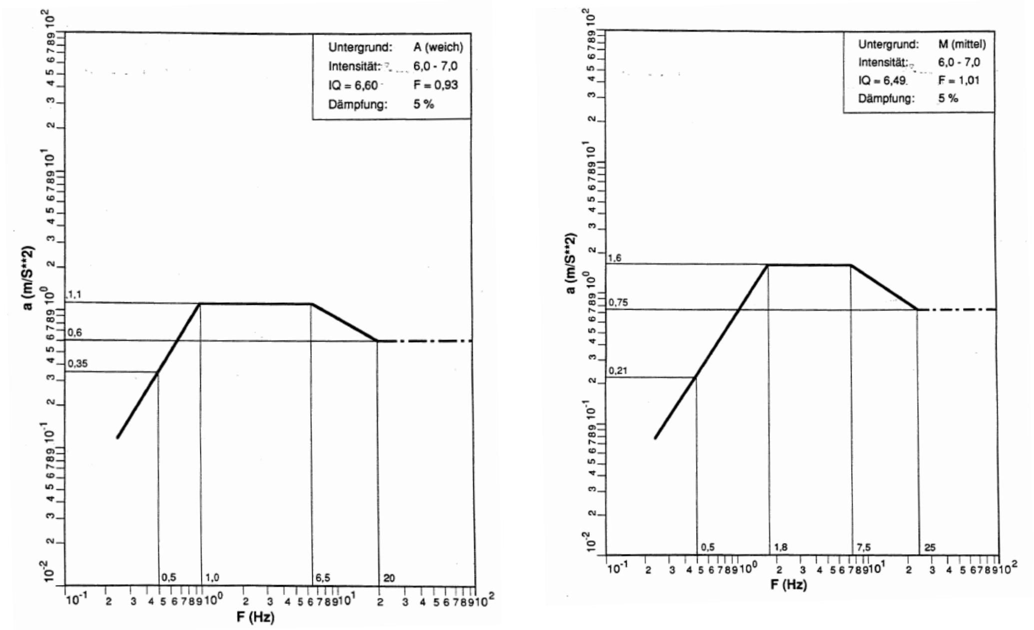


Figure 2.3 Modified Hosser spectra used in the second 10 yearly safety evaluation

These **modified Hosser spectra** were subsequently used for the seismic evaluation in the second 10 yearly safety evaluation.

NB: In the media earthquakes are always characterized by their magnitude on the Richter scale. This earthquake constant expresses the energy released by the earthquake, but says nothing about the impact of the earthquake on the site location, since the site may be many kilometers (and many different earth layers) removed from the source of the earthquake. Only in the epicentre (the point on the earth's surface, directly above the earthquake) a rough comparison between intensity (MMI) and the Richter scale can be given (see Table 2.1).

MMI value	Summary description	Full description	Magnitude (Richter)
I	Not mapped	Not felt.	1.9
II	Not mapped	Felt by people sitting or on upper floors of buildings.	2.5
III	Not mapped	Felt by almost all indoors. Hanging objects swing. Vibration like passing of light trucks. May not be recognized as an earthquake.	3.1
IV	Not mapped	Vibration felt like passing of heavy trucks. Stopped cars rock. Hanging objects swing. Windows, dishes, doors rattle. Glasses clink. In the upper range of IV, wooden walls and frames creak.	3.7
V	Light	Felt outdoors. Sleepers wakened. Liquids disturbed, some spilled. Small unstable objects displaced or upset. Doors swing. Pictures move. Pendulum clocks stop.	4.4
VI	Moderate	Felt by all. People walk unsteadily. Many frightened. Windows crack. Dishes, glassware, knickknacks, and books fall off shelves. Pictures off walls. Furniture moved or overturned. Weak plaster, adobe buildings, and some poorly built masonry buildings cracked. Trees and bushes shake visibly.	4.9
VII	Strong	Difficult to stand or walk. Noticed by drivers of cars. Furniture broken. Damage to poorly built masonry buildings. Weak chimneys broken at roof line. Fall of plaster, loose bricks, stones, tiles, cornices, unbraced parapets and porches. Some cracks in better masonry buildings. Waves on ponds.	5.5
VIII	Very strong	Steering of cars affected. Extensive damage to unreinforced masonry buildings, including partial collapse. Fall of some masonry walls. Twisting, falling of chimneys and monuments. Wood-frame houses moved on foundations if not bolted; loose partition walls thrown out. Tree branches broken.	6.1
IX	Violent	General panic. Damage to masonry buildings ranges from collapse to serious damage unless modern design. Wood-frame structures rack, and, if not bolted, shifted off foundations. Underground pipes broken.	6.7
X	Very violent	Poorly built structures destroyed with their foundations. Even some well-built wooden structures and bridges heavily damaged and needing replacement. Water thrown on banks of canals, rivers, lakes, etc.	7.3
XI	Not mapped because these intensities are typically limited to areas with ground failure	Rails bent greatly. Underground pipelines completely out of service.	7.9
XII	Not mapped because these intensities are typically limited to areas with ground failure	Damage nearly total. Large rock masses displaced. Lines of sight and level distorted. Objects thrown into the air.	8.5

Table 2.1 Rough comparison between intensity (MMI) and the Richter scale, valid at the epicentre of the earthquake

2.1.1.2 Methodology used to evaluate the design-basis earthquake

Methodology for defining the intensity of the DBE

Firstly, it should be noted that both definitions of the DBE are based on the prior definition of the **intensity** of the DBE, i.e. an intensity of VI½ in the Modified Mercalli Intensity scale (MMI). This definition of the intensity of the DBE is conservative, since the largest intensity observed in a greater radius around the KCB site was V½ after the earthquake near Tournai (Belgium) on 11 June 1938. This definition of the DBE intensity hence involves a margin of one unit on the MMI scale, compared to the historical observation.

The median 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}/yr.

Methodology for defining the PGA of the DBE

In the original definition of the DBE , the PGA has been derived using correlation formulas between the intensity and the PGA. The adopted correlation formula (by Krinitzsky and Chang) was evaluated using the intensity selected for the DBE (VI½ MMI).

This method for defining the PGA in a generic (i.e. not site-specific) way, is considered to introduce significant conservatism , since it does not account for the seismically favourable site conditions (small shear wave velocity and consequently high material damping).

Since the PGA of the DBE is derived from the intensity, the results on the return period and the exceedance frequency apply equally: the median return period of the PGA of the DBE is 30,000 years, and the median frequency of exceedance is approximately 3.10^{-5}/yr. This has been adopted in the seismic hazard curve of the PSA , which is based on the original determination of the DBE ,(Figure 2.4).

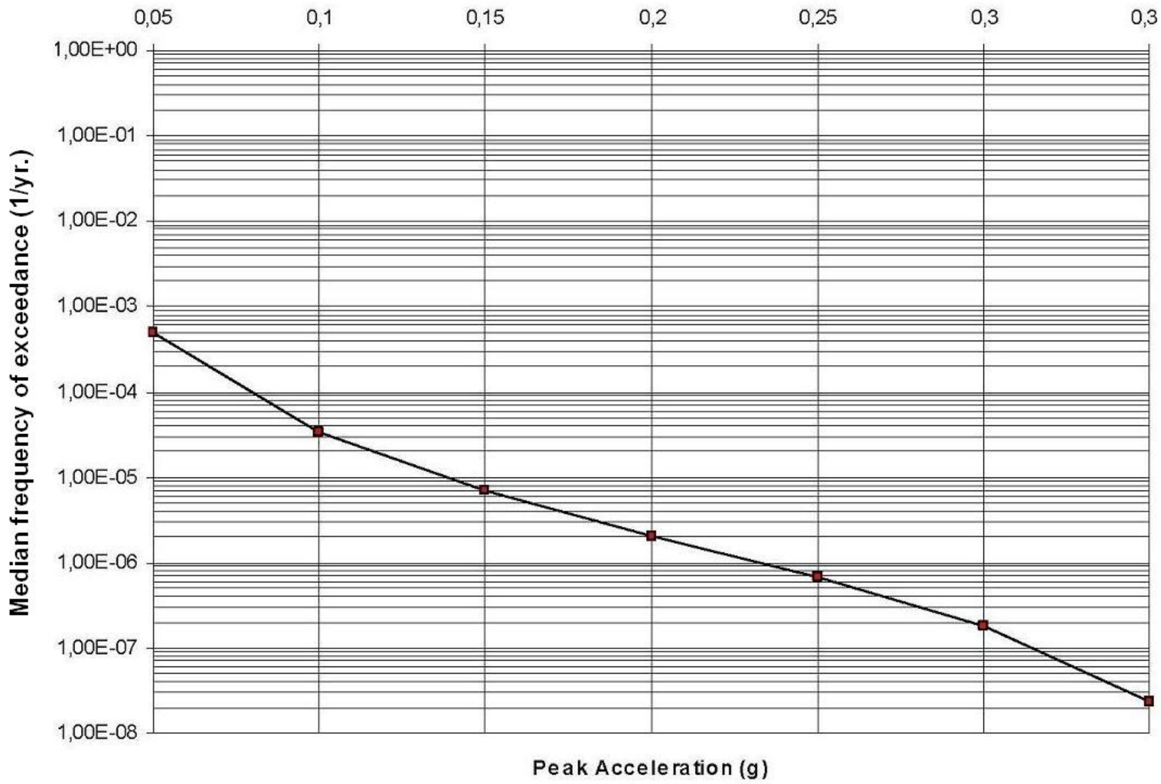


Figure 2.4 Seismic hazard curve: median frequency of exceedance (1/yr.) vs. peak ground acceleration (g)

In the subsequent definition of the DBE the PGA was not quantified by itself, but as by-product of the ground response spectrum, which was derived directly from the intensity of the DBE (see the following subsection: Methodology for defining the ground response spectrum of the DBE).

This latter approach has the advantage that the site conditions are taken into account. It leads to a reduced PGA (0.6 m/s² for the ground level and 0.75 m/s² for the level at the basis of the pile foundation).

Methodology for defining the ground response spectrum of the DBE

In the original definition of the DBE , the ground response spectrum was obtained by scaling a standard (generic) spectrum, namely the USAEC spectrum (an 84% fractile spectrum), to the PGA of the DBE (0.98 m/s²). This approach was considered to be inadequate for KCB for the following reasons :

- the USAEC spectra are based on seismic hazard conditions in the (Western) United States, which are totally different from those in Central Europe;
- 84% fractiles are unrealistically rich in energy, due to the enveloping process inherent in the underlying statistical analysis;
- 84% fractiles are highly sensitive to statistical uncertainties;
- the ground conditions at the site are not taken into account.

In the subsequent definition of the DBE , the ground response spectra were obtained directly from the intensity of the DBE, by applying a methodology developed in Germany by D. Hosser that was suitable for regions of low to moderate seismicity, and thus relevant for KCB. This method has been calibrated using numerous ground motion records of historical seismic events, mainly in Europe, and takes the characteristics of the ground at the site into account. Furthermore, the resulting ground response spectra are median spectra; hence the above-mentioned disadvantages of using 84% fractile spectra are avoided.

2.1.1.3 Conclusions on the adequacy of the design basis for the earthquake

As a preliminary remark, the following judgement on the adequacy of the design basis make strong reference to the German KTA. This is justified in view of the fact that the seismic conditions at KCB are generally comparable to the German seismic conditions (in particular those in the seismically calm region of Northern Germany).

The adequacy of the original definition of the DBE (PGA based on intensity of DBE and correlation formulas between PGA and intensity, and scaled USAEC spectrum) is judged as follows:

- the definition of the intensity of the DBE (VI½ MMI) is considered to be adequate, both on deterministic and probabilistic grounds:
  - deterministic: there is a margin of one unit in the MMI scale from the largest observed intensity (V½ MMI) and the DBE; this margin is expected to have led to a conservative definition of the DBE intensity, compared to the KTA 2201.2 , even in consideration of the fact that the deterministic KTA-approach involves the conservative assumption that the epicentre is located at the one point of its tectonic unit that is closest to the site ;
  - probabilistic: the median **return period** of 30,000 years of the DBE is in line with the requirements in the European national regulations on this matter. More specifically, this return period is located just in-between the less demanding national regulations (10,000 years in the UK and Switzerland) and the more demanding ones (100,000 years in Germany and Finland);

- the adopted, generic correlation formulas leading to the PGA (calculated directly from the intensity) do not credit the seismically favourable site conditions (small shear wave velocity and consequently high material damping). While this introduced conservatism, a more adequate approach is one that takes into account the site conditions.

The adequacy of the subsequent definition of the DBE ('modified Hosser spectra') is judged as follows:

- the definition of the intensity of the DBE (VI½ MMI) has not been modified compared to the original definition and hence the same judgement applies, i.e. the definition is considered to be adequate;
- the definition of the PGA (0.6 m/s^2 for the ground level and 0.75 m/s^2 for the level at the basis of the pile foundation) is considered to be adequate for the following reasons:

    - the site conditions are taken into account;
    - the numerical values of the PGA are compatible with the corresponding values at sites with seismically comparable conditions;
    - the PGA's are derived as a by-product of the ground response spectra; hence their adequacy is given – by virtue of the law of transitivity – by the adequacy of the ground response spectra, which is argued in the following paragraph;
    - the PGA of 0.75 m/s^2 is consistent with the new Eurocode 8 standard (EN 1998) according to which the site is situated in seismic hazard zone 'B', for which a PGA of 0.22 m/s^2 (return period of 475 years) has to be assumed. According to Figure 2.4 this corresponds to a PGA of 1 m/s^2 at a return period of 33,000 years. Therefore, the DBE is consistent with Eurocode 8;

- the definition of the site-specific ground response spectra directly from the intensity of the DBE is judged to be adequate for the following reasons:

    - generally speaking, the approach leading to the modified Hosser spectra is in line with the state of the art for central European sites with low-to-moderate seismicity. In this context it is important to note that this approach is compatible with the update of KTA 2201,2 for which the draft version was released in November 2010;
    - the ground conditions at the site are taken into account;
    - the method produces median site-specific spectra, which avoid the drawbacks of 84% spectra such as the USAEC spectra (excessive energy content, sensitivity to statistical uncertainties);
    - this method has been calibrated using ground motion records of mainly European seismic events, which may be viewed as more relevant for the KCB site.

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

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

The plant's protection against earthquakes is based on six levels of defence:

The first level of defence aims at preventing the challenge to the safety systems due to earthquakes. This prevention is firstly established by proper site selection. As described above, the site is characterised by a very low seismicity; the median seismic hazard curve (Figure 2.4) demonstrates for example that peak ground accelerations of 0.04 g (**inspection level** according to KTA 2201.6 are exceeded with a probability of about 1 in 1,000 years only.

Secondly, prevention is provided by an engineered design of all SSCs supporting normal and emergency plant operation. The design-basis earthquake intensity being defined at MMI[13] intensity VI½, a severe disturbance of normal plant operation needs generally not to be considered due to the following reasons:

All KCB buildings are engineered reinforced concrete or steel structures or steel structures designed for different static and dynamic load cases following well-established conventional or nuclear codes and standards and having significant potential for plastic deformation. Thus even the operational buildings can be assigned to vulnerability class C or higher (see Table 2.2).

Type of Structure		Vulnerability class					
		A	B	C	D	E	F
REINFORCED CONCRETE (RC)	frame without earthquake-resistant design (ERD)						
	frame with moderate level of ERD						
	frame with high level of ERD						
	walls without ERD						
	walls with moderate level of ERD						
	walls with high level of ERD						

Table 2.2 Differentiation of buildings in vulnerability classes

The European Macroseismic Scale (EMS) , presenting an improved version of the MSK scale, defines the expected damage grade at an intensity of VII as follows:

[13] In this range of intensity the MMI and MSK scales are basically identical (see Annex 2.1 of this Chapter)

- a few buildings of vulnerability class C sustain damage of grade 2;
- a few buildings of vulnerability class D sustain damage of grade 1.

Damage grades 1 and 2 are defined as follows in Figure 2.5:

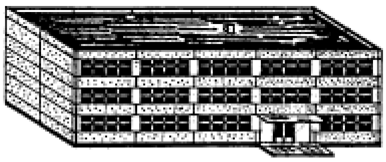
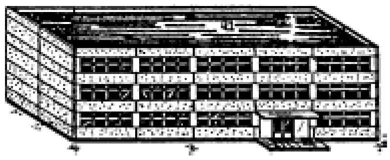
Classification of damage to buildings of reinforced concrete	
	**Grade 1: Negligible to slight damage** **(no structural damage,** **slight non-structural damage)** Fine cracks in plaster over frame members or in walls at the base. Fine cracks in partitions and infills.
	**Grade 2: Moderate damage** **(slight structural damage,** **moderate non-structural damage)** Cracks in columns and beams of frames and in structural walls. Cracks in partition and infill walls; fall of brittle cladding and plaster. Falling mortar from the joints of wall panels.

Figure 2.5 Damage grades 1 and 2

Damage grade 2 is limited to slight structural damage. There are no KCB buildings which are expected to lose their functions following an earthquake within the design basis.

Similarly the mechanical, electrical and instrumentation and control (I&C) systems are not expected to lose their function following a design-basis earthquake. Industrial earthquake experience as well as extensive seismic qualification programmes conducted in the US and Europe demonstrate high seismic capacities even if there is no explicit design against earthquake at all. Examples include:

- The Albstadt (Baden-Württemberg, Germany) earthquake in 1978 had an intensity of between VII1/2 and VIII (i.e. at least one intensity level more than the KCB design basis) and represents one of the three most severe earthquakes ever recorded in Germany . Structural damage at conventional steel and reinforced concrete structures did not occur (with one exception). Damage at mechanical, electrical and I&C systems in the conventional industry can be summarised as follows:

    - damage to electrical equipment occurred where either unrestricted displacements were possible (not the case with KCB electrical equipment which is anchored properly) or where equipment was damaged by debris from unqualified structures such as masonry chimneys (not relevant for KCB). Furthermore, Buchholtz relays of ca. 40 transformers (not resistant against induced vibrations) were actuated. While this may also be a relevant failure mode for the transformers connecting KCB to the grid, sufficient defence against LOOP is provided by the different design and the redundancy of auxiliary and stand-by transformers. Electrical cabinets, turbines and generators were not impacted at all;
    - damage to mechanical equipment (tanks, pumps, fans, piping, etc.) was generally very low. The limited number of findings was related to anchorage problems of tanks inducing neither leakages nor other functional failures. Remarkably, mechanical systems showed no functional failure even if masonry structures housing the systems were subject to significant structural damage;

- Similarly the seismic qualification utility group (SQUG) examined the consequences of the San Fernando (California) earthquake in 1971 with respect to damages in the conventional industry. Damages to 16 fossil power plants which had experienced peak ground acceleration of between 0.35 g and 0.5 g were examined. None of these plants was limited in its power production capability. 2,600 components whose operability was required after the earthquake were inspected.
- None of these items was damaged by induced vibrations. Only one valve failed due to an interaction effect ;
- Similar results were obtained in a study of six earthquakes in the Pacific region which occurred between 1971 and 1985 and were characterised by magnitudes of between 6.2 and 8.1 on the Richter scale. It was shown that equipment not designed for seismic loads explicitly is likely to withstand peak ground accelerations of up to 0.5 g if some basic rules for equipment design are considered . This is generally the case for the KCB design, considering various operational transients;

- Finally even the Niigata Chüetsu Offshore Earthquake in 2007 and the Tohuko earthquake in 2011, in which the design-basis PGA of impacted nuclear power plants were exceeded by factors of up to 2.5, induced limited seismic damage to structures and components only;
- IAEA TECDOC 1333 summarises various studies of earthquake performance of piping systems as follows: 'Concerning piping systems, it clearly appears from the feedback experience that they survive earthquake shaking motion particularly well, even amplified by the bearing structures. It is worth mentioning that most of them were not designed against earthquakes.

In conclusion, a challenge to the safety systems following a design-basis earthquake of VI½ intensity is unlikely.

The second level of defence is provided by a deterministic protection concept relying on seismically qualified safeguards only, whereas all SSCs not designed for DBE loads are assumed to have failed conservatively. Seismic qualification has been established either by designing the SSCs for DBE loads explicitly or ensuring their successful performance in an exhaustive seismic design assessment and retrofit programme (calculations, tests, walkdowns, etc.) during the second 10 yearly safety evaluation.

The three fundamental safety functions for PWRs are:

- control of reactivity;
- removal of heat from the core and from spent fuel;
- confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limiting accidental radioactive releases.

A functional analysis considering both at-power and shutdown conditions has been carried out in the second 10 yearly safety evaluation in order to identify plant-specific safety functions required after earthquakes.

The following functions have been identified:

RCPB isolation and RCS inventory make-up

- Integrity of the RCPB;
- closure of the RCPB isolation valves;
- automatic injection of borated water to the RCS with the back-up coolant make-up system TW.

The integrity of the RCPB is ensured by verifying the integrity of the primary circuit, including all connected pipes up to the second isolation valves. Automatic closure of the RCPB isolation valves is triggered passively by check valves or by the reactor protection system (PS) at low PZR level and low RCS pressure or at high containment pressure and low RCS pressure. These criteria would also trigger the automatic start-up of the back-up coolant make-up system TW by the PS.

Shutdown of the reactor ensuring long term subcriticality

- Fast negative reactivity insertion due to SCRAM function;
- long-term subcriticality injecting borated water to the RCS with the back-up coolant make-up system TW.

To ensure reactor shutdown operability of the control rods for SCRAM after an earthquake has been demonstrated. SCRAM is initiated by the reactor protection system (PS), e.g. on low RCP speed. While the relevant PS channels are qualified for operability after an earthquake, their failure would also lead to SCRAM (fail-safe principle).

Decay heat removal

- Feedwater supply to the steam generators with the back-up feedwater system RS and Main and auxiliary feedwater system RL;
- steam release with the Main steam system;
- manual secondary cooldown to LHSI/RHR conditions;
- long-term decay heat removal from both reactor pressure vessel (RPV) and spent fuel pool (SFP) using the Back-up residual heat removal and cooling system (TE/VE, TG080/VE).

Feedwater supply to the SG with the RS systems is actuated upon low SG level by the PS. The PS will also control the SG level.

Steam release is performed with the main steam relief trains (MSRT), which will automatically reduce and maintain the secondary pressure at 75 bar. The MSRT will also support the manual secondary cooldown to RHR pressure in the longer term. The spring-loaded main steam safety valves would also ensure decay heat removal for situations where the MSRT are unavailable.

The main steam piping inside the reactor building is seismically qualified. Similarly, a non-isolable steam break downstream of the reactor building does not need to be postulated because the main steam lines are supported by the nuclear auxiliary building and the turbine building for which global stability after an earthquake has been shown. Furthermore, thermohydraulic safety analyses demonstrate that acceptance criteria will be met, even if a DN150 break is assumed in each main steam line and main steam isolation is assumed to be unavailable conservatively, as a result of the flow limiters in the pipelines.

For both power and shutdown states long term decay heat removal is ensured by the Back-up residual heat removal system TE and the Back-up cooling water system VE supplied from the wells. Operation of this system is only required in the longer term[14] to allow manual actions, such as proper aligning of valves, especially in the Safety injection and residual heat removal system TJ. Similarly decay heat removal from the SFP using the alternate spent fuel cooling system TG 080 is only required in the longer term and is therefore actuated manually.

Containment Isolation

Containment isolation is only required in case of additional failures induced by the earthquake, especially a break in piping connected to the RCS. The governing scenario would be a rupture of the Volume control system TA letdown line between the second RCPB isolation valve and HP cooler. The relevant lines penetrating the containment wall and potentially leading to significant radioactive releases are the supply and exhaust lines of the Nuclear ventilation system (TL004, TL010, TL075). The containment isolation valves in these lines are proven for

[14] The RS system provides feedwater capacity for 10 h autarky time plus 3 h for shutdown to RHR conditions and 72 h autonomy time.
Regarding fuel pool cooling, the grace time is higher than 6 h (start of boiling if the reactor core is just fully unloaded). Uncovery of the spent fuel pool does not occur in the first 100 h.

operability after an earthquake and would be automatically closed by the PS on low RCS pressure in combination with high containment pressure or with low PZR level.

Depending on their functional role with respect to these safety functions, the SSCs are assigned to one of the following safety classes:

- **Class 1:**
  All SSCs required to support the safety functions identified above are assigned to class 1. Additionally class 1 encompasses SSCs whose failure could lead to a release of radioactive substances and structures that are supposed to prevent an unacceptable release of radioactive substances to the environment;
  Class 2a:
  SSCs whose seismic failure may have an impact on seismic class 1 equipment;
- **Class 2:**
  All other SSCs are assigned to class 2.

The seismic classification gives further details regarding the functional requirements after an earthquake (stability, integrity and operability). Operability is typically required for those SSCs which only directly support the essential safety functions. Class 2a SSCs are typically qualified for stability (avoidance of interaction effects due to falling equipment) or integrity (avoidance of internal flooding).

In particular, the following requirements are established in seismic class 1:

- integrity of the RCPB up to the second isolation valves;
- operability of the RCPB isolation valves;
- operability of the Back-up coolant make-up system TW for boration and primary make-up;
- operability of the SCRAM system;
- operability of the primary-side Back-up residual heat removal systems TE;
- integrity of the primary RHR systems TJ for as far as it is required for the operability of TE;
- operability of the Back-up residual heat removal and cooling system TE/VE;
- operability of the alternate spent fuel cooling system TG080 and integrity of the Spent fuel cooling system TG for as far as is necessary for the operation of TG080;
- operability of the containment isolation valves in the lines TL04, TL10, TL75;
- integrity of the main steam lines inside the reactor building;
- operability of the feedwater isolation valves required for feedwater isolation;
- operability of the secondary-side Back-up feedwater systems RS;
- integrity of the feedwater piping and operability of relevant isolation valves for as far as is necessary for SG isolation from a broken main steam line and operation of the secondary-side Back-up feedwater system RS;
- operability of the diesel generators EY040 and EY050 to provide a power supply to the D2 400 V power distribution system;
- stability of buildings 01, 02, 33 and 35 housing the SSCs described above;
- operability of the Reactor protection system inside building 35;
- operability of the remote shutdown station installed inside building 35;
- operability of the HVAC systems inside buildings 33 and 35 to ensure ambient temperature conditions;
- integrity of the spent fuel pool racks;
- integrity of the containment lock gates.
- Integrity of outside doors and/or hatches to buildings 03, 33 and 35 up to a level of at least 7.3 m NAP (to address consequential flooding).

Class 2a encompasses firstly the buildings whose failure could impact the class 1 SSCs:

- nuclear auxiliary building 03;
- switchgear building 05;
- turbine building 04;
- ventilation stack 13.

Secondly, class 2a encompasses the SSCs installed in the buildings 01, 02, 33 and 35 whose seismic induced failure could directly or indirectly impact class 1 SSCs, especially:

- integrity of high-energy equipment such as the accumulators and the high energy heat exchangers of the volume control system (HP cooler, recuperative heat exchanger);
- stability of large components, such as the polar crane and the refuelling machine;
- stability of ventilation ducts and piping installed in the vicinity of safety equipment;
- integrity of piping whose failure could lead to internal flooding or unacceptable environmental conditions (temperature, humidity, pressure);
- tightening of loose components and structures.

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. An overview of the seismic design assessment carried out is given in Figure 2.6.

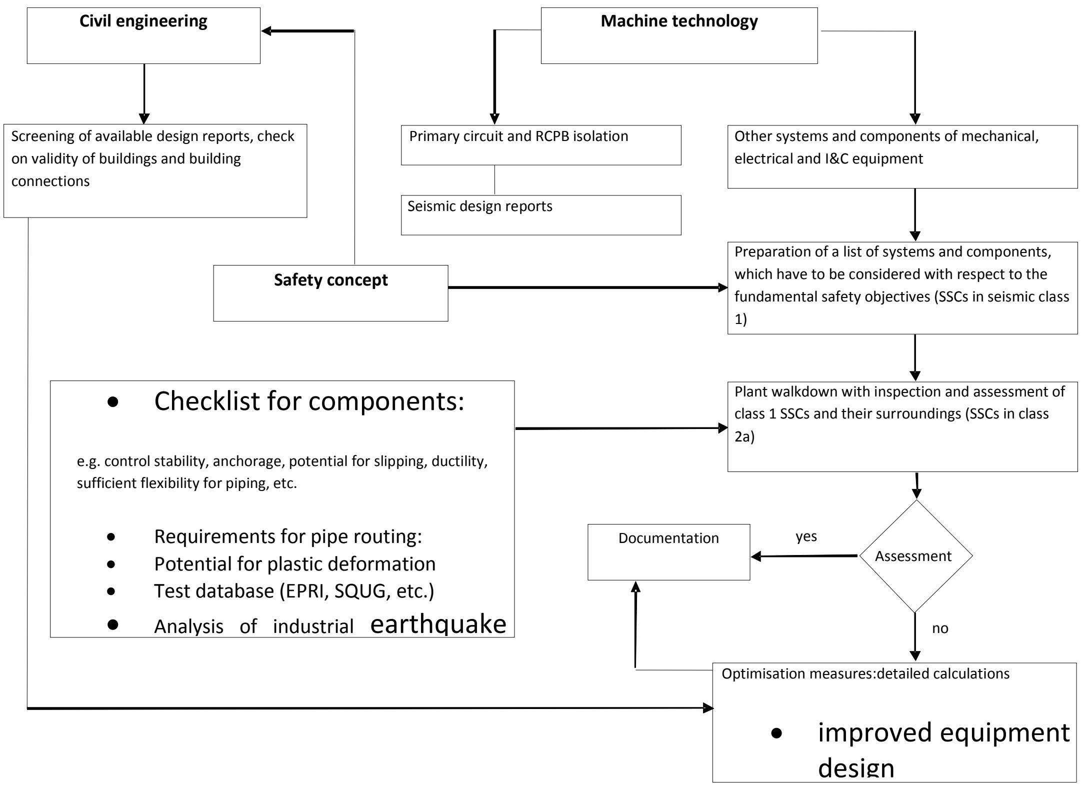


Figure 2.6 Overview of the seismic design assessment performed in the framework of the second 10 yearly safety evaluation

The third level of defence is established by the 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. These sources of seismic margins are detailed in section 2.2. It is shown that a loss of vital safety functions is practically excluded for earthquakes that exceed the design basis by one intensity degree.

The fourth level of defence is established by preventive emergency measures implemented to cope with design-exceeding situations induced by a loss of vital safety functions. The emergency measures for, in particular 'secondary bleed and feed' and 'primary bleed and feed' ensure decay heat removal in scenarios potentially induced by design-exceeding earthquakes such as total loss of AC power and loss of ultimate heat sink. The emergency procedures also include further measures to cope with a loss of primary RHR in shutdown states with reduced RCS inventory (mid-loop) and measures to maintain the SFP cooling.

The fifth level of defence is established by mitigative emergency measures foreseen to maintain the containment barrier after the onset of core melt. Based on an exhaustive set of severe accident management guidelines (SAMGs), this level encompasses, in particular, passive autocatalytic recombiners which eliminate the risk of a hydrogen explosion, and a containment filtered venting system, which ensures that the containment pressure can be reduced to and maintained at an acceptable level without uncontrolled release of radioactive substances. Similar to the measures in defence level four, these features are also available in conditions without any AC power supply and/or without any available heat sink.

The sixth level of defence is established by a sound emergency preparedness programme that ensures proper on- and offsite emergency responses to protect, as far as is practicable, the employees, public and environment against the effects of radioactive releases from the plant.

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

The following operating provisions are connected to the defence levels outlined in section 2.1.2.1:

With the **first level of defence** established by a robust design of operational equipment, no further special provisions are necessary. However, proper seismic housekeeping (including fastening of potentially loose parts in the buildings) backed-up by periodic shift walkdowns and general seismic load awareness will preclude a failure of this first line of defence.

The second level of defence is mainly ensured by the back-up feedwater system RS, the Back-up residual heat removal system TE, the Back-up coolant make-up system TW, the alternate spent fuel cooling system TG080, the Back-up cooling water system VE and their supporting systems. These engineered safeguards are actuated either automatically by the Reactor protection system (RS and TW) or manually if sufficient time is available (TE, TG080, VE). For the latter case, emergency procedures are available in both the main control room and the emergency control room. The proper use of these emergency procedures is covered during regular training with the shift team on the simulator.

The availability of the SSCs qualified for defence level 2, is ensured by a proper design in accordance with nuclear codes and standards (such as KTA), operating technical specifications and a regular test programme as well as sound maintenance and in-service inspection programmes (see 2.1.3).

The third level of defence is established by the significant margins in the design of SSCs and does not require special operating provisions. However, the maintenance, testing and in-service inspection programmes indicated above ensure that the margins considered are available on demand.

The fourth level of defence is covered by preventive emergency measures, especially 'primary feed and bleed' and 'secondary feed and bleed', for which dedicated emergency procedures are available in the MCR and the emergency control room. The correct identification of the need to carry out these measures, as well as their execution, is regularly covered in simulator training with the shift teams. The SSCs necessary to conduct these measures (mobile fire-fighting pump for SG feed after secondary bleed, bleed pilots for the PZR safety relief valves etc.) are regularly tested.

The fifth level of defence is covered by the severe accident management guidelines (see Chapter 6 on severe accident management), which aim at preventing a loss of the containment barrier after onset of core melt. The equipment necessary to conduct the relevant measures, e.g. the passive autocatalytic recombiners and the containment filtered venting system, are qualified for the harsh environmental conditions in a severe accident. Their availability is ensured by technical specifications and a regular test programme.

The sixth level of defence is established with an Emergency Preparedness Programme (see Chapter 6 on severe accident management).

2.1.2.3 Protection against indirect effects of an earthquake

2.1.2.3.1 Assessment of potential failures of heavy structures, pressure retaining devices, rotating equipment, or systems containing large amounts 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 flooding

The protection concept against earthquakes considers in particular the following indirect effects which may be induced by an earthquake.

Failure of non-classified SSCs

Equipment not directly needed to ensure the essential safety function but whose failure could impact the performance of class 1 SSCs is assigned to seismic class 2a and designed for stability and/or integrity as described above. The scope of class 2a SSCs has been exhaustively verified in the framework of the second 10 yearly safety evaluation.

Internal flooding

Seismic class 2a in particular includes piping systems with the potential to cause internal flooding. However, in the second 10 yearly safety evaluation it was shown that neither a failure of piping nor the proven proper seismis design would lead to unacceptable flood levels in the affected areas (e.g. regarding the conventional emergency cooling water system piping VF inside the reactor building annulus). Further defence is established by physically separating redundant safeguards wherever possible.

Internal fire

Protection against the risk of fire in safety-related buildings is generally ensured by the physical separation of safety redundancies, limited fire loads and proper seismic housekeeping. Furthermore, potential sources of consequential fire inside the safety-related buildings have been identified and excluded by the measures resulting from the second 10 yearly safety evaluation. One example of a failure path that was ruled out is the release of reactor coolant pump lube oil following DBE loads.

Note: In building 33 the fire-fighting system is also designed for operability. However, this is not the case for the systems in buildings 01, 02 and 35.

This is listed as a weakness and subsequently as a possible modification in section 2.2.4

Failure of high energy equipment

As described above, high-energy equipment (tanks, heat exchangers, piping, etc.), installed in the buildings 01, 02, 33 and 35, and whose failure may impact seismic class 1 SSCs was identified in 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.

LOCA

Consequential LOCA is excluded from the design basis since the RCPB is designed to withstand DBE loads. There is no cliff-edge effect regarding consequential LOCA anyway since the back-up coolant make-up system provides immediate make-up capability at high primary pressure. Additionally a fast secondary cooldown via the MSRT is actuated automatically by the PS decreasing the primary pressure and leak rate rapidly and allowing passive make-up with the accumulators of the safety injection and residual heat removal system . In the medium term, the back-up residual heat removal system TE can be actuated manually to allow make-up from the refueling water (TJ-) storage tanks and the containment sump.

Main steam line breaks

Consequential main steam line breaks downstream of the containment are generally precluded, because the nuclear auxiliary building and the turbine building have been designed for global stability. Fixed points have been installed inside the containment and on the roof of the nuclear auxiliary building to ensure reliable main steam isolation if a break was to occur. Furthermore, accident analyses, considering a DN150 break in each line and assuming a failure of main steam isolation, was performed, showing that acceptance criteria are met because of the flow restrictors installed in the pipelines..

Loss of conventional emergency cooling water and loss of ultimate heat sink

A loss of the Conventional emergency cooling water VF is conservatively assumed in the deterministic earthquake protection concept. This loss is covered by a seismically qualified and diversified Back-up residual heat removal and cooling system TE/VE, supplying water from groundwater wells.

A total loss of ultimate heat sink, i.e. a simultaneous loss of VF and VE, is excluded from the design-basis earthquake, but 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 (see Footnote 14 on page 15).

Possible on-site water sources include the demineralised water tanks UA and the Low-pressure fire-fighting system and tanks UJ. Feedwater can also be provided from a nearby fire-fighting pond via hoses and mobile pumps. As a last resort, salt water from the River Westerschelde could be utilized.

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

Decay heat removal from the SFP is also ensured when the water inventory in the pool starts to convert into steam. Manual water make-up needs to be provided to avoid uncovery of the spent fuel in this case. Sufficient grace time for this make-up is available since the spent fuel is not uncovered before 100 hours have elapsed. A corresponding emergency procedure is presently under elaboration.

ATWS

Seismic-induced ATWS caused by a mechanic blockage of the rods is excluded from the design basis due to an adequate seismic design of the rods and the RPV internals demonstrated in generic shake-table tests performed by tests by Siemens/KWU in the 1980s. There is no cliff-edge effect regarding seismic-induced ATWS because the Back-up coolant make-up system TW would be available for RCS boration.

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

Loss of offsite power can normally be mitigated by the diesel generators installed in the diesel generator buildings 10 and 72, but in the deterministic earthquake protection concept the loss of the EDGs (i.e. a station blackout) is considered conservatively. Plant safety is ensured by the Back-up residual heat removal system RS/TE backed-up by diesel generators installed in the Back-up systems bunker 33. Building 33 provides immediate fuel oil storage capacity for 24 hours if both of the diesel generators are running. An additional two days are provided by a tank on the roof of building 33. The diesel fuel for the main diesel generators EY010 to EY030 and EY080 may also be utilised. Afterwards, either another electrical power source needs to be provided (e.g. restoration of main or stand-by grid connection, 10 kV grid supply via buried cables, mobile diesels) or diesel fuel oil and lubricants need to be brought to the site, for example via the Westerschelde or via helicopters.

A total loss of AC power is excluded from the design-basis earthquake but could also be mitigated by the preventive emergency measures of ‘secondary bleed and feed’ and ‘primary bleed and feed’.

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

As explained in section 2.1.1, 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).

The Reactor protection system ensures the automatic transfer of the plant to a controlled state without any manual action. This state can be maintained using the secondary-side Back-up feedwater system for at least 10 hours afterwards. A secondary-side cooldown is actuated manually in the longer term in order to reach primary RHR conditions. After that, the Back-up residual heat removal and Cooling system TE/VE and TG080/VE is manually actuated.

Qualified shift personnel for these actions are likely to be immediately available since the switchgear building housing the main control room is designed for seismic loads (global stability). Due to the autarky time of 10 hours, there is also sufficient time to staff the emergency control room in building 35 with other qualified personnel (e.g. the maintenance shift team working in other buildings, plant engineers, operators living near the site, crisis team).

Once primary RHR with the back-up residual heat removal and cooling system is initiated, a durable heat removal from both RPV and SFP is ensured without any need for water make-up. The autonomy of the diesel generators has been described in section 2.1.2.3.2.

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

Fire

Consequential fire has been addressed in section 2.1.2.3.1.

Liquefaction

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 < 1m / s^2. In case there is a longer duration of strong motion (> 9 s), however, liquefaction of certain ground layers cannot be ruled out . In the case of large-scale full liquefaction, though, there will still be sufficient geotechnical safety margin against large-scale post-liquefaction slope failure, which could otherwise ultimately lead to failure of or damage to the plant's foundations. Also there will be sufficient safety margin against a loss of pile-bearing capacity and/or local foundation slip failure in case of liquefaction.

2.1.3 Compliance of the plant with its current licensing basis

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

The plant’s compliance with its current licensing basis is laid down in the Safety Report (VR) and the Technical Specifications (TS).

The availability of safety-related SSCs is ensured by adhering to strategic maintenance and surveillance plans and (in more detail) to extensive maintenance and in-service inspection programmes.

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

A check, performed immediately after receipt of the Significant Operating Experience Report (SOER), issued by the WANO following the Fukushima NPP accident, has shown that with respect to the availability and preparedness of auxiliary mobile equipment, e.g. mobile generators, fire-fighting equipment, hoses, etc., further improvement is possible. Corrective measures have been taken, or are planned to be executed .

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

A check has been performed immediately after receipt of the Significant Operating Experience Report (SOER), issued by the WANO following the Fukushima NPP accident (see 2.1.3.2).

Several deviations have been found and remedial actions have been defined. All findings have been reported to WANO.

Specific findings for earthquakes are:

- the availability of SSCs can be improved further by extending the availability requirements in the Operational Technical Specifications;
- the availability of mobile equipment can be improved further (see 2.1.3.2);
- cooling/filling/make-up water for the spent fuel pool in case of unavailability of in- and external AC-power supply (see also 2.2.4);
- qualification of fire-fighting equipment (see also 2.2.4).

2.2 Evaluation of the margins

2.2.1 Range of earthquakes leading to severe fuel damage

Neither a seismic PSA nor an explicit Seismic Margin Assessment has been performed in the past. The available seismic margins are elaborated below. The concept of seismic margins is introduced first, followed by how sources of seismic margins applicable to KCB are derived. These are concluded by an estimation of the KCB seismic margins regarding the fundamental safety functions.

Concept of seismic margins

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 found the widest application and is used hereafter.

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 .

Example: Subcriticality can be ensured by the SCRAM function or by boration of the RCS with the Back-up coolant make-up system TW. The seismic capacities of these diverse measures are provided by the minimum capacities of the SSCs involved. The seismic capacity of the safety goal of subcriticality is provided by the maximum of the capacities determined for SCRAM and TW.

Sources of seismic margins

- Choice of ground motion characteristics:
  Ground floor response spectra used in the verification study of the seismic adequacy are design spectra, i.e. the spectra have been subject to smoothing and broadening. In particular, this introduces a substantial artificial increase of energy content of the excitation.
- Conservatisms in determining the seismic demand :
  Earthquakes induce oscillations to a chain of different oscillators at a site: starting from the soil, the base plate and the upper floors of buildings up to smaller pieces of equipment installed in instrument racks or connected to flexible piping (Figure2.7). These different oscillating sub-systems show more or less significant interaction effects. For example, the heavy weight of the reactor building has a damping effect on ground oscillations. Similarly heavy equipment installed inside the containment dampens the vibrations of the supporting floors. A realistic description of this complex oscillation behaviour would therefore require modelling the entire system and considering non-linear effects as well, such as plastic deformation.

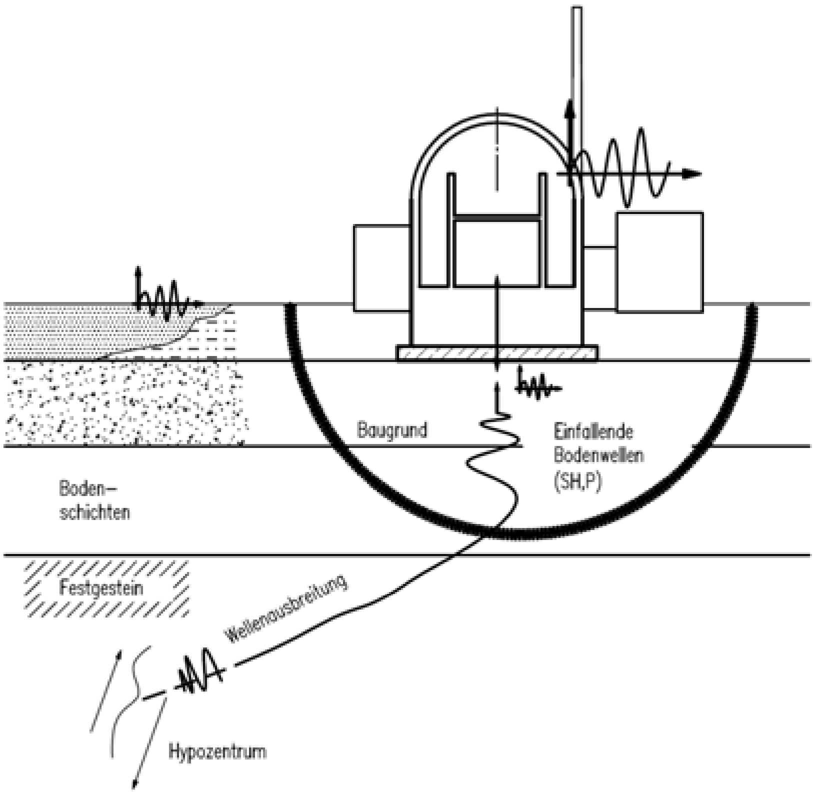


Figure2.7 Earthquakes and their impact on civil structures

Since such a global analysis of the entire path of the vibrational energy from the surrounding soil to the component is not practicable, the overall system is split into several sub-systems which have been analysed separately. The following example illustrates this approach.

SUB-SYSTEM	EXCITATION	RESPONSE
Ground + buildings	Ground response spectrum and corresponding time histories	Floor response spectra (secondary spectra) and corresponding time histories
Primary circuit, civil sub-structures	Floor response spectra (secondary spectra) and corresponding time histories	Tertiary spectra and corresponding time histories
Equipment (e.g. pipes)	Tertiary spectra and corresponding time histories	Quaternary spectra and corresponding time histories
Built-in components (e.g. valves)	Quaternary spectra and corresponding time histories	Loads on components

This approach induces several sources of conservatism:

- excitation characteristics and physical parameters considered for the different models are chosen conservatively to ensure that the conclusions made are valid without a high degree of uncertainty. The resulting level of conservatism may be moderate considering only one model. However, by combining several models in a calculation chain these conservatisms are multiplied and result typically in high factors of safety;
- bounding assumptions are typically made, especially with respect to input characteristics. For example, many relevant codes such as ASME and KTA require smoothing, broadening or even increasing of input spectra. Similarly the combination of load cases which can physically not occur simultaneously is often applied. Also the orthogonal vibration components are typically combined conservatively;
- interaction effects and, in particular, damping effects between the sub-systems are neglected. The resulting conservatism is especially important if the sub-structure is excited with a frequency near its resonance frequency, i.e. if the resulting vibrations are the most vigorous;
- realistic anchorage characteristics that allow small displacements and significantly influence the vibration's behavior (shifting of natural frequencies and thus avoiding resonant situations) of the anchored system are neglected;
- other nonlinear effects, such as plastic deformation and friction, are typically also not modelled and the corresponding energy dissipation is neglected or only considered with simplified approaches.

- Conservatisms in determining the resistance to seismic loads

These conservatisms result especially from:

- application of safety factors following the relevant codes;
- use of conservative material properties (typically 95-percentiles are used). In the case of concrete, the time-dependent material strengthening is neglected;
- neglecting plastic deformation capability;
- lasty, seismic margins result of course from stress utilisations < 1, i.e. if the resulting stresses are smaller than the code allowable stresses. In particular, the design of components is frequently governed by other load cases (large break LOCA, etc.).

- Conservatisms in the qualification of SSCs by shake-table tests

Operability of electrical and I&C equipment installed in building 33 and 35 is generally not demonstrated by calculation but by shake-table tests. This procedure provides the following margins :

- margins between required response spectra (RRS) and actual test response spectra (TRS). In practice it is not possible to perform a test where TRS and calculated RRS fit exactly. To avoid acceptance problems of the TRS they are generally chosen in a way that they cover the RRS conservatively;
- margins between test response spectra and damaging spectra. The shake-table test demonstrates that the tested component remains functionable under the test conditions. Damage will only occur at higher seismic demands. The USA's standards recommend assigning a safety factor of between 1.4 (operability during) and 1.95 (operability after) to account for this effect;
- limited energy content of floor response spectra with local maxima. The floor response spectra, which often show local maxima of the energy content of the calculated RRS, are significantly overestimated;
- frequency shifts between demand maxima and capacity minima. Both RRS and TRS are bounding spectra covering the seismic demand conservatively. Exceeding the actual capacity requires a local demand maximum (peak) to overlap a local capacity minimum. In practice these extrema are usually shifted against each other. Following the USA's practice, a safety factor of 1.1 is assigned to this;
- margins due to test duration. The typical test durations significantly exceed the KCB strong motion duration. For example, a test of between 5 and 35 Hz with 1 octave/minute results in > 3,000 load cycles. The strong motion duration in KCB is < 5 s thus limiting the number of load cycles in this frequency range to 5 x 35 =175.

Quantification of seismic capacities

Seismic capacities are typically quantified using the Separation of Variables Method . The median seismic capacity is expressed as the product of design-basis ground motion (PGA of 0.075 g for KCB) and an overall factor of safety F.

For structures, the factor of safety can be modelled as the product of three random variables:

$$F = F_S . F_\mu . F_{RS}$$

The strength factor, F_S, represents the ratio of ultimate strength (or strength at loss-of-function) to the stress calculated for A_{DBE}. In calculating the value of F_S, the non-seismic portion of the total load acting on the structure is subtracted from the strength as follows:

$$F_S = \frac{S - P_N}{P_T - P_N}$$

where S is the strength of the structural element for the specific failure mode, P_N is the normal operating load (i.e. dead load, operating temperature load, etc.) and P_T is the total load on the structure (i.e. the sum of the seismic load for A_{SSE} and the normal operating load).

The inelastic energy absorption factor (ductility factor), F_μ, accounts for the fact that an earthquake represents a limited energy source and many structures or items of equipment are capable of absorbing substantial amounts of energy beyond yield without loss-of-function. A suggested method to determine the deamplification effect resulting from inelastic energy dissipation involves the use of ductility modified response spectra . The deamplification factor is primarily a function of the ductility ratio μ defined as the ratio of maximum displacement to displacement at yield. More recent analyses have shown the deamplification factor to also be a function of system damping. One might estimate a median value of μ for low-rise concrete shear walls of 4.0. The corresponding median F_μ, value would be 2.45 at 7% damping.

The structure response factor, F_{RS}, is based on recognition that in the design analyses, structural response was computed using specific (often conservative) deterministic response parameters for the structure. As many of these parameters are random (often with wide variability), the actual response may differ substantially from the calculated response for a given peak ground acceleration.

F_{RS} is modelled as a product of factors influencing the response variability:

$$F_{RS} = F_{SA} . F_{GMI} . F_\delta . F_M . F_{MC} . F_{EC} . F_{SSI}$$

where

F_{SA} = spectral shape factor, which represents variability in ground motion and associated ground response spectra;

F_{GMI} = ground motion incoherence factor, which accounts for the fact that a travelling seismic wave does not excite a large foundation uniformly;

F_{δ} = damping factor, which represents variability in response due to the difference between actual damping and design damping;

F_M = modelling factor, which accounts for uncertainty in response due to modelling assumptions;

F_{MC} = mode combination factor, which accounts for variability in response due to the method used in combining dynamic modes of response;

F_{EC} = earthquake component combination factor, which accounts for variability in response due to the method used in combining earthquake components;

F_{SSI} = factor to account for the effect of soil-structure interaction, including the reduction of input motion with depth below the surface.

Similarly for equipment and other components, the overall factor of safety is composed of a capacity factor, F_C; a structure response factor, F_{RS}; and an equipment response (relative to the structure) factor, F_{RE}. Thus,

$$F = F_C \,.\, F_{RE} \,.\, F_{RS}$$

Past seismic PSA experience applicable to the KCB plant

Quantitative seismic capacity information applicable to KCB includes:

- an explicit assessment of the seismic capacities of first-generation Siemens/KWU PWRs, which has been carried out in the framework of the seismic PSAs for the 3-Loop plants: Neckarwestheim 1 (Germany) and Gösgen (Switzerland);
- generic fragility information about capacities of SSCs, which was provided by a survey of fragilities used in past seismic PRAs ;
- EPRI NP-6041 , which provides caveats for equipment to meet in order to assign a generic seismic capacity. These caveats have been developed using earthquake experience databases, qualification test databases, results of past seismic PSA and margin studies using the Separation of Variables Method mentioned above, as well as expert judgment. Tables 2-3 and 2-4 in EPRI NP-6041 provide generic seismic capacities of structures and equipment in terms of 5%-damped peak spectral acceleration measured at ground level. The generic capacity is then the High Confidence of Low Probability of Failure (HCLPF) capacity in terms of peak spectral acceleration at ground level. These tables are reproduced herein as Table 2.3 and Table 2.4. There are three ranges of acceleration (< 0.8 g, 0.8 -1.2 g and > 1.2 g) called 'bins'. For each structure or equipment type, the screening criteria for the three bins are provided. These criteria establish the type and level of evaluation that needs to be conducted before the component (i.e. structure or equipment) is deemed to meet the generic seismic capacity indicated by the bin. For example, the HCLPF capacity of seismic category I concrete frame structures could be generically assigned as 0.8 g (peak spectral acceleration) if the caveat 'e' is satisfied, i.e. the design was for a DBE of 0.1 g peak ground acceleration or greater.
  Considering the 5%-damped median NUREG/CR-0098 ground response spectral shape, this corresponds with a HCLPF of 0.3 g peak ground acceleration;
- similarly, EPRI NP-5223-SLR1 provides 'ruggedness spectra', i.e. response spectra at which qualified equipment is not assumed to fail provided that certain rules regarding the seismic design have been considered;
- generic information can be obtained from various publications (transactions of relevant conferences like SMiRT, Nuclear Engineering and Design etc.) as well as from different guidelines and positions stipulated by the USA's NRC.

	5 Percent-damped	Peak spectral	Acceleration
Type of Structure	<0.8 g	0.8 - 1.2 g	>1.2 g
Concrete containment (post-tensioned and reinforced)	no	(a)*	(b)
Freestanding steel containment	(c) (d)	(c) (d)	yes
Containment internal structures	(e)	(f)	yes
Shear walls, footings and containment shield walls	(e)	(f)	yes
Diaphragms	(e)	(g)	yes
Category I concrete frame structures	(e)	(f)	yes
Category I steel frame structures	(e)	(h)	yes
Masonry walls	yes	yes	yes
Control room ceilings	(i)	(i)	yes
Impact between structures	no	(j)	yes
Category II structures with safety-related equipment or with potential to fail Category I structures	(k)	yes	yes
Dams, levees, dikes	yes	yes	yes
Soil failure modes, soil-liquefaction and slope instability	(l)	(l)	(l)

NOTES:

(a) Major penetrations should be evaluated.

(b) Major and minor penetrations should be evaluated. The concrete containment structure only needs to be evaluated for a 5-percent damped peak spectral acceleration exceeding 2.0 g.

(c) No evaluation required if base mat is integral part of pressure boundary or steel pressure boundary is keyed to base mat to prevent slipping.

(d) Mark I tori require evaluation for earthquakes exceeding the design basis.

(e) Evaluation not required for Category I structures if design was for a SSE of 0.1 g or greater.

(f) Evaluation not required for Category I structures if design was by dynamic analysis for a SSE of 0.1 g or greater, and if the structure complies with ACI 318-71 or ACI 349-76 or later editions ductility detailing requirements.

(g) Evaluation not required for Category I structures if design was by dynamic analysis far a SSE of 0.1 g or greater, and if the diaphragm complies with ACI 381-71 or ACI 349-76 or later editions ductility detailing requirements, provided the diaphragm seismic loads were explicitly calculated.

(h) Evaluation not required if structures were designed using dynamic analysis and meet the requirements of AISC, 7th Edition, 1970 or later.

(i) Inspect for adequacy of bracing or safety wiring.

(j)	Investigation can be limited to potential for electrical malfunction (relay or contactor chatter) and loss of equipment anchorage in immediate vicinity of impact.
(k)	Evaluation not required provided the structure is capable of meeting the 1985 UBC Zone 4 requirements.
(l)	Refer to Appendix C and Section 7 for screening criteria.

Table 2.3 Civil structures screening criteria from EPRI NP-6041

	5 Percent-Damped	Peak Spectral	Acceleration
Equipment Type	<0.8 g	0.8 - 1.2 g	>1.2 g
NSSS Primary Coolant System (piping and vessels)	no (a)	no (a)	yes
NSSS Supports	(b)	(b) (c)	yes
Reactor internals	(z)	yes	yes
Control rod drive housings and mechanisms	(d)	yes	yes
Category I piping	(e)	(e)	yes
Active valves	no[2]	(f)	yes
Passive valves	No	no	(g)
Heat exchangers	(h)	(i)	yes
Atmospheric storage tanks	Yes	yes	yes
Pressure vessels	(h)	(i)	yes
Buried tanks	(j)	(j)	yes
Batteries and racks	(k)	(k)	yes
Diesel generators (includes engine and skid-mounted equipment	(l)	(l)	yes
Horizontal pumps	no	no	yes
Vertical pumps	no	(m)	yes
Fans	(n)	(o)	yes
Air handlers	(n)	(o)	yes
Chillers	(n)	(o)	yes
Air compressors	(n)	(o)	yes
HVAC ducting and dampers	(e)	(e) (p)	yes
Cable trays	no	(q)	yes
Electrical conduit	no	(r)	yes
Active electrical power distribution panels, cabinets, switchgear, motor control centers	(s) (t)	(s) (t)	yes
Passive electrical power distribution panels, cabinets	(s)	(s)	yes
Transformers	(u) (v)	(u) (v)	yes
Battery chargers	(w)	(w)	yes
Inverters	(w)	(w)	yes
Instrumentation and control panels and racks	(s) (t)	(s) (t)	yes
Temperature sensors	no	(x)	yes
Pressure and level sensors	no[3]	(x)	yes
[1] In addition to the screening criteria anchorage for equipment and subsystem needs to be evaluated.			
[2] The SRT should be cognizant of potential situations where extremely large extended operators are attached to 2-inch or smaller piping.			
[3] Note that pressure and level sensor will not fail at spectral accelerations below 0.8 g; however, systems engineers should be aware that these sensors may record a			

change in state due to the earthquake motion.
NOTES:
(a) BWR piping with suspected intergranular stress corrosion cracking may require evaluation.
(b) Evaluation not required if supports are designed for combined loading determined by dynamic SSE and pipe break analysis.
(c) Regardless of footnote (b), evaluation is recommended for PWR pressurizer supports and BWR reactor vessel and recirculation pump supports.
(d) Evaluation not required if CRD housing has lateral seismic support.
(e) Walkdown of representative piping and ducting systems should be conducted following Section 5 guidance.
(f) Evaluation recommended for MOVs in piping lines of 2 inches diameter or less.
(g) Walkdown to assure that valves do not impact adjacent structures or equipment.
(h) Margin evaluation on1y needs to consider anchorage and supports.
(i) For vessels designed by dynamic analysis or equivalent static analysis enveloping vessel inertial and piping loading, only the anchorage and supports require evaluation. For vessels not meeting these criteria, all potential failure modes require evaluation.
(j) Evaluation of piping connections is required. Other failure modes do not require evaluation.
(k) Batteries mounted in braced racks designed for seismic loads or qualified by dynamic testing do not require evaluation. Rigid spacers between batteries and end restraints are required. Batteries should be tightly supported by side rails.
(l) Margin review should be conducted for anchorage and attachment of peripheral equipment. Can be done by visual inspection for a peak spectral acceleration of 0.8 g or less.
(m) Margin evaluation required for vertical pumps with unsupported lengths of casing below the flange exceeding 20 feet or pumps with shafts unsupported at their lower end.
(n) All units supported on vibration isolators require evaluation of anchorage.
(o) Evaluation should focus on anchorage and supports.
(p) Evaluation required only for potentially large relative displacements between structures or equipment and structures.
(q) See Appendix A, “Cable Trays and Cabling” for guidance.
(r) No evaluation required if supports generally meet the National Electrical Code.
(s) Walkdown should be conducted to verify that the instruments are properly attached to the cabinets.
(t) Relays, contactors, switches, and breakers must be evaluated for chatter and trip if functionality during strong shaking is required.
(u) Anchorage evaluation required.
(v) Liquid-filled transformers require evaluation of overpressure safety switches. The transformer coils should be restrained within the cabinet for dry transformers.
(w) Solid state units require anchorage checks. Others require evaluation.
(x) Insufficient data are available for screening guidelines. Emphasis should be on attachments.
(y) Units mounted on structures at elevations exceeding 40 feet above grade should be reviewed if realistic (median centered) SME 5% damped horizontal floor spectra exceed 2g.
(z) Insufficient data to enable recommendations to be made.

Table 2.4 Screening criteria for equipment and sub-structures from EPRI NP-6041

Estimation of the KCB seismic capacity with respect to fundamental safety functions

i) Capacity of buildings supporting the fundamental safety functions

As indicated under a) both operational and safety-related KCB buildings are not supposed to fail after an earthquake. However, for the assessment of seismic margin, only buildings 01, 02, 33 and 35 are specified here.

- Reactor building with annulus (buildings 01, 02)

A seismic re-evaluation was performed during the second 10 yearly safety evaluation for the reactor building, which was initially not designed for DBE loads. Results indicated that the allowable stresses are generally not exceeded . Taking into account various conservative assumptions, the seismic performance was found to be acceptable.

Seismic PSA experience in the USA with regard to reactor buildings indicates median capacities of between 2.5 g and 9 g peak ground acceleration . However, while there are various sources of seismic margin available, as indicated above, such a high capacity cannot be assigned to KCB without detailed analysis.

The HCLPF capacities of the reactor buildings in Neckarwestheim 1 and Goesgen exceeded the design-basis PGA by a factor of 2 or more 33]. The primary failure mode was shear failure of the missile shield. Sliding of the steel containment against the concrete structures as well as a shear failure of the outer shield have been found to have high margins.

The overall median factor of safety

$$F = F_S \,.\, F_\mu \,.\, F_{RS}$$

is typically expected to be in the range of 4 to 12.

In the absence of a detailed fragility analysis, a HCLPF estimate for the KCB reactor building may be derived as follows:

The strength factor F_S is expected to be larger than 1 since the allowable stresses in the piles are not exceeded (load factor of 0.81) . Taking into account the fact that the allowable stress is an 84% fractile, the corresponding median exceedance factor is 0.81/1.4 = 0.58, therefore a strength factor of 1/0.58 = 1.7 is assumed in the sequel.

Since no detailed fragility analysis has been made, a conservative value of 1.25 will be assumed for the inelastic energy absorption factor F_μ, in accordance with EPRI NP-6041.

The seismic response factor F_{RS} is significantly higher than 1 (conservative spectra, soil structure interaction, load combination method, method of combining orthogonal components, conservative damping, etc.).

The following may be assumed considering past PSA practice:

$$F_{RS} = F_{SA} \cdot F_{GMI} \cdot F_{\delta} \cdot F_{M} \cdot F_{MC} \cdot F_{EC} \cdot F_{SSI}$$

$$F_{SA} = 1.4$$

(factor 1.4 results from a recent EPRI study for 28 sites in the USA)

$$F_{GMI} = 1.05$$

(value often used in recent seismic PSA studies)

$$F_{\delta} \cdot F_{M} \cdot F_{MC} \cdot F_{EC} = 1.2$$

(engineering judgment in absence of detailed information about the methods used in)

$$F_{SSI} = 1.3$$

(conservative judgement taking into account various effects introduced above)

Thus F_{RS} could be estimated to be 1.4 x 1.05 x 1.2 x 1.3 = 2.3

The overall factor of safety F is therefore: 1.7 x 1.25 x 2.3 = 5.0

The median seismic capacity is therefore:

$$A_m = 5.0 \times 0.075g = 0.37 \text{ g}$$

Using a generic composite uncertainty β_C of 0.4 as recommended in IAEA TECDOC 1487 , the corresponding HCLPF is

$$\text{HCLPF} = A_m \times \exp(-2.33 \times 0.4) = 0.37 \text{ g} \times 0.39 = \mathbf{0.15\ g}$$

This HCLPF capacity estimate should be interpreted as the HCLPF of the reactor building with respect to the support of engineering safeguard systems, such as the Backup coolant make-up system TW and the Backup feedwater and Backup residual heat removal systems RS and TE. Large displacements of the reactor building against other buildings and galleries which exceed the design basis may impact the integrity of pipes penetrating the reactor building.

- Backup systems bunker 33 and Remote shut-down building 35:

These buildings have been erected with consideration for seismic loads, as well as loads from other external hazards (an aeroplane crash, explosion pressure waves). There are no concerns regarding the seismic design. The EPRI NP-6041 screening criteria for buildings (Table 2.2) are applicable and would support a HCLPF of at least 0.3 g (lower screening level). Indeed the seismically qualified supporting buildings of Neckarwestheim 1 and Gösgen (emergency feedwater buildings, emergency diesel generator buildings, etc.) have been found to have HCLPF capacities in a similar order of magnitude.

Therefore the seismic capacities of buildings 33 and 35 do not dominate the seismic capacity of the fundamental safety functions.

ii) Fundamental safety function: subcriticality

Subcriticality is ensured by SCRAM or boration with the Backup coolant make-up system TW. Moreover RCS boration may also be possible with the Volume control system TA which is not credited conservatively.

Control rod drives and RPV internals of the Siemens/KWU design were subject to extensive real-scale shake-table tests in the 1980s. Proper SCRAM capability without significantly delayed rod-drop time has been demonstrated for response spectra enveloping existing and targeted sites for KWU PWRs, including in Japan and Iran. Application of a generic HCLPF capacity of 0.3 g proposed by the USA's NRC is therefore judged to be applicable.

The TW system, installed in 1986, has undergone a careful seismic design. All sources of seismic margins mentioned above are therefore generally applicable. Proper installation and an absence of critical interaction effects with non-qualified equipment have also been documented in the plant walkdown. Thus the EPRI NP-6041 screening criteria presented in Table 2.3 are applicable for the relevant SSCs and a HCLPF of 0.3 g or more could be assigned.

In conclusion, the safety function subcriticality is judged to be highly reliable after seismic events. An explicit SMA using the Separation of Variables Method and the MIN-MAX approach to account for both reactivity control systems would probably indicate a HCLPF of higher than 0.3 g. The HCLPF is therefore expected to be governed by building capacities.

iii) Fundamental safety function: decay heat removal (secondary side)

When an earthquake occurs during power operation, decay heat removal is initially ensured by the secondary-side heat removal. The main and auxiliary feedwater system RL, as well as the Backup feedwater system RS, are available for SG feed but only the RS system is explicitly qualified for the DBE as described in section0. Thus only the SG feed with the RS system is taken into account for a conservative capacity estimation.

The RS system was installed in the second 10 yearly safety evaluation following well-established seismic design criteria. An absence of interaction effects (e.g. inside the containment) was demonstrated in the plant walkdown. The two redundant subsystems inside building 33 are also physically separated. The EPRI NP-6041 criteria are generally applicable and a HCLPF of 0.3 g could be claimed. The seismic PSA for Neckarwestheim 1 identified a similar margin in a detailed analysis for the emergency feedwater pump system .

A further seismically diversified feedwater injection capability is provided by the emergency measure of secondary feed and bleed. However, while the main steam valves necessary to depressurise the SG are highly reliable (HCLPF capacities > 0.3 g have been found for both Neckarwestheim 1 and Gösgen), the seismic capacity of the injection from the main feedwater storage tank or with a mobile pump from building 33 cannot be predicated without a detailed assessment.

In conclusion, a seismic capacity of higher than 0.3 g can be claimed for the RS system supporting secondary-side heat removal. A detailed assessment, considering also secondary bleed and feed, would probably show a much higher margin. However, in that case, the overall capacity of seismic capacity would be expected to be governed by the capacity of the reactor building.

iv) Fundamental safety function: decay heat removal (primary side)

Primary-side heat removal is ensured by the Backup residual heat removal system TE and the Backup cooling water system VE. Both systems were introduced in the second 10 yearly safety evaluation and are designed for seismic loads. The normal nuclear cooling chain TJ/TF/VF is not supposed to fail after DBE loads corresponding with an MSK intensity of VI½ but it is not formally qualified and therefore not considered in the capacity estimation.

The careful seismic design of both TE and VE is backed-up by a seismic re-evaluation of TJ system parts necessary for TE operation. Interaction effects from non-qualified SSCs have been excluded in the plant walkdown. Eight wells placed at different site locations are available to ensure sufficient VE flow, even if the water level of some of these wells may be affected by earthquake effects on the ground. The Backup residual heat removal and Backup cooling water systems are therefore considered to be highly reliable after an earthquake. EPRI NP-6041 criteria are applicable to TE and support a high capacity. The Backup cooling water system is judged to be comparable with a similar system in Neckarwestheim 1, which is used for emergency diesel generator cooling. No seismic concerns were identified in Neckarwestheim 1 and an EPRI screening value of 0.3 g HCLPF was established. The capacity of the wells themselves cannot be estimated without detailed geotechnical investigations but is judged not to dominate the capacity considering past earthquake experience.

A further option to remove decay heat via the primary is provided by primary feed and bleed. However, durable heat removal would also rely on the availability of the back-up residual heat removal and cooling system in the longer term. Thus primary feed and bleed will not enhance the seismic capacity of primary heat removal directly. Its benefits are mainly the additional grace time provided in case there is a total loss of AC power (RCS make-up by accumulators) and the reliable depressurisation of the RCS, which ensures proper containment performance after the onset of core melt.

In conclusion, the primary heat removal systems of KCB would be highly reliable after an earthquake. Assigning a HCLPF capacity of 0.3 g would probably be justified, considering past seismic PSA experience. The seismic capacity of the groundwater wells supporting VE cannot be assessed without geotechnical considerations but this is not expected to be required.

v) Fundamental safety function: decay heat removal (spent fuel)

The spent fuel pool is an integral part of the concrete structure of the reactor building and is provided with an additional steel liner. In case of damage to the liner, the migration of the pool water can be stopped because the leak-off pipes are provided with shut-off valves. Spent fuel pools have therefore not been found to be of concern in past seismic PSA experience, including at Gösgen and Neckarwestheim 1.

Spent fuel cooling is initially ensured by the thermal inertia of the water inventory in the spent fuel pool. The alternate spent fuel cooling system TG080 was installed as a result of the second

10 yearly safety evaluation and provides a sound design as regards seismic loads. The EPRI NP-6041 criteria are judged to be applicable for the SSCs of the system and a generic HCLPF of at least 0.3 g can be assumed.

As described in section 2.1.1, there are emergency measures in addition to relying on the provision of cold water to the pool. A seismic capacity for these measures has not been estimated since the overall seismic capacity for the spent fuel cooling is probably governed by the containment building fragility.

vi) Fundamental safety function: confinement of radioactive substances

A capacity estimation for this function is provided in section 2.1.1.

vii) Support functions

Important support functions include, in particular, the automatic actuation of safeguards by the reactor protection system, the manual actuation of the decay heat removal system from the emergency control room in building 35, Emergency Grid 2 backed-up by dedicated batteries and the diesel generators.

All these support functions have been carefully designed for seismic loads.

EPRI NP-6041 criteria are generally applicable and past seismic PSA experience indicates that electrical and I&C cabinets provide high capacities as long as specific failure modes can be ruled out (relay chatter in switchgears and motor control centres, interaction effects between cabinets, masonry walls impacting electrical and I&C equipment, etc.), which is the case for KCB. The reactor protection system, the panels in the emergency control room, the motor control centres and switchgears have all undergone an intensive test programme on shake tables to introduce the significant margins described above.

Diesel generators, which are already subject to significant vibrations in normal operating conditions, are typically of no seismic concern. Following the EPRI screening criteria, a capacity of at least 0.3 g could be established for KCB. The SSCs supporting the diesel generators (fuel tanks and fuel pumps, batteries, starting air receivers) as well as the batteries are explicitly designed for DBE loads. Interaction effects have been ruled out. Thus the lower 0.3 g screening level from EPRI NP-6041 is applicable.

Cable trays and HVAC ducts have also undergone careful seismic design. With regard to cable trays, Siemens/KWU shake-table tests with seven representative trays were taken into account: no structural failure was detected despite a demanding test response spectra of 14 seconds of sinusoidal excitation with maximum horizontal spectral accelerations of 1 g (vertical 1.65 g) . Similarly the design of HVAC ducts was carried out with regard to test results regarding the strength of the flange connections .

Summary:

In Table 2.5 Overview of the estimated HCLPF seismic capacities for the different buildings are presented.

Building / Fundamental safety function	HCLPF capacity
Reactor building with annulus (01, 02)	0.15 g
Backup systems bunker (33)	0.3 g
Remote shut-down building (35)	0.3 g
Safety function: subcriticality	0.3 g
Safety function: decay heat removal (secondary side)	0.3 g
Safety function: decay heat removal (primary side)	0.3 g
Safety function: decay heat removal (spent fuel)	0.3 g
Safety function: confinement of radioactive substances	0.3 g
Support functions	0.3 g

Table 2.5 Overview of the estimated HCLPF seismic capacities for the different buildings

In a simplified assessment, it has been 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**.

It should be noted that the European Utility Requirements expect a HCLPF for new builds to be at least 40% higher than the design-basis ground motion . A HCLPF of 0.15 g, when compared to a design-basis PGA of 0.075 g, allows significantly more margin (100%) to be established for KCB.

2.2.2 Range of earthquake leading to loss of containment integrity

The capacity with regard to confinement failure is judged to be dominated by sliding of the steel containment against the concrete structures, which 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. A similar HCLPF capacity has also been established in a detailed analysis of Neckarwestheim 1; however, this plant is designed for 0.17 g peak ground acceleration and therefore not directly comparable.

In conclusion, the HCLPF capacity, i.e. the peak ground acceleration where the probability of confinement failure is low under high confidence, is expected to be in the range of **0.3 g**. The median capacity, i.e. the peak ground acceleration where the failure probability exceeds 50%, is in the range of **0.7 g**.

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) – see also real earthquake experiences from fossil power plants in the Pacific region which support this statement.

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

In general, an earthquake at a sea side location like KCB can give rise to external flooding in two ways:

- an earthquake at sea may cause a tsunami, which in turn may cause a flooding;
- an earthquake nearer to KCB may cause a dike failure, which may in turn cause flooding of the KCB site. The high water level required for this flooding may be caused by the earthquake itself or by high tide, possibly combined with storm surge.

In the first case, a hypothetical tsunami that is formed in the North Sea will not grow to unacceptable amplitudes, due to the relatively shallow water. A tsunami formed at greater depth at sea must travel a far greater distance through e.g. the North Sea or the Street of Dover to reach Borssele which will decrease its amplitude to a negligible magnitude. See Chapter 3 for more information on tsunamis.

For the second case, in general, the chance of damage to soil structures due to an earthquake is very small (< 10^{-4} per year) . Nevertheless, for the location of KCB various dike failure mechanisms have been investigated . For stability of the flood defense of KCB, so-called liquefaction of the soil is of particular interest. Liquefaction is the process in which loosely packed sand layers are affected by an earthquake and loose their foundation stability which may lead to large landslides or slope failures. The investigation has shown that the distance between the Westerschelde embankment and the KCB buildings is large enough to prevent liquefaction that could influence the stability of the KCB buildings or the flood defense.

As earthquakes and extreme high tides are two independent events, the contribution to the probability of flooding is negligible. Despite the improbability of a flood caused by or followed by an earthquake, adequate measures are in place to cope with this situation. In the event of a beyond design basis earthquake of 0.3 g and consequential external flooding of the KCB site, the systems that are required for safe shutdown will remain available. These systems are located in buildings 01, 02, 33 and 35, which are designed against both earthquake and external flooding.

To this effect, also building 03 is of interest because of the entrance from building 02 to 03. Like buildings 01, 02, 33 and 35, also building 03 has a water resistance beyond 7.3 m + NAP (the design basis flood level, see Chapter 3.) However, a critical parameter is the joint opening between the walls of the reactor dome (building 02) and the building 03. During an earthquake with a peak ground acceleration of 0.3 g the joint may open up to a maximum of 83 mm at a level of 26 m + NAP. At lower levels, the opening will be smaller. The water stop in this joint, fitted up to a height of 10 m + NAP, is able to take this deformation without very large strains. Therefore, an earthquake will not influence the water tightness of the joint.

In conclusion, a beyond design basis earthquake is not expected to lead to external flooding of the KCB site. In case external flooding does occur, this will not lead to unavailability of systems that are required for safe shutdown of the plant.

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

Potential cliff-edge effects result from:

Unavailability of shift personnel after 10 hours.

There is the potential for a cliff-edge with design-exceeding earthquakes if the MCR is destroyed and the site becomes inaccessible. This would lead to a HP 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 (see below).

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.

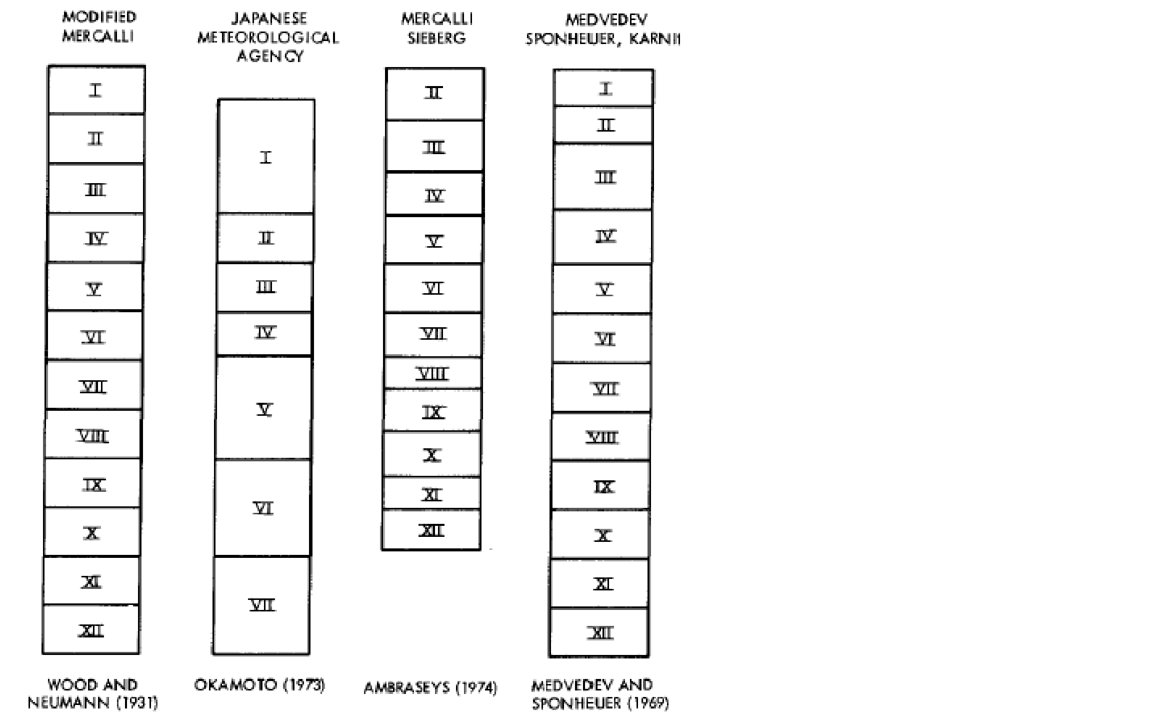
Possible inoperability of the fire-fighting systems in buildings 01, 02 and 35.

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

The following modifications/investigations could be envisaged:

- Emergency Response Centre facilities that could give shelter to the alarm response organisation after all forseeable hazards would enlarge the possibilities of the alarm reponse organisation;
- storage facilities for portable equipment, tools and materials needed by the alarm response organisation that are accessable after all forseeable hazards would enlarge the possibilities of the alarm reponse 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/suppresion;
- 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-Probalilistic 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 programm;
- develop check-lists for plant walk-downs and needed actions after various levels of the forseeable hazards.

Annex 2.1. Comparison of different intensity scales



Extracted from .R. Murphy and L.J. O'Brien, 'The correlation of Peak Ground Acceleration Amplitude with Seismic Intensity and other Physical Parameters', *Bulletin of the Seismological Society of America* 67 (1977) 3, pp. 877-915

Figure 2.8 Comparison of different intensity scales

In most cases, there should be no difficulty in converting between MSK values and EMS values on the system MSK = EMS. The most likely difference is that some uncertain values such as IV–V MSK or VI–VII MSK would now be assessed more accurately as IV EMS and VI EMS. Other differences may result from literal or restrictive interpretations of the MSK scale. For example, on a literal reading of the text of the MSK scale, the threshold of damage was intensityVI. Practical experience showed that the damage actually occured sometimes on occasions when all other data suggested lower intensities, and investigators who recognised this were allowing for the possibility of intensities being assessed as lower than VI MSK even when damage was reported. Other investigators who did not make this allowance may find that intensities assessed as VI MSK may in some cases become V EMS

Chapter 3 Flooding

3.1 Design basis

3.1.1 Flooding against which the plant is designed

The design level adopted for NPP Borssele is 7.3 m + NAP (in Dutch: ingestelde Nucleair OntwerpPeil or N.O.P.)

3.1.1.1 Characteristics of the design basis flood (DBF)

With regard to flood protection two distinct levels of elevation of structures, systems and components (SSCs) can be distinguished:

- 5 m + NAP ;
- 7.3 m + NAP.

Originally, NPP Borssele was designed to withstand a flood level of 5 m + NAP. Currently this level is denoted 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 m + NAP.

The controlled area buildings 01, 02 and 03[15] are leak tight. The (support) systems needed for safe shutdown and decay heat removal, located in the buildings 04, 05 and 10 are placed at an elevation higher than 5 m+ NAP. The cooling water inlet building (21) is water tight.

The generator transformer AT and its house load transformer BT (building 11[16]) are placed at elevations of respectively 3.4 m and 4.9 m+ NAP. Due to their dimensions, they can operate during a flood level of at least 5 m + NAP. The same applies to the auxiliary transformers BS001 (building 12) and BS002 (building 41) which are placed at an elevation of 5 m + NAP. The 150 kV switchyard outside the plant's perimeter, which connects the plant to the public grid, is also capable of withstanding flooding of at least 5 m + NAP. All buildings and transformers are placed on piles to ensure stability during flooding conditions.

A possible weak point, however, is the end pylon of the overhead line connecting the step-up transformer to the grid. This construction is not placed on a foundation, which could influence its stability under flooding conditions. However in case of flooding, the normal procedure will be to switch to house-load operation, a situation where the transmission line is not needed.

In addition to the original design, buildings 33, 34, 35 and 72 were erected for back-fitting measures after the completion and commissioning of the NPP. They are capable of

[15] See Annex 1.2 for building code descriptions.
[16] This is not a real building; it is the location of the transformer.

withstanding flooding of at least 5 m + NAP. All the above-mentioned buildings are depicted in Figure 3.1.

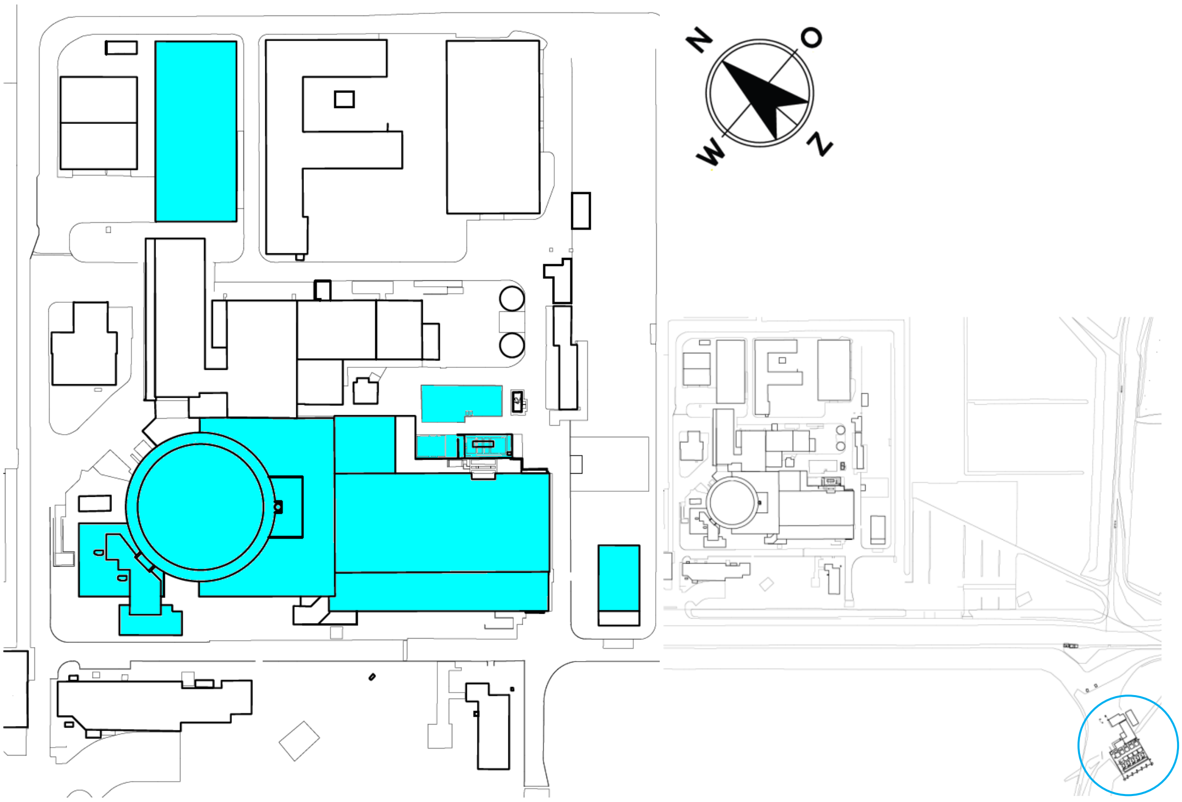


Figure 3.1 Buildings that are part of the ‘Laag Hoogwater Concept’ (low high-water concept) (left). The relative position of building 21 is indicated by a circle in the detail (right).

Currently, the flood protection level is 7.3 m + NAP due to back-fitting measures carried out in the following periods:

- 1983: radioactive waste storage building 34;
- 1980-1984: twofold redundant bunkered systems as a backup for the primary coolant make-up and secondary feed water systems, including independent basins for primary coolant and secondary feed water and dedicated diesel generators in building 33;
- 1991-1997: bunkered reactor protection system and emergency control room in building 35, twofold redundant diesel generators in building 72 and an additional deep well backup cooling water system (VE[17]).

The level of protection of 7.3 m + NAP applies to buildings 01, 02, 03, 33, 34, 35 and 72. See Figure 3.2.

[17] For an overview of system codes, see Annex 1.1

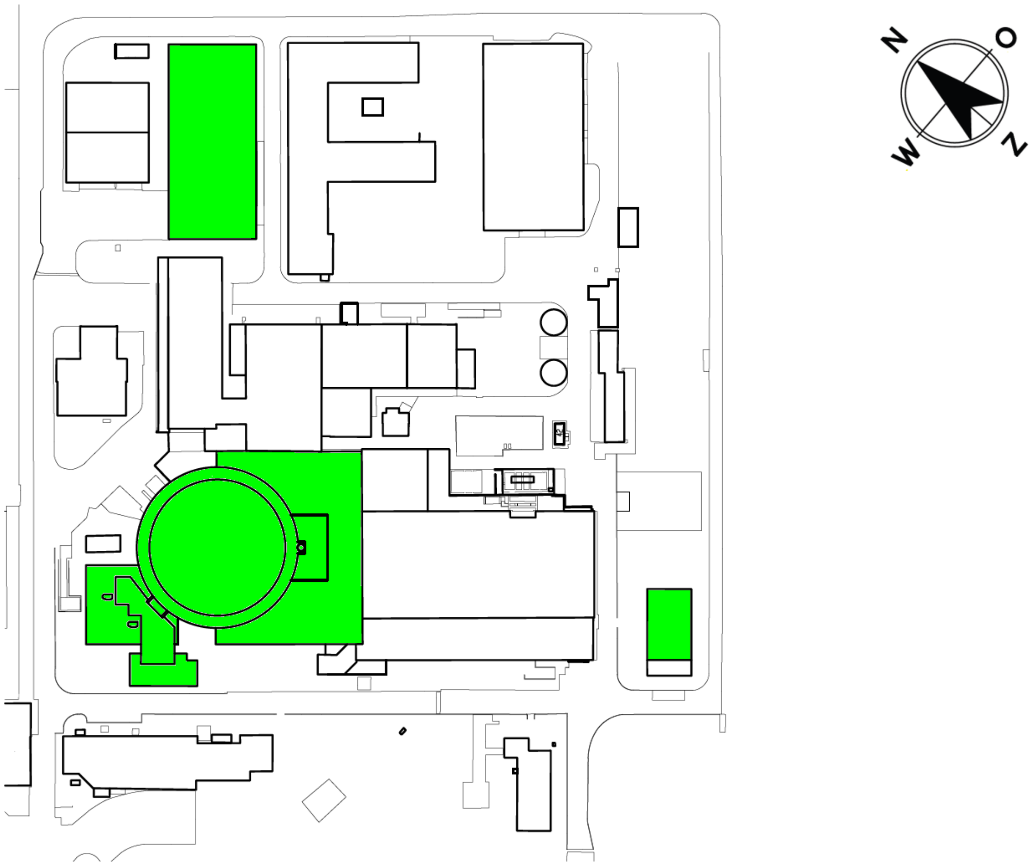


Figure 3.2 The highlighted buildings are capable of withstanding at least 7.3 m + NAP flooding.

3.1.1.2 Methodology used to evaluate the design basis flood

5 m + NAP

Background information of the original flood level of 5 m + NAP could not be traced. However, it can be assumed that it is based on the highest known water level at Borssele that was reached during the storm surge of 1 February 1953, which amounted to 4.7 m + NAP. See also Annex 3.1.

7.3 m + NAP

The design basis flood (DBF) adopted for NPP Borssele is 7.3 m + NAP. The background to the DBF is explained below.

N.B.P.

In 1980, the Nuclear Base Level (in Dutch: Nucleair BasisPeil, or N.B.P.) was introduced and defined for NPP Borssele. The N.B.P. results from the requirement that a nuclear power plant should be protected against external hazards in such a way that the probability of an accident with serious consequences caused by external events - in this case floods - will be small compared to the risk of serious accidents originating from causes within the plant itself. This requirement is met if the safety measures are such that an external event with a return period of 1 million year (frequency of 10^{-6} per year) can be withstood.

The N.B.P. is calculated for Borssele at 6.18 m + NAP. This value is extrapolated from the High Tide Total Exceedance Frequency chart (Annex 3.2). This chart is based on an analysis that made use of the astronomical and meteorological tide components that directly affect high water. The total high tide is the sum of these two components. The astronomical component is deterministic by nature and is calculated by the Rijkswaterstaat (RWS) for the location Borssele. The meteorological component is random by nature and is based on a long-term series of water-level measurements in the Vlissingen area of the Westerschelde.

Nuclear design level (N.O.P.) calculation

For flood-resistant design of a power plant, a more realistic 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.). Because of the wave effects of water, the calculated N.O.P. can be distinguished in the following:

- **Static N.O.P.** This is the level at which a constant water load acts on the walls of the buildings in which the safety-related systems and components are housed. This water level is used in the stress-strength calculations for the building design, to calculate the water pressure it needs to withstand.
- **Dynamic N.O.P.** This level is used to determine the minimum elevation at which systems have to be placed or at which buildings should be water-tight.

The N.O.P. of NPP Borssele is defined as the N.B.P., with the addition of the following:

- **Effects due to showers** According to the Rijkswaterstaat (RWS), a surcharge of 0.15 m is applicable for showers 5 to 10 km inland from the mouth of the Westerschelde.
- **Closing of the Oosterscheldedam** According to the RWS, a surcharge of 0.06 m is applicable when the Oosterschelde storm surge barrier is closed.
- **Compensation for rising sea level and decreasing soil levels** According to the RWS, there is an average sea-level rise of 0.30 m per century. For the next 20 years, this means a surcharge of 0.06 m. In view of the recent discussion on the impact of global warming on the mean sea-water level the Royal Dutch Meteorological Institute (KNMI) has been consulted. Their conclusion is that this figure is still acceptable.
- **Settlement surcharge** It can be seen from the building measurements that an average settlement of 0.035 m has taken place. Given the course of the settlement, see Annex 3.3, the increase in the settlement over the next 20 years will be small. For the settlement over 40 years, a one-time surcharge of 0.04 m is applicable.
  The settlement is evaluated every five years and the last review took place in 2007. It was then concluded that the settlement of the buildings after 60 years will be well within the margin of 40 mm.
- **Reduction due to hinterland** During overtopping of the dyke, it will take time before the water level on site reaches the same level as that on at the seaward side of the dyke as a vast area has to be filled. There could even be insufficient time for the whole polder to fill. However, as the size of this reduction factor of the water level is not known and because it is a reduction, it is not taken into account. This will result in a conservative approach to the N.O.P.
- **Wave height** On site, the dyke and buildings will limit the fetch, resulting in waves with a maximum height of 0.4 m according to the RWS. To account for reflection effects at the vertical walls of buildings, twice this wave height (0.8 m) is used in the calculation of the dynamic N.O.P. To determine the static N.O.P. a reduction with a half-wave height is used.

The contribution of the various factors is given in Annex 3.4.

The N.O.P. for NPP Borssele is then calculated as:

- 6.29 m + NAP (static);
- 7.29 m + NAP (dynamic).

For comparison, the highest flood level recorded near the location is 4.7 m + NAP, see Annex 3.1.

Regarding safety functions, the dynamic N.O.P. level is decisive. In practice, 7.3 m + NAP is adopted as the DBF and is used to demonstrate the flood resistance of structures.

N.O.P. and dykes

The N.B.P. does not take the effect of dykes into account. The exceedance frequency is based on the observed water levels in the Westerschelde. In the N.O.P. however the sea dyke is implicitly taken into account when determining the on-site wave height. The dyke will reduce the wave height on-site by reducing the fetch. Dykes for coastal defence in the province of Zeeland are designed to deal with flood levels with a return period of 4,000 years. In 2012 the sea dyke will be improved, so that the dyke will not fail in storm situations with a return period of 10,000 years, as will be discussed in section 3.1.2.2.

This return period complies with KTA 2207, but differs from the return period of the N.B.P. and N.O.P. If the dyke has failed and there is an open connection between the Westerschelde and water on site, the assumed reduction of the wave height is incorrect. In this scenario the foreland, the limited depth on site (ground level: 3 m + NAP), the remains of the dyke and the (ruins of collapsed) buildings in front of the plant will not necessarily limit the fetch but will reduce the wave height compared to the wave height in the Westerschelde. In case of a storm surge with a return period of 1,000,000 years, wave heights of 3.9 m can be expected on the Westerschelde. On site, wave height will be reduced.

3.1.1.3 Conclusion on the adequacy of protection against external flooding

Nuclear design level (N.O.P.) evaluation

Currently the N.O.P. is under review within the fourth 10 yearly safety evaluation. Due to the fact that the N.B.P. can be extrapolated with different fitting algorithms and can be based on different datasets/periods, it was decided to wait on the upcoming re-evaluation of the RWS.

The RWS will publish a new report in 2012 on the boundary conditions for the primary dykes protecting the Netherlands against flooding. This report will contain a new statistical analysis of the return periods of extreme high water levels based on a recent dataset.

For the Complementary Safety Margin Assessment charged by the European Council, EPZ has consulted the RWS and the KNMI in order to verify whether the current N.B.P. level is still valid. It can be concluded that it is still acceptable, but that a moderate change is possible.

Whereas no 'official', unambiguous or exact figure is available, it was decided to remain with the present value of the N.B.P.

Regarding the additions to the N.B.P. it can be stated that only the compensation for a rising sea level is under discussion, due to the impact of global warming. Both the Intergovernmental Panel on Climate Change (IPCC) and the KNMI predict an average rise of around 40 cm / 100 years. If the IPCC 2007 upper limit of 54 cm / 100 years is taken, the N.O.P. for 2011 is increased by a maximum of 3 cm. For 2034 this would result in a 12 cm increase. When taking the IPCC 2007 lower limit of 16 cm / 100 years, the N.O.P. for 2011 would be decreased by 3 cm.

The evaluation of the additions to the N.B.P. shows that they not only have no ambiguities but also have no major impact on the N.O.P. Therefore, it was decided to remain with the present value of the N.O.P.

Systems, structures and components

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. The design level of 7.3 m + NAP is used as the current DBF for NPP Borssele. With this level, it can be determined whether the safety-related systems and components continue to be available in case of flooding. The present analysis shows the following:

- the three safety functions (control of reactivity, cooling of fuel and confinement of radioactivity) are guaranteed at the DBF as long as fuel for the diesel generators is available;
- 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.

Air intakes of safety-related systems located in flood-resistant buildings are placed well above the level of 7.3 m + NAP to limit the probability of a system failure due to splashing of waves.

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.

Emergency Operating Procedures and Severe Accident Management Guidelines

In case of function loss of systems or components Emergency Operating Procedures (EOPs) and (Severe) Accident Management Guides (SAMGs, see Chapter 6) are available. However, in most procedures and guides the specific circumstances related to flooding are not taken into account, although water on the site can severely hamper execution of procedures. Even relatively low levels of water will make it difficult to go from one building to another, or to reach equipment. Besides that, personnel from outside the plant will have difficulties in reaching the site. This will especially be the case if flooding situations reach the DBF of the plant.

3.1.2 Provisions to protect the plant against the design basis flood

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

Annex 1.2 gives the current building code description of NPP Borssele. The buildings that date from the original design are those with a number below 30. Buildings with a higher number have been added as a result of projects resulting from ten-year safety evaluations.

All buildings of the protected zone (buildings 01, 02, 03) plus the bunkered systems in buildings 33 and 35, are designed to withstand external events up to a certain magnitude. For flooding they are designed for the DBF of 7.3 m + NAP, as described in section 3.1.1.2. In addition, building 72 was constructed flood-proof to a water level of 8 m + NAP, even though it is not one of the bunkered buildings (01, 02, 03, 33 and 35).

No detailed studies are present concerning the dynamic effects of waves or debris. However, the allowable momentum in the design base of the bunkered buildings, including resistance against small aircraft impact and explosions pressure waves, can be considered to be higher (see Chapter 7).

Building 01

The reactor vessel and coolant system are located in building 01. This building is accessible from building 03 through a lock at 18.7 m + NAP. Building 1 itself is located in building 2 and in the remainder of the text tyherefore often referred to as building 01/02 (see Figure 3.3).

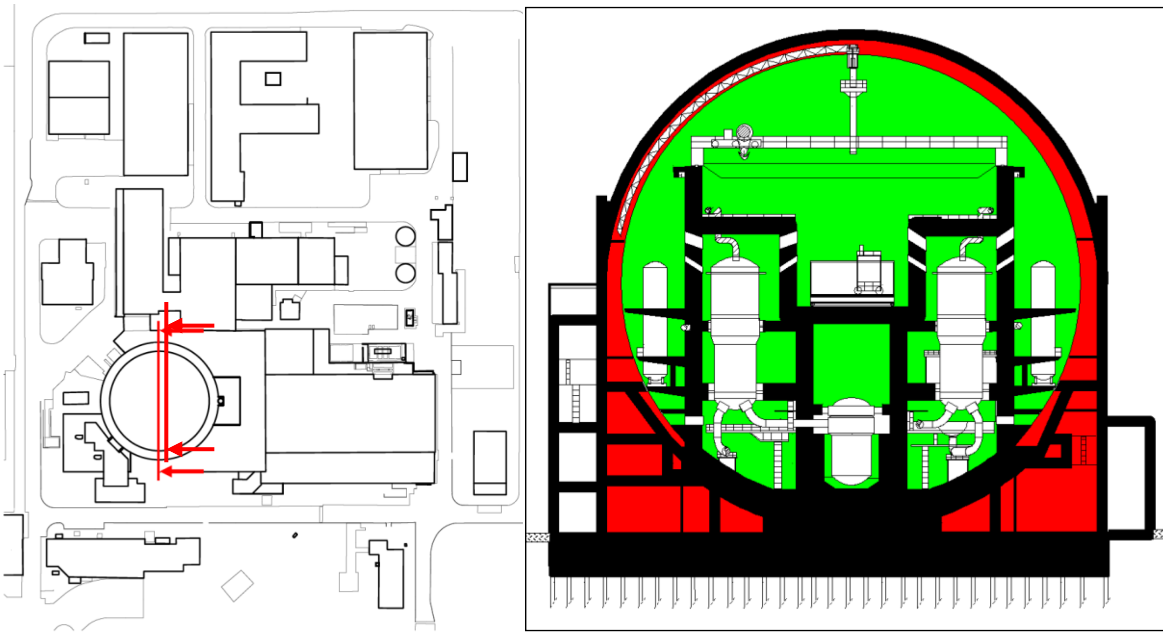


Figure 3.3 Location and cross-section of building 01 (green) and 02 (red).

Building 02

The reactor building (building 02) houses the auxiliary systems and safety systems, such as pumps and coolers of the safety injection and (backup) residual heat removal system (TE/TJ) and the spent fuel pool cooling system (TG). The flood resistance (7.3 m + NAP) depends on the flood resistance of building 03; the lowest entrance from building 02 to 03 is positioned on the ground floor at 3.2 m + NAP.

Building 03

The nuclear auxiliary building is a multi-storey building with at least four main floor levels; it is a controlled zone that contains nuclear auxiliary systems essential for the safe operation of the plant. It also provides staff access to building 02/01.

In building 03, all originally existing penetrations below the level of 7.3 m + NAP have been elevated or sealed so that no water can enter the building and therefore not into the reactor building (01/02). To this end, the following measures have been taken, including:

- sealing penetrations of the conventional emergency cooling water system (VF) pipes between buildings 03 and 04;
- elevating the position of the opening of the under pressure protection of building 03;
- sealing the cable space between buildings 03 and 04 with a high water-resistant hatch,
- inspecting the moulded rubber seals in the joints between building 02 and 03;
- waterproofing the door between rooms 05.104 and 03.141;
- waterproofing the escape door between room 03.144 and the outside.

The resistance of building 03 against water at the height of 7.3 m + NAP has been demonstrated by Siemens, and was reassessed in the last 10 yearly safety evaluation and reconfirmed in existing surveillance programmes.

Building 04

The turbine building contains the main components of the water/steam cycle system. This building is not water-tight, so water will enter the building in case flooding occurs. Important equipment is protected by not placing it at ground level. The bottom floor is at 3.2 m + NAP and houses mainly pipes. The safety relevant systems and components that are susceptible to flooding damage are located at 6.7 m + NAP and higher.

Building 05

Building 05 has six storeys and contains the electrical equipment for operating the nuclear power plant. This building is not water-tight. Flood protection is comparable to building 04. At 11.5 m + NAP 24 V DC and 220 V DC rectifiers and batteries are located, together with a part of the non-interruptible 380 V AC system with battery back-up. This floor also contains the 220 V DC control rod switch breakers. The bottom floor (3.2 m + NAP) is the basement with the cable duct to building 04. At 6.7 m + NAP part of the 6 kV and 380 V switchgear and transformer rooms are located. The main control room, the control electronics and the process-computer are situated at 18.7 m+ NAP.

Building 10 / 11 / 12 / 41

In diesel generator building 10, several components of Emergency Grid 1 are located. The diesel generator EY030 of Emergency Grid 1 is located in this building at 6.7 m + NAP as well as the 0.4 kV bus bar CV. The 6 kV bus bar BV is located on the 11.0 m + NAP floor. As discussed in section 3.1.1.1building 11 houses the generator transformer AT and its house load transformer BT, building 12 houses the auxiliary transformer BS001 and building 41 the auxiliary transformer BS002. The transformer elevations are also given in section 3.1.1.1.

Building 21

In the cooling water inlet building 21 (outside the plant's perimeter) the main cooling water pumps (VC) and the emergency cooling water pumps (VF) are located. The building is watertight up to 7.4 m+ NAP.

Building 33

Building 33, the backup systems bunker, houses the backup feedwater system RS and the backup coolant makeup system TW. Also the Reactor Protection System (RPS) sub-system for safety-related process parameters and its 24 V DC battery backup systems are located here. The cable ducts from building 33 enter building 01/02 at 20 m+ NAP. Entrance doors are located at 8.55 m + NAP. The air inlet of the building is placed at a height of 9.6 m + NAP. The air intakes of the diesel generators (Emergency Grid 2) located in building 33 are elevated at a height of 9.8 m + NAP. Two internal gooseneck pipe/cable corridors up to 7.3 m + NAP, prevent water influx in case of a failure of the MCT Brattberg transits used in the pentration at 3.2 m + NAP.

Building 34

The waste storage building serves as the interim storage of radioactive waste. Slightly radioactive contaminated materials such as tools and scaffolding can also be stored at this location. A concrete waterproof wall with a height of 9.2 m + NAP was constructed to protect the storage area against flooding.

Building 35

The remote shutdown building (35) houses the emergency control room (see Annex 3.5) as well as parts of the 24 V DC system, the I&C of the RPS and the plant-internal parts of the emergency communication systems. Similar to building 33, the staff entrances are located above the design flood level at 8.55 m + NAP. Internal gooseneck pipe/cable corridors up to 7.3 m + NAP, prevent water influx in case of a failure of the MCT Brattberg transits used in the penetration at 3.2 m + NAP.

Building 72

This building houses several components of Emergency Grid 1. The diesel generators EY010 and EY020 are located on the floor at 8 m + NAP The rotating converters (ER010 and ER020) are located on the floor at 3.7 m + NAP. The cable duct between building 72 (diesel generators) and building 05 (bus bars) is sealed at the location of building 72. The building is constructed water-tight and water can enter only through the air intakes located at 8 m + NAP.

Buildings and systems not available during flooding

The buildings and systems mentioned above are within the defence concept against flooding. Other buildings and systems often used in EOPs, but not available under flooding conditions, are :

- the Alarm Coordination Centre (ACC) (in the basement of building 15). This will flood and become inaccessible long before the 5 m + NAP level is reached. In such a case, room 05.615 (building 5 at 22.7 m + NAP) will serve as an emergency response centre back-up ;
- the fire-extinguishing system UJ and other water sources, like the Demineralised water plant UA. The pumps needed to use these water supplies become flooded long before the 5 m + NAP level is reached;
- the fire fighting pump. This is an adapted crash tender that is not capable of operating in water deeper than 50 cm;
- the mobile emergency diesel generator EY080. This generator is placed on a lorry and as such is not protected against flooding. This, combined with its limited ‘on board’ fuel storage, disqualifies it for use during flooding conditions.

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

The site (see Figure 3.4) is enclosed by three dykes.

- **Dyke A** The west side of the site is located directly behind the sea dyke. This dyke has a crest height of 9.4 m+ NAP;
- **Dyke B** The north side is bounded by an apparent inland dyke (crest height at 7.65 m + NAP) separating the site from the industrial area of the Sloe harbour. As the harbour is open to the Westerschelde this dyke is also a sea dyke;
- **Dyke C** At the south and east side an inland dyke forms the border between the site and the Borssele polder. The crest height of this dyke is around 4 m + NAP.

The ground level of the area enclosed by dyke A, B and C is approximately 3.0 m+ NAP.

Sea dykes A and B are part of the dyke ring protecting a part of Zuid-Beveland 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. However, the protection against wave erosion will be further improved so that the dyke will not fail in storm situations with an average return period of 10,000 years. In these situations the dyke will still limit the fetch as discussed in section 3.1.1.2.

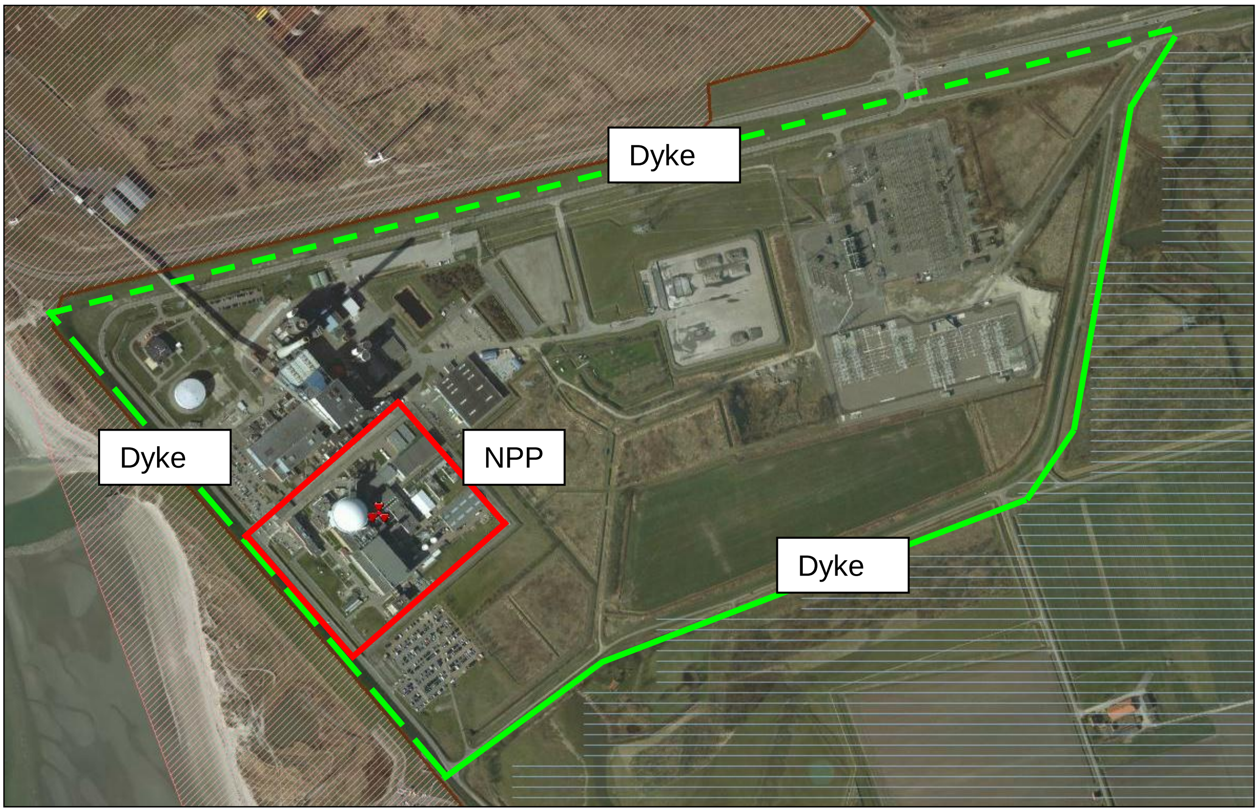


Figure 3.4 Location of NPP Borssele and its surrounding dykes (source: www.risicokaart.nl).

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. 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.

At a water level of 4.3 m + NAP, the district water board 'Waterschap Scheldestromen' informs EPZ on the current situation and keeps EPZ informed regarding developments that may require intervention. At that moment, the Emergency Response Organisation (ERO) of EPZ is already in place (initiated by S-VF-01). The ERO will monitor the forecasts of the RWS regarding the course of the situation and the water levels that can be expected.

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

1. Cooling down to 'cold shutdown' mode.
   All cooling options remain available in case of LOOP, plus most efficient use of cooling water.
2. Mobilisation of additional personell (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.

This decision model must be based on the current situation such as (un)availability of the cooling water inlet system and on the actual water level on the site of NPP Borssele combined with forecast data provided by the RWS.

To control a situation in which all systems have failed, Severe Accident Management Guidelines (SAMGs) are available (see Chapter 6). SAMGs are symptom oriented (NOT event orientated) and therefore flooding conditions are not taken into account specifically. Although this is considered a strength of the SAMG's, it is recommended to develop a set of Extensive Damage Management Guidelines (EDMG) focused on flooding conditions.

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

As mentioned in section 3.1.2.3, mobilization of additional personell is necessary in case a dyke failure is to be expected within the Zeeland region. 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 personell (operators en 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. Telephone (conventional and mobile), internet and C2000-system can be considered lost at a few centimeters of flooding (somewhere in the region). This reinforces the precaution measures mentioned in section 3.1.1.2, i.e. mobilization of personell and extension to the alarm procedure, as well as the mobile accident management precautions mentioned in Chapter 6 (for example satelite telephones).

Whereas the weather must be so extreme in order to generate dyke failure and flooding, also 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 (see Chapter 4).

3.1.3 Plant compliance with its current licensing basis

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

Ensurance of design levels

The flood design levels are evaluated every ten years, during the 10 yearly safety evaluations. The most recent evaluation was held in 2003. Based on the outcome of these evaluations, modification projects are initiated.

The modifications based on the evaluation of 1993, included the following flood related measures:

- sealing of all openings up to a level of 7.3 m + NAP in buildings 01/02, 03 and 33;
- designing the new building 72 flood proof up to a level of 8 m+ NAP.

Based on the evaluation of 2003, the following modifications were carried out:

- raising of the air intakes for the diesel generators in building 33 to a level of 9.8 m + NAP. This measure removed a potential cliff-edge;
- reintroducing the Low Flood Concept (5 m + NAP, see section3.1.1.1) for building 21 by incorporating this concept in the Technical Information Package (TIP) and by introducing a set of maintenance and inspection procedures. This modification increases the safety margin by maintaining the additional barrier (defence in depth);
- installing a second pump for the Backup residual heat removal system (TE). This measure compensates for the first TE pump failing;
- defining additional inspections and audits to maintain the water tightness of building 02 and 03.

Surveillance programme

The surveillance programme of NPP Borssele dictates the following actions to ensure the design levels:

- inspections of structural measures taken in the protected area (building 01/02, 03, 33, 35 and the wells of the Backup cooling water system(VE) so as to ensure a safe shutdown during a flood level of the DBF;
- inspections of buildings 10, 21 and 72 to ensure functionality during a flood level of 5 m + NAP;
- inspections to check that the basins of the Demineralised water supply system (RZ) and the Conventional component cooling water system (VG) are waterproof.

Inspections are carried out at least every four years.

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

As mentioned in 3.1.2.4 (and Chapter 6), 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.

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. In section 3.1.1.2 the evaluation of the N.O.P. is discussed and it is concluded that a moderate change is possible. Depending on the exact outcome of this evaluation appropriate measures shall be taken. These measures can include for example raising air intakes, improving strenght and water tightness of certain buildings or installing recurves or parapets[18] to prevent large wave reflections against vertical walls.

[18] In general a recurve or parapet is a design feature of a seawall or breakwater in the form of a seaward overhang. It is constructed to reduce overtopping.

3.2 Evaluation of margins

3.2.1 Estimation of safety margin against flooding

3.2.1.1 Description of the plant condition up to 5 m + NAP

Structures (buildings), systems and components

The buildings ensuring the plants resistance to a flooding level of 5 m + NAP (see Figure 3.1 in section 3.1.1.1) are:

- all buildings of the controlled area (01/02 and 03);
- buildings 04, 05 and 10 (steam/water cycle, main control room and electrical systems);
- building 33 and 35 (RS, TW, YZ (RPS), diesel generators and emergency control room);
- building 11, 12, and 41 (main transformers) ;
- building 34 and 72 (waste storage and diesel generators);
- building 21 (cooling water inlet).

Up to an on-site water level of 5 m + NAP all systems that can be used to reach safe shutdown and to remove decay heat are available, including the main condenser and connection to off-site power (availability of external grid is questionable). The plant can still be operated from the main control room.

All (safety) systems are located in water-tight buildings (01, 02, 03, 21, 33, 35, and 72) or are placed at sufficient levels of elevation in buildings 04, 05 and 10 (minimum elevation is 6.7 m + NAP).

Loss of off-site power is backed up by:

- the plant house load power supply (via the main generator);
- three diesel generators (from Emergency Grid 1) which are redundant and located in building 72 (EY010 and EY020) and building 10 (EY030).

The fuel tank of EY030 is located on the 3.6 m + NAP floor of building 10. The room where the tank is placed will confine the diesel fuel in case of tank leakage or rupture. The entrance of the room is at approximately 4.6 m + NAP. The tank has no openings, so water cannot enter the tank, even as the water exceeds 4.6 m + NAP. The tank is secured to the floor and if the room is flooded, diesel will still be available.

Safety systems to withstand external events (RS, TW, YZ (RPS)) and the emergency control room) are in standby condition and have two dedicated additional diesel generators (EY040, EY050) to supply electrical power.

In normal situations, the above-mentioned safety systems in buildings 33 and 35 are fed by two (one per redundancy) 10 kV / 0.4 kV power transformers located in building 33. The 10 kV feeders to building 33 arrive from 10 kV sub-station AL19BF (Red. 1) on the premises of the coal fired plant BS12 (CCB) and AL19BE (Red.2) which is located in building 40 on the NPP Borssele premises. This sub-station becomes unavailable at a water level of 4.3 m + NAP. As a result the diesel generators EY040 and EY050 (of Emergency Grid 2) will start automatically and restore electrical power in buildings 33 and 35. In the unlikely event that both EY040 and EY050 fail, it is possible (by following EOPs) to feed Emergency Grid 2 from:

- Emergency Grid 1 using the transformers CT15 and CT16 on the 12.3 m + NAP level of building 33, connecting respectively bus CW to bus BU and bus CX to bus BV;
- the standby mobile diesel generator EY080 ;
- an external source, for example from the CCB.

In case of flooding the last two options will be very uncertain. Loss of off-site power and station black-out scenarios are described in detail in Chapter 5.

The heat sink is provided for by VF/TF. The VF pumps are located in building 21 at 4.3 m + NAP Water can only penetrate this building at a level of 7.4 m + NAP. However, building 21 is constructed to resist the oppressive force of water and is designed to be flood proof up to a level of 5 m + NAP. For scenarios on loss of the ultimate heat sink is referred to Chapter 5.

Occurrence frequency

A flood level of 5 m + NAP has a return period of approximately 900 years ($1.1 \cdot 10^{-3}$ per year). The return period for flooding of the site will be lower, as the plant is protected by dykes. The dykes are designed to withstand a flood level with a return period of 4,000 years, which is higher than 5 m + NAP This means that the failure probability of the dykes at a flood level of 5 m at the Westerschelde will be much smaller than one.

Cliff-edge effects at 5 m + NAP design level

There are no cliff-edge effects in this situation because:

- the plant is designed to withstand a much higher level of flooding, namely 7.3 m + NAP;
- between 5 m and 7.3 m + NAP a gradual loss of safety systems will occur. These situations are described in section 3.2.1.2.

3.2.1.2 Description of the plant condition between 5 m and 7.3 m + NAP

Between 5 m and 7.3 m + NAP two critical levels can be discerned with respect to loss of systems:

- 5 m + NAP;
- 6.7 m+ NAP.

Situation between 5 m and 6.7 m + NAP

Structures (buildings), systems and components

Although the cooling water inlet building (21) is water tight up to 7.4 m + NAP, it has been designed to withstand a static water level of up to 5 m + NAP. It is therefore assumed that this building is lost at higher water levels. This means that at water levels above 5 m + NAP the normal heat sink via VC and VF will be lost.

Although buildings 04, 05 and 10 have the same design level as building 21, these buildings can withstand much higher water levels as water can freely enter these buildings, in contrast to building 21 which is water-tight. The free influx of water limits the load on the walls of the buildings. Therefore the main control room remains available.

As a result of losing VC/VF:

- closed loop secondary cooling via the condenser is no longer possible;
- the residual heat removal system TJ can not be used because the nuclear component cooling water system TF, upon which TJ relies, can no longer be cooled;
- the spent fuel pool cooling system TG can no longer be cooled via the normal method using the TF/VF systems;
- the plant's own main generator cannot be used to supply electrical power.

Electrical power can be provided by:

- the external grid (if available);
- the diesel generators EY010 and EY020 (feeding Emergency Grid 1 and Emergency Grid 2). Diesel generator EY030 (which is the backup diesel generator for EY010 and EY020) is not available because it is cooled by VF. Emergency Grid 2 can also be fed by its dedicated diesel generators EY040 and EY050. The cable connecting diesel generators EY010 and EY020 in building 72 to building 05 (Betobar bus bars) has no splices and can operate under water (IP68.7). The cable duct is sealed at the location of building 72.

Cooling down through secondary side cooling can be maintained by using the main or auxiliary feedwater systems (RL-Main and RL-Emergency) in combination with the secondary blowdown valves of the main steam system (RA). The feedwater inventory of RL is not sufficient to reach the decay heat removal phase. The demineralised water supply system RZ can be used to replenish RL[19]. The alternative water source is the backup feedwater system RS (with its own pumps and power supply) located in building 33, which is resistant to external events. The water inventory of RS is enough to reach the decay heat removal phase. Water from the fire-fighting system, the main condensate system (RM) or the demineralised-water storage cannot be relied on in case of flooding, as the electrical power supply, pumps and valves are not designed against flooding and are located below the 5 m + NAP level.

[19] One should be aware, that because TJ cannot be used, it will take more secondary water to reach the decay heat removal conditions that TE/VE requires.

Decay heat removal is maintained via TE/VE. The spent fuel pool can still be cooled via TG, but TG080/VE has to be used instead of TF/VF.

In case of LOOP the diesel fuel stored is enough to cover a 72-hour period. Please refer to Chapter 5 for details.

Cliff-edges

There are no cliff-edges as the ultimate design level for flooding is not reached in this situation.

In the low high-water concept (5 m + NAP), 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.

Situation between 6.7 m and 7.3 m + NAP

Structures (buildings), systems and components

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. As a result of the flooding, the bus bars BA/BB/BU[20] and their dependent 0.4 kV bus bars CU and CV will become unavailable. In this case the battery backup for all DC consumers (in majority I&C and MOV's) on the 11.5 m + NAP floor will kick in and remain operational for a guaranteed period of two hours. In reality there is a rather large margin here because the batteries' discharge time is much longer than two hours. In this situation the Emergency Grid 2 in building 33 will remain available and consequently the systems (fed through Emergency Grid 2) that will be operational are those in the buildings 01, 02, 03, 33, 35 and the VE wells. The diesel fuel storage of Emergency Grid 2 is sufficient for 72 hours.

Building 72 houses the diesel generators EY010 and EY020. Although they are still both functional, only EY020 can be used as its dedicated bus bar BV (located in building 10 on the 11 m + NAP floor) is not flooded. Although the systems fed by this bus bar (TF, residual heat removal via TJ, RL pumps) cannot be used anymore because of loss of VF or because the system itself is flooded (RL), the diesel generator can be used as backup for the diesel generators of Emergency Grid 2. The use of EY020 secures the availability of electrical power with at least another 72 hours but this possibility depends on the survival of building 10. Safety injection via TJ remains available for one redundancy.

Cliff-edges

The cliff-edge of 6.7 m + NAP inside a building corresponds to a significantly higher dynamic water level outside because the absence of waves. Note that a static level of 6.7 m + NAP corresponds to a return period in excess of 1.000.000 year. However, most of the 6 kV / 0.4 kV

[20] Apart from the loss of the bus bars BA/BB, the external grid feeding these bus bars is lost as the switch yard is flooded.

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 guarantueed. But its functionality is to be expected because of the availability of (part of) Emergency Grid 1, rectifiers, batteries and the dispatcher.

3.2.1.3 Description of the 7.3 m + NAP situation

Structures (buildings), systems and components

The buildings resistant to a flooding level of 7.3 m + NAP (see Figure 3.1 in section 3.1.1.1) are:

- all buildings of the controlled area (01/02 and 03);
- building 33 and 35 (RS, TW, YZ (RPS), diesel generators and emergency control room);
- building 34 and 72 (waste storage and diesel generators).

External flooding with a water level of the DBF (7.3 m + NAP) is covered by the RS-concept. The systems required for reactor shutdown and long term cooling (and available at 7.3 m + NAP flooding level) within this concept are:

- Electrical power supply: Emergency Grid 2;
- Backup cooling chain (TG080/VE, TE/VE),
- Backup spent fuel pool cooling system (TG080);
- Backup residual heat removal system (TE);
- Backup cooling water system (VE);
- Backup coolant makeup system (TW);
- Backup feedwater system (RS);
- Reactor protection system (YZ) ;
- Chilled water system (UV) ;
- Emergency control room.

Diesel generator EY020 is available as backup for the diesel generators of Emergency Grid 2.

The availability of the main control room is not guarantueed. 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 distinction is made for initial operating modes. The determining factor is whether or not the reactor vessel lid is closed. The following situations are observed:

- **Reactor head closed.** For decay heat removal (secondary feed and bleed), systems RS and RA are used. This process is self-sufficient (autarkic) for at least ten hours. Shrinkage of the primary coolant occurring in the same period is compensated by the TW system. During the autarky period, hot subcritical operating conditions are maintained automatically. The total amount of RS water is sufficient for a 72-hour period. 13.5 Hours after scram, the system can be cooled further to cold subcritical conditions by manually switching over to the backup decay heat removal system TE/VE. The part of the spent fuel pool cooling system that is resistant to external events, TG080, may even be activated immediately after scram. Water for systems RS and TW, both non-borated and borated, is stored in tanks in building 33. The energy supply must be secured by the

Emergency Grid 2 and the two associated diesel generators EY040 and EY050 due to the failure of the external 10 kV power supply.

- **Reactor head open.** Heat dissipation through the steam generators is not possible because the pressure in the reactor coolant system is atmospheric. Due to the increasing temperature in the primary system as heat removal via TJ is lost, a reactor protection signal is generated that will start the TW pumps. The decay heat is removed by evaporation to the containment. Furthermore the manual actions are largely similar to the situation in which the reactor lid is closed, i.e. switching over from TJ cooling to TE/VE cooling. The waiting time of 13.5 hours after scram does not apply because the decay heat is greatly reduced in the period between scram and opening of the reactor head.

Occurrence frequency

The frequency of occurrence of a dynamic flooding up to a level of 7.3 m + NAP is 10^{-6} per year.

Cliff-edges

Possible cliff-edges are discussed in section 3.2.1.2.

3.2.1.4 Description of the situation beyond 7.3 m + NAP

When the dyke in front of the plant has failed, the reduced wave height is not applicable, as is discussed in section 3.1.1.2 . In this situation, a storm surge with a return period of 1,000,000 years (N.B.P. + surcharges: 6.5 m + NAP) and a wave height of 3.9 m[21] will lead to a dynamic water level of 8.45 m + NAP (without reflection to the walls).

Structures (buildings), systems and components

Flooding of building 03

Building 01 is completely sealed. Building 02 itself is completely water-tight on the outside. The weak points are the doors between building 02 and building 03. These internal doors are not water-tight and are on an elevation level of 3.2 m + NAP.

Building 03 was originally designed to withstand a water level of 5 m + NAP. After several alterations this was increased to 7.3 m + NAP. Strength and water tightness calculations are not available above this level, so it is assumed that this building will be flooded above a dynamic water level of 7.3 m + NAP.

The consequence of flooding building 03 is that building 02 is also flooded up to a static level of about 6.5 m + NAP (7.29 m + NAP - 0.8 m due to wave run, see Annex 3.4). This means that the pumps of the Backup residual heat removal system (TE) become unavailable as a result of flooding of electronics. However, the pumps can be restarted in building 33 using a forced start. The pumps themselves are waterproof. Also, the spent fuel pool cooling system pumps (TG) are not affected, as they are located on the 13.7 m + NAP floor of building 02. The ground-water well pumps of the Backup cooling water system (VE) are not affected by flooding of building 03 and are capable of withstanding all flood levels. So flooding building 03 will not worsen the situation (system wise) in comparison to a flood just below the 7.3 m + NAP level:

- closed loop cooling for residual heat removal via TE/VE is still possible because TE remains available;
- spent fuel pool cooling remains possible, as TG remains available;
- cooling can be maintained by the use of RS and the secondary blow-down valves (RA). If the RS water supply is depleted, it is possible to switch to primary feed and bleed through TW and the pressuriser valves of the Pressure control system YP.

In case the residual heat removal via the closed loop cooling possibility TE/VE is lost, cooling can be maintained by primary feed and bleed using TW and the pressuriser valves or by switching back to secondary cooling by RS and RA.

In principle, the water inventory of RS is sufficient for at least 72 hours. In open loop cooling, a supply of water after this period will be necessary to prevent core damage. Water inventory of TW is sufficient for at least ten hours at maximum injection rate. For the specific scenarios, not

[21] Maximum wave height in the Westerschelde near Borssele corresponding to return period of 1,000,000 year (communicated by district water board 'Waterschap De Scheldestromen').

coinciding with a Loss-of-Coolant Accident (LOCA), available mission time for TW is much longer and at least comparable with RS.

The fuel supply for the diesel generators of Emergency Grid 2 is also sufficient for a 72-hour period. The use of diesel generator EY020 of Emergency Grid 1 adds at least another 72 hours. In a closed loop cooling situation, a supply of fuel after this period is necessary to prevent core damage.

Flooding of building 72

Under the condition that bus bar BV remains available, electrical power from diesel generator EY020 in building 72 will be available up to a flood level of 8 m + NAP, when the air intake will become submerged. Losing building 72 will not lead to loss of systems, but will shorten the time electrical power can be delivered to Emergency Grid 2.

Flooding of buildings 33 and 35

In both redundancies of building 33, the pipes of the backup cooling water system VE enter the building through penetrations at approximately 3.2 m + NAP. Every pipe rises inside building 33 through an open shaft with an internal height of 7.3 m + NAP. Water-tightness above 7.3 m + NAP depends on the water-tightness of the penetrations of pipes and cables at 3.2 m + NAP. These penetrations are fitted with MCT Brattberg transits to provide water-tightness. A similar construction is used in building 35. This makes building 33 and 35 resistant to a level of at least 8.55 m + NAP, at which elevation the access doors to buildings 33 and 35 are located. Air intakes of building 33 are above this level at 9.6 m + NAP.

If the MCT Brattberg transits leak, building 33 and 35 would be resistant to flooding up to a static water level inside the bunker of approximately 7.3 m + NAP, which would correspond with a dynamic water level of approximately 8.1 m + NAP outside. As the RS and TW pumps and the diesel generators are located at the lowest level of building 33, these systems will be lost very quickly. This is also true for the plant's internal emergency communication systems. Flooding will not be immediate, but will take time depending on the leak rate, as the pipe shaft has to be filled up first. Moreover, it will also depend on the duration of the water level being above 8.1 m + NAP.

If water enters the building through the pipe shaft, the DC batteries are not affected, so the emergency control room and YZ might still be available. The batteries, however, will no longer be charged. If the water enters the building through the doors then also the batteries, YZ and the emergency control room are lost (building 35).

The availability of the batteries is only of importance for the instrumentation. Although YZ is available, there is no power to operate the necessary valves automatically. The resulting situation is station blackout. As no electrical power is available to feed the pumps, closed loop cooling of the RCS via TE/VE and cooling of the spent fuel pool via TG/VE is lost.

When the dyke in front of the plant has failed, a storm surge with a return period of 1,000,000 years will lead to a dynamic water level of 8.45 m + NAP (when not considering reflection to the walls). The concrete landing in front of the entrance doors of building 33 serves as a parapet which deflects waves and will prevent large amounts of water to leak into the building through the entrance doors at 8.55 m + NAP. However, reflection of waves will cause leakage into the building through the air inlet in the wall at 9.6 m + NAP. It is also possible that water enters the downwards oriented air intakes of the diesel generators at 9.8 m + NAP.

Splashing of waves against the air intakes will lead to small amounts of water entering building 33. This however, does not directly lead to problems; the diesel generators are placed at a small elevation and on the lowest floor of the building drainage pumps are placed. Improvement is possible by constructing a recurve of parapet just below the air intakes so that waves cannot splash against the air intakes.

Reactor Core

The remaining available method to cool the reactor core is secondary cool down via hand-operated RA blow-down valves. The valves will be accessible given that building 03 is accessible. Additional water can be supplied to the steam generators by a mobile (fire-fighting) pump using an entry point in building 33. However, these actions are not guaranteed, given the flood level.

The time available before additional water MUST be supplied is approximately 2.5 to 3 hours under worst case conditions. After the approximately 50 to 60 minutes it takes to boil off the steam generators, the primary safety valves will start to open / close automatically. This primary blow-down will lead to uncovery of the top of the core in approximately 120 minutes.

If the batteries of Emergency Grid 1 are available there could be more time, because motive power would be available for a limited number of components. The batteries could be available as the whole Uninterrupted Power Supply (UPS) system is located on the 11.5 m + NAP floor of building 05. However the strength of building 05 is uncertain and with the batteries no longer being charged, time is limited.

Spent Fuel Pool

The flooding of building 33 will lead to the loss of all AC power and with the loss of AC power the spent fuel pool can no longer be cooled, as the TG and VE pumps will stop. Boiling of the water in the spent fuel pool will start after 16 hours at the earliest. The top of the spent fuel assemblies will become uncovered after a period of between, 80 hours (complete unloaded core stored in spent fuel pool) to over two weeks in the case of one third of a core, which is a more realistic but still conservative assumption.

Occurrence frequency

The frequency of occurrence of a flooding level higher than 7.3 m + NAP is less than 10^{-6} per year.

Cliff-edges

Cliff-edges are difficult to identify above a flooding level of 7.3 m + NAP, because a margin may exist before buildings (and therefore systems) are lost.

Strength and water-tightness calculations are not available above the level of 7.3 m + NAP for building 03, therefore this building is considered to be flooded above this level. As a consequence, building 02 is also flooded; however the impact on the situation is minor.

For building 33, water-tightness above 7.3 m + NAP depends on the water-tightness of pipe and cable penetrations at 3.2 m+ NAP. If these penetrations remain water-tight, the next possibility for water to enter will be at 8.55 m + NAP through the entrance doors. Water-tightness of these doors is unsure. This also applies to building 35. If the doors are water-tight water will enter building 33 at 9.6 m + NAP, the height of the air inlet of building 33.

Evaluation of the margins and possible improvements

For the as-built situation, the availability of systems at various levels of flooding is depicted in Figure 3.5.

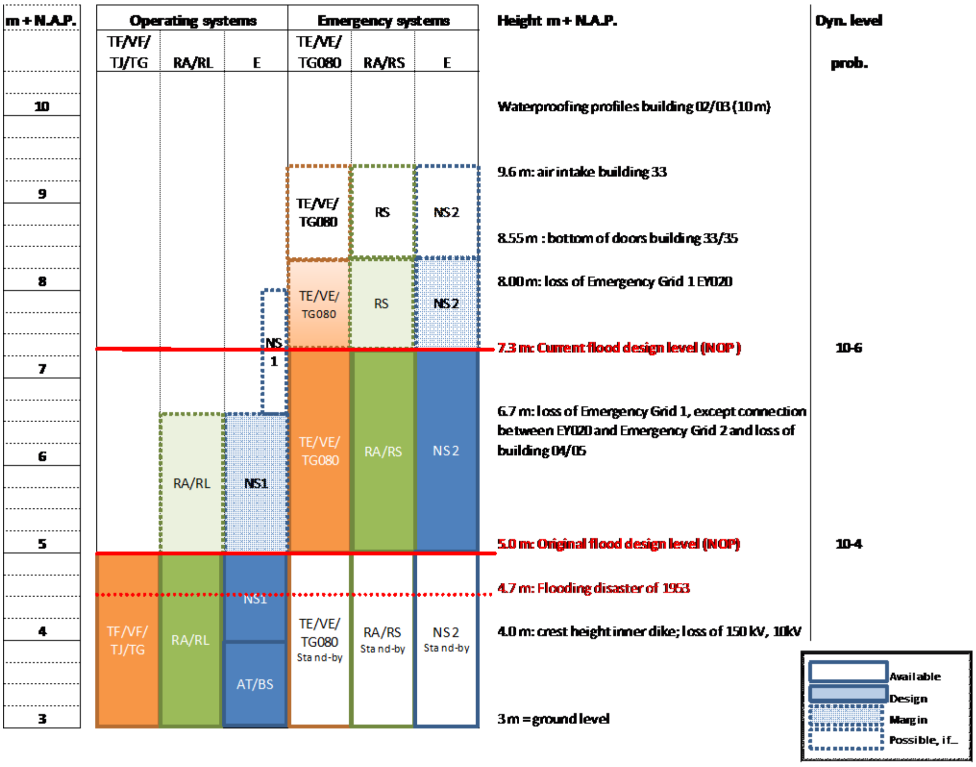


Figure 3.5 Availability of systems at various flooding levels.

For flooding levels above 7.3 m + NAP, a margin of 8.55 m + NAP exists for systems RS and TW due to the flood resistance of building 33. At these flooding levels (with a probability of less than 10^{-6} per year) the systems that are especially designed to cope with flooding situations are available. The three safety functions (control of reactivity, cooling of fuel and confinement of radioactivity) are guaranteed providing fuel for the diesel generators is available.

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

As discussed dyke A will be improved in 2012. The failure frequency of the dyke will remain once in 4,000 years. However, the protection against wave erosion will be further improved so that the dyke will not fail in storm situations with an average return period of 10,000 years. In line with foreign countries no guarantee can be given that the dyke will not erode and breach due to increased overtopping with an average probability of once in 1,000,000 years.

Regarding structural measures, 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.

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.

To improve plant robustness during actual flooding situations the following measures can be taken:

- develop a set of Extensive Damage Management Guides (EDMG) and implement a training programm. 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 (all forseeable hazards) would increase the options of the alarm reponse organisation;
- storage facilities for portable equipment, tools and materials needed by the alarm response organisation that are accessable after flooding (all forseeable hazards) would increase the options of the alarm reponse organisation;
- establising 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.

3.3 Flood sources

The possible flood sources are:

- high tide, possibly combined with a storm surge;
- tsunami ;
- extreme rainfall ;
- VC pipe rupture.

High tide / storm surge

A high tide, possibly combined with a storm surge, may lead to flooding. The flooding level (N.B.P.) caused by high tide with an incidence of 10^{-6} per year is 6.18 m + NAP, based on the High Tide Total Exceedance Frequency chart (Annex 3.2). From the N.B.P., the dynamic N.O.P. is calculated at 7.29 m + NAP by adding several factors as discussed in section 3.1.1.2.

In addition, the flooding impact may be influenced by the failure of sea dykes. Dykes fail as a result of a large amount of water combined with other factors; for example, due to run-over (dyke too low for static water level), wave overtopping (waves too high or too much run-up) causing leeside erosion, piping (local groundwater flow), sediment transport and erosion below/behind the dyke.

Tsunami

Though rare, the North Sea has been the site of a number of historically documented tsunamis (see Figure 3.6):

3. Storegga Slide (8200 calBp)

Large parts of the now submerged North Sea continental shelf (Doggerland) were flooded by the Storegga Slide tsunami, one of the largest tsunamis that took place during the Holocene, which was generated on the Norwegian coastal margin by a submarine landslide. The characteristics of the wave triggered by this ancient event have been simulated. A calculated initial wave height of 3 m at the source of the model resulted in maximum deviations of about 0.5 to 0.7 m at the tidal gauges in the German Bight. It would have taken approximately 8.5 hours for the first wave to reach the German coast-line.

Geological models suggest that for a tsunami such as the one following the Storegga slide, another glaciation (time scale ~100,000 years) is needed to re-establish the conditions required for a similar failure at that location. However, there are other sections of the neighbouring continental slope that have the potential for a landslide, possibly triggered by an earthquake.

4. Lisbon Earthquake (1755)

The tsunami triggered by the 1755 Lisbon earthquake reached Holland, although the waves had lost their destructive power. Waves at the origin had an amplitude of 1 m. After five to eight hours, waves with a height of 0.8 to 2 m reached the coast of Cornwall with localised amplification enhancing the elevations to approximately 4 m. A tsunami entering the North Sea from the English Channel will not have any severe consequences in the North Sea, since this wave will be reflected and dampened in the English Channel.

5. Dogger Bank earthquake (1931)

This earthquake measured 6.1 on the Richter scale and caused a small tsunami (wave amplitude of 1 m at the origin). After one to two hours, waves with a height of 0.8 to 2 m reached the Yorkshire and Humberside coastlines.

Furthermore, a possible source of a tsunami in the future has been identified:

6. La Palma landslide

Research has suggested that the western flank of La Palma Island is vulnerable to collapse. British coastlines likely to be affected by a tsunami arising from such a collapse are those of Cornwall and Devon. Wave elevations of 2 m (with an estimated original amplitude of 1 to 2 m) can be expected to reach the British coast in seven to eight hours. Again, a tsunami entering the North Sea through the English Channel will not have any severe consequences in the North Sea, since this wave will be reflected and dampened in the English Channel.

Possible impact at Borssele

In a study carried out in 1993, it was concluded that a hypothetical tsunami would result in a maximum elevation of the water level of 1.4 m along the Dutch coast. Based on this conservative assumption, the risk of flooding due to a tsunami is regarded as non-existent, because a tsunami combined with the most extreme recorded storm surge (4.7 m + NAP, 01/02/1953) would result in a water level of 6.1 (1.4 + 4.7) m + NAP, which is still below the DBF of 7.3 m+ NAP.

This conclusion is supported by more recent research. In 2007 a study concluded that Cuxhaven (German Bight) is protected from the catastrophic impacts of a hypothetical tsunami of 0.5 m. As far as the Belgian coast is concerned, research concluded in 2005 that a hypothetical tsunami will not grow to an amplitude of several meters but to a maximum of 0.7 m, due to damping in the relatively shallow North Sea. This makes a tsunami, in the first approximation, of the same order as the 1953 storm surge. Therefore it was concluded that the Belgian coast, including the Westerschelde, is at a lower risk of a potential tsunami compared to other extreme meteorological effects. This conclusion was confirmed in a recent benchmark with NPP Doel (Belgium).

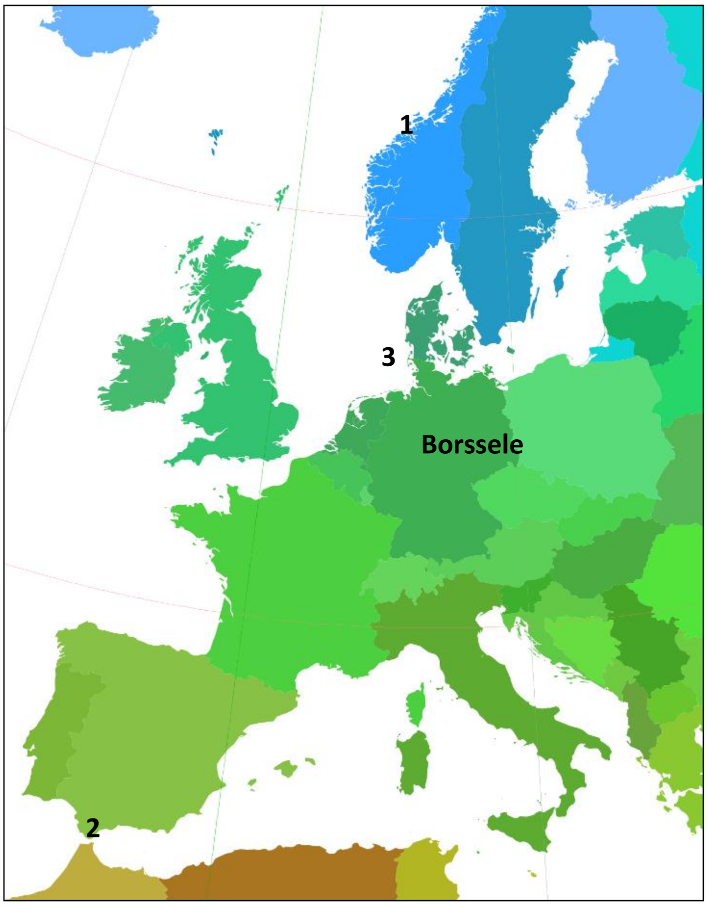


Figure 3.6 Historic locations of tsunami origins.

Extreme rainfall

Heavy rainfall may cause flooding. The average annual rainfall between 1961 and 1990 was 731.2 mm. Statistics from the KNMI show that, for the area of Borssele, 108 mm of rainfall can be expected within a period of 24 hours once every thousand years (10-3 per year). Under the same conditions, 123 mm of rainfall can be expected within 48 hours. In the climate scenarios given in KNMI ’06, the worst-case scenario of expected rainfall within 24 hours by the year 2050, predicts an increase with 19 mm. The largest amounts of rainfall within 24 hours measured in the vicinity of Borssele in the period 1951-2010 were approximately 81 mm at Vlissingen and 93 mm at Schoondijke.

For the site to flood up to a level of 5 m + NAP, an unrealistic amount of rainfall (2,000 mm) would be required. Therefore, based on the statistics mentioned, such an event is not considered credible.

The influence of extreme rainfall on the integrity of structures of the plant is treated in Chapter 4, Extreme Weather.

VC pipe rupture

The cooling water inlet building is shared between the NPP and the conventional plant. Both plants have their own main cooling water pipe. The lay-out of protective measures is the same for both plants. Two scenarios can be discerned:

- a break or rupture on site (on the landward side of the dyke);
- a break or rupture on the seaward side of the dyke.

Rupture location on site

A burst or rupture on site of the main cooling water inlet or outlet lines (VC) could lead to flooding of the NPP area. To prevent flooding a spring loaded, oil-dampened vacuum aerator is mounted at the highest point of the intersection of the VC inlet and outlet line with the dyke. The aerator breaks the siphoning in case of a pipe rupture after shutting down the pumps. The aerators are additional to motor-operated valves on the pressure side of the pumps. Tripping the main cooling water pumps has to be done manually, based on the plant's reaction to (partly) losing condenser cooling. If the main cooling water piping of the adjacent conventional plant fails, operators from that plant have to react to stop the flow of water.

The piping of both Main cooling water systems (VC), as well as the piping of the Conventional emergency cooling water system (VF), are located close to each other. Failure of one system can therefore lead to the failure of the others. A loss of primary heat sink scenario can be the result. Loss of the Conventional emergency cooling water system (VF) in the case of the failure of the Main cooling water systems (VC) will become impossible as from 2012, due to the restructuring of the VC and VF piping. The vulnerability of the plant to loss of heat sink is discussed in Chapter 5.

A conservative assessment of the impact of pipe rupture has been made. Taking the 30 minutes autarky time into account before action is taken and the full capacity of the five VC pumps, 52,000 m^3 of water can be pumped into the NPP area that is enclosed by dyke A, B and C (the area indicated in Figure 3.4). This area is approximately 0.7 km^2. The area of buildings that may be deducted from this figure is assumed negligible. Consequently, the water level in the enclosed area would theoretically have increased by approximately 0.08 m.

It can be concluded that a VC pipe rupture does not contribute substantially to the flooding of NPP Borssele, because:

- 3.08 m + NAP (ground level of 3 m + NAP + 0.08 m) is negligible compared to the original design flood level of 5 m + NAP;
- the VC pumps are most probably shutdown much earlier than 30 minutes after the pipe rupture;
- the water level can never exceed 4 m + NAP due to the height of dyke C (in the highly unlikely event that the VC pumps do not switch off), which is below the original design flood level of 5 m + NAP, so no safe shutdown equipment is affected.

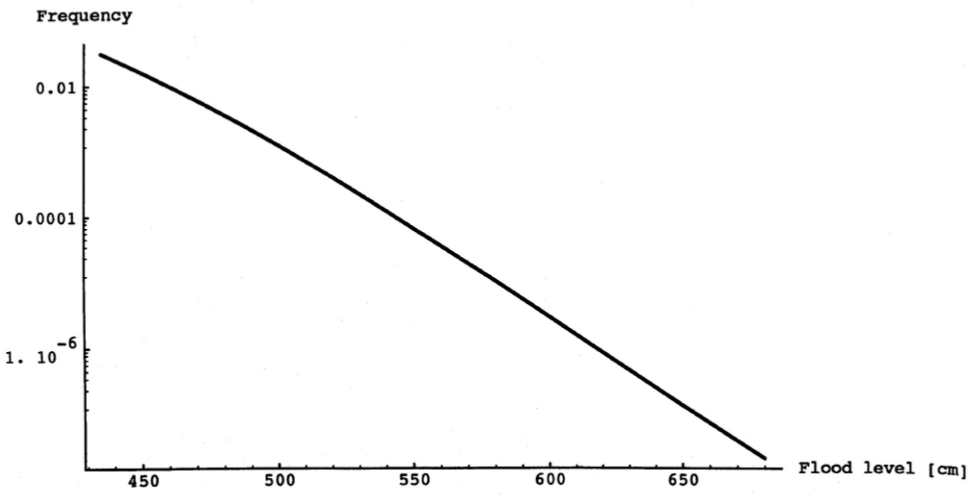
Rupture location on seaward side

A rupture on the seaward side or landward side of the dyke will not always lead to flooding of the site, but can severely damage the dyke. Next to a loss of ultimate heat sink, the plant will be more vulnerable to high tides and storm surges during the time needed to repair the dyke.

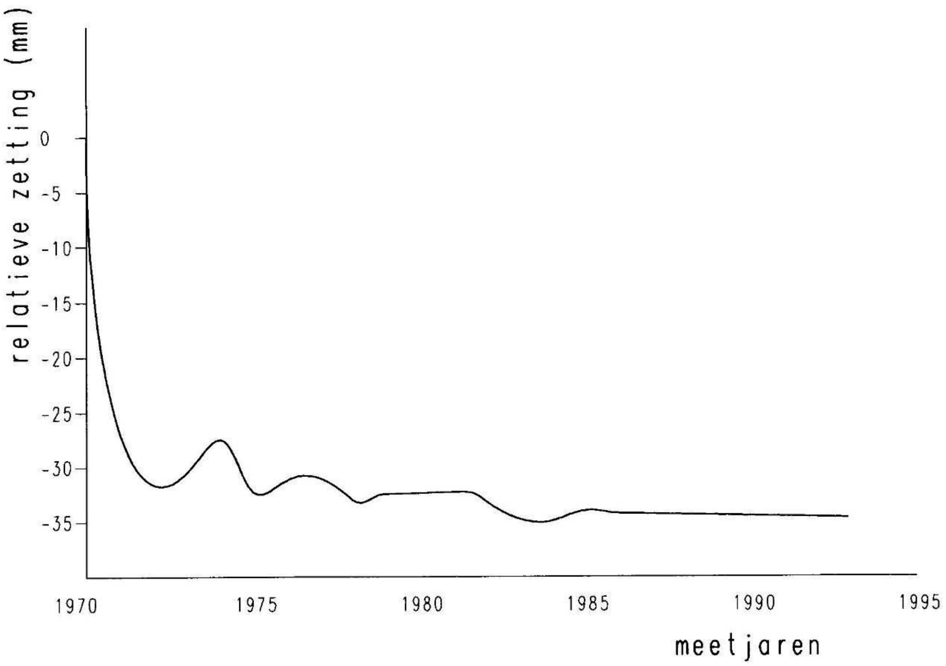
Annex 3.1. Typical water heights

Water height	High water (m + NAP)	Low water (m – NAP)
Average tide Average neap tide Average spring tide	2.02 1.53 2.39	1.80 1.49 2.03
1 February 1953 at Borssele	4.70	
Crest height sea dyke NPP Borssele	9.4 m + NAP	

Annex 3.2. Total high tide exceedance frequency



Annex 3.3. Average settlement of buildings 01, 02, 03



Annex 3.4. Composition of the nuclear design level (N.O.P.)

Component	Static N.O.P. (m + NAP)	Dynamic N.O.P. (m + NAP)
- Nuclear Base Level (N.B.P.)	6.18	6.18
- Effects due to showers	0.15	0.15
- Closing the Oosterschelde dam	0.06	0.06
- Compensation for rising of sea		
level and	0.06	0.06
decreasing soil levels	0.04	0.04
- Settlement surcharge	0.00	0.00
- Reduction hinterland	-0.20	0.80
- Wave run up		
Total required design level	**6.29**	**7.29**
Adopted Nuclear Base Level	7.30	7.30

Annex 3.5. Systems operable from the emergency control room

System code	System description	Tasks
TW	Backup coolant makeup system	Leak compensation, volume compensation during shutdown, reduction of primary pressure due to spraying, increasing the boron concentration for shutdown
TJ	Safety injection system & residual heat removal system	Primary side shutdown and long-term decay heat removal (as far as necessary for shutdown with TE)
TG/ TG080	Spent fuel pool cooling system, including its backup system TG080	Cooling of the spent fuel pool, removal of decay heat from TG
TE	Backup residual heat removal system	Part of the backup cooling chain Residual heat removal from TJ
RS	Backup feedwater system	Supply of feedwater to steam generators
RA	Main steam system	Secondary side shutdown, if available
YD	Reactor coolant pump	Securing primary circuit closure, check containment seal, switching off main coolant pumps
YZ	Reactor protection system	Range switching, manual reset reactor shutdown (scram), sealing control primary circuit
div.	Accident instrumentation	Information on safety-related facilities operated from the backup control room
VE	Backup cooling water system	Part of the backup cooling chain for residual heat removal from TJ and TG
UL	Drainage of building 33	
TL	Nuclear ventilation system (inclusive the filter for containment venting)	Monitoring

Chapter 4 Extreme weather conditions

4.1 Design basis

This chapter describes the design basis of NPP Borssele with respect to extreme weather conditions. The weather conditions taken into account are:

- 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 a design basis

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

Water temperature

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. Analyses have shown that sufficient cooling is guaranteed up to a Westerschelde water inlet temperature (daily-average value) of at least 25 °C. A minimum allowable water temperature is not specified as a design basis.

Observations in the period from 1973 to 2010 show that the average water temperature (measured daily at 12.00 h) in Vlissingen is about 11 °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. On this occasion, a water temperature of more than 22 °C was measured for 12 consecutive days. The longest period of 18 days was recorded in july 1994. On both occasions, normal plant operation was not jeopardized. In general, change of water temperature is a gradual process that will be anticipated adequately.

Air temperature

The minimum or maximum allowable outside air temperature is not specified in the design basis. The average air temperature inside the operational area of building 01 is not allowed to exceed 35 °C in order to limit the pressure increase in building 01 in case of a LOCA. The temperature around the reactor vessel should stay below 60 °C because of aging of the concrete. A temporary increase will have no measurable effect on the accelerated aging and does therefore not directly impact safety. Due to conventional health safety requirements (ARBO) and the allowable heat load of equipment, the temperature in the main control room is not allowed to exceed 25 °C. Sufficient cooling capacity is available to maintain these temperatures under normal conditions. The air inside the buildings is related to the outside air temperature and to the Westerschelde's water temperature. However, changes in inside air temperature are significantly dampened by the gradual warming or cooling of the buildings. In addition, cooling systems will still be available at high ambient temperatures but will be less effective, even in case of prolonged periods of high air temperatures. In case of recorded heat waves[22], the capacity of the cooling systems was sufficient to continue normal plant operation. Annex 4.1 gives an overview of long-term averages at Vlissingen, including measured air temperatures.

At extremely low outside air temperatures, different effects must be avoided, including:

- decrease in the quality of the diesel fuel inventory and battery capacity,
- freezing of coolant for the diesel generators,
- freezing of the fire extinguishing water inventory.

To prevent degradation of the diesel fuel inventory, the diesel in tank EY100 on the roof of building 33, which supplies diesel to the generators EY050 and EY050 of Emergency Grid 2 during beyond design events, is capable of withstanding temperatures of −18 °C. This is assumed to be reasonable since the outside temperature reaches below −10 °C only one day a year on average. EY040 and EY050 themselves are cooled by the Backup feed water system (RS), and can alternatively be cooled by the Backup cooling water system (VE). These systems are not affected by low air temperatures.

Diesel generators EY010 and EY020 of Emergency Grid 1 are cooled by the outside air using a coolant. To prevent the coolant freezing, additives are used to increase the frost resistance down to approximately −18 °C. Diesel generator EY030, which is the backup for EY010 or EY020, has its own cooling system, that is cooled by the Conventional emergency cooling water system (VF). This EY030 cooling system is equipped with a heater which excludes freezing of the coolant.

Diesel generator EY080, which can be used in beyond design basis events, is also equipped with a heater in the cooling chain which excludes the possibilty of freezing of coolant at low temperatures.

[22] A heat wave is defined as a period of at least 5 consecutive days in which the maximum temperature in De Bilt exceeds 25 °C, provided that on at least 3 days in this period the maximum temperature in De Bilt exceeds 30 °C.

The capacity of the emergency batteries in building 33 that power the reactor protection system is partly determined by the ambient temperature in the building. Dedicated equipment ensures that a temperature between 20 °C and 25 °C is maintained in building 33 regardless of outside temperatures.

On the KCB site, the fire hose connections (part of the Low-pressure fire extinguishing system including the fine water spray system, UJ) which are located in the open, are protected against freezing. UJ pipes 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 can be ruled out. When not in operation, a valve in the fire hose connection, located below ground level, prevents that water rises to a level at which it could freeze. After use, the fire hose connections automatically dehydrate so there is no possibility of any water freezing.

Wind

In order to determine the possible damage to buildings caused by wind the occurring wind speed and the associated probability of occurrence at the site of Borssele is of interest. According to IAEA, a distinction is made between two different wind phenomena:

- severe storms with strong winds;
- tornadoes or hurricanes[23].

With respect to severe storms with strong wind gusts, the probability of occurrence for the location is determined by extrapolating data from the KNMI. This results in a maximum wind speed of 202 km/h at a height of 40 m with a probability of 10^{-6} per year. For comparison, the highest wind gust measured at Vlissingen between 1961 and 2007 is 148 km/h (1990) which did not pose any problems (see Annex 4.2).

With respect to tornadoes, the maximum expected wind speed with a probability of 10^{-6} per year is determined based on the following data from the KNMI. The KNMI has developed a frequency curve for whirlwinds which gives the maximum wind speed as a function of the probability of occurrence. For determining protection against tornadoes, a whirlwind with a probability of occurrence of 10^{-6} per year with a corresponding maximum wind speed of 450 km/h has been chosen.

The characteristics of the design basis tornado are:

- maximum wind speed: 450 km/h;
- rotational velocity: 360 km/h;
- translational velocity: 90 km/h.

The maximum wind speed is the sum of the rotational velocity and translation velocity. Table 4.1 gives an impression of the wind speeds and the corresponding Beaufort numbers.

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[23] A tornado or hurricane is a violent whirlwind with a vertical axis, around which a funnel-shaped cloud is composed of water droplets and dust. In this phenomenon, large wind speeds and rapid pressure changes occur.

Beaufort number	Speed (km/h)	Land conditions
0	0 - 1	Calm. Smoke rises vertically.
1	1 - 5	Smoke drift indicates wind direction and wind vanes are not moving.
2	6 - 11	Wind felt on exposed skin. Leaves rustle and wind vanes begin to move.
3	12 - 19	Leaves and small twigs constantly moving, light flags are extended.
4	20 - 28	Dust and loose paper lifting. Small branches begin to move.
5	29 - 38	Branches of a moderate size move. Small trees in leaf begin to sway.
6	39 - 49	Large branches in motion. Whistling heard in overhead wires. Umbrella use becomes difficult. Empty plastic rubbish bins tip over.
7	50 - 61	Whole trees in motion. Effort is needed to walk against the wind.
8	62 - 74	Some twigs broken from trees. Cars veer on road. Progress on foot is seriously impeded.
9	75 - 88	Some branches break off trees, and some small trees blow over. Construction/temporary signs and barricades blow over.
10	89 - 102	Trees are broken off or uprooted and saplings are bent and deformed. Poorly attached asphalt shingles and shingles in poor condition come away from roofs.
11	103 - 117	Widespread damage to vegetation. Many roofing surfaces are damaged; asphalt tiles that have curled up and/or fractured due to age may break away completely.
12	>117	Very widespread damage to vegetation. Some windows may break; mobile homes and poorly constructed sheds and barns are damaged. Debris may be hurled about.

Table 4.1 Wind speeds in Beaufort numbers and km/h with typical conditions

The design load of buildings 01/02, 33 and 35 (belonging to the protected zone) is higher than the design wind load. The resulting thrust pressure at a wind speed of 450 km/h is below the maximum expected static pressure of 0.1 bar (= 10 kN/m^2) in the event of an explosion. It is demonstrated that the maximum expected wind speed is sufficiently covered by the design explosion pressure wave. More details for each building are presented below.

Building 01/02

Building 01/02 is resistant against an external blast load pressure of 0.30 bar. It is demonstrated that the KCB reactor building's outer shell can withstand a load due to an explosion pressure wave of 0.36 bar with a peak reflected pressure of 0.54 bar. Therefore, building 01/02 is also resistant against the maximum expected wind load of less than 0.1 bar.

Building 03/04/05

Buildings 03, 04 and 05 are resistant against wind speeds of at least 12 Bft.

Building 21

The resistance of Building 21 against extremely high wind speeds is covered by its resistance against static water pressure. Therefore, based on engineering judgment it can be concluded that Building 12 is resistant against a wind speeds of at least 12 Bft.

Building 33

Building 33 is resistant against an external blast load of 0.30 bar with a peak reflected pressure of 0.45 bar and therefore also resistant against the maximum expected wind load of less than 0.1 bar.

Building 35

Based on engineering judgment (similarities between the construction of buildings 33 and 35), building 35 is considered resistant against an external blast load of 0.30 bar with a peak reflected pressure of 0.45 bar and therefore also resistant against the maximum expected wind load of less than 0.1 bar.

Building 72

Building 72 is resistant against a wind speeds of at least 12 Bft. Also the gangway that connects building 72 to building 04 is resistant against speeds of at least 12 Bft.

Electricity pylons

Although not likely, failure of pylons due to extreme weather conditions is credible, as is evident from an event in 2010 where pylons failed due to high winds. Failure of the pylon on the KCB site could lead to loss of offsite power (LOOP, see chapter 5), but will not result in the loss of safety systems required for the safe shutdown of the plant.

Wind missiles and hail

Wind missiles are projectiles propelled by extreme winds. These projectiles are assumed to be significantly smaller than a small airplane. A credible effect caused by projectiles could be loss of offsite power due to damage to the power lines or the switchyard. This type of event is included in the loss of offsite power sequences (see Chapter 5). The resistance of buildings 01/02, 33 and 35 against wind missiles is covered by the resistance against a small aeroplane crash since the design-basis aeroplane crash involves a velocity that exceeds the highest ever measured (and credible) wind speed. Thus, projectiles from extreme winds are concluded to have no significant impact on plant safety.

Hail is defined as precipitation in the form of spherical or irregular pellets of ice larger than 5 mm in diameter. Depending on the size of the pellets, some damage to objects on the ground can be expected. However, the effect of hail is negligible compared to the possible effects due to the design-basis aeroplane crash, which the bunkered systems can resist. Therefore it is concluded that hail has no impact on plant safety.

Salt deposition

A side effect that can be caused by dry winds are salt deposits (carried by the air from the Westerschelde) on structures. If salt is deposited on electrical components, this may lead to a loss of offsite power. In this event, the Emergency Grid 1 and Emergency Grid 2 will still be available. As a measure against salt deposits, electrical components can be disconnected from the grid and hosed down. In 2002, this measure proved successful after a storm had caused salt deposits at the KCB site.

Formation of ice on the Westerschelde

In general, the formation of drifting ice is a relatively slow process that can be adequately anticipated. Since the commissioning of KCB (1973), heavy ice formation on the Westerschelde has occured on three occasions, the last in 1997. Formation of ice may start to occur at a water temperature of approximately −1 °C . At this temperature, the intake of cooling water (in building 21) may become blocked.

Ice in the River Westerschelde may cause a blockade of the cooling water intake (building 21), which can result in a reduction of the cooling water flow. In worst case a complete blockade is possible. Although very rare in the Westerschelde, severe drifting ice may also damage the building.

Annex 4.3 contains a cross section view of the cooling water building (21), which houses the five VC pumps (three for NPP Borssele, two for coal-fired power plant CCB) and four VF pumps. It depicts the intake screens of the Cooling water filtering system (VA) and the level of the lowest recorded water level at the intake. Even at the lowest known water level, formation of ice does not instantly block the cooling water intake.

To anticipate a reduction or blockade of the cooling water flow by ice, three levels of safety exist:

1. To prevent icing of the cooling water intake screens (VA), the coarse and fine intake screens in building 21 can be equipped with hot air guns during the winter;
2. If use of the hot air guns cannot prevent a flow reduction of the cooling water intake, VC will be switched off in order to maintain a sufficiently low pressure difference over the (safety-related) VF intakes. By this measure, the total intake flow is reduced to 2%;
3. In case of complete loss of VC and VF due to complete icing, the VE system 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.

Rainfall

The statistics of the KNMI show that for the area of Borssele once every thousand years (10^{-3} per year) 108 mm of rainfall can be expected within a period of 24 hours[24]. For the same conditions, 123 mm of rainfall can be expected within 48 hours.

Extreme rainfall may induce additional force on building roofs. Because of raised edges around the roofs of buildings 33, 35, 03, 05 and 72[(25)], water can accumulate on top of these buildings if all drainage pipes are blocked. This water causes additional load on the civil structure. Building 01/02 has a spherical dome, and building 04 has a saddle shape roof, and the raised edges have a negligible height. Building 21 has no roof edges. Therefore, there is no need to analyse water accumulation due to heavy rainfall for these buildings.

The height of roof edges are presented in Table 4.5. Because all roof edges are higher (with the exception of the gangway between building 04 and 72, which is lower) than the highest expected water level within 48 hours, the maximum roof load resulting from rain is equal for all buildings. The above result shows that 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 is blocked.

Extreme rainfall may also lead to flooding. This topic is covered in Chapter 3.

Snowfall

All the building structures of KCB are designed so that they can withstand all the credible consequences of snowfall. Building standard NEN 6702 specifies a maximum snow load on roof tops of 0.7 kN/m^2; this corresponds to a level of fresh snow[27] of approximately 0.7 m. On average, occasions of more than 20 cm of snowfall occur once every 10 years and more than 35 cm occurs once every 50 years. However, if it continues to snow for an extended period, larger values of snow build up can be expected. Also, if snow thickens due to alternating high and low temperatures, density may increase.

Buildings 33 and 72 are designed for a total of 5 kN/m^2, made up of variable and fixed loads. Building 35 is designed for a total of variable and fixed loads of 2 kN/m^2. For building 21, the snow load is 10 kN/m^2. The allowable maximum variable load for various buildings is specified in table 4.2. Building 01/02 has a spherical roof shape, which prevents a large accumulation of snow. Since this building is designed to withstand external loads it will have ample resistance against the loads caused by snow accumulation.

[24] The largest amounts of rainfall within 24 hours measured in the vicinity of Borssele were approximately 81 mm at Vlissingen and 93 mm at Schoondijke. These measurements cover the period 1951 – 2010.
[25] The gangway that connects building 72 to buidling 04 has a roof edge height of 70 mm and is constructed agains rainfall according to Building standard NEN 6702.

Building (code no.)	Allowable variable load (design) (kN/m^2) (26)	Corresponding depth of snow (m) (27)
01/02	17.4 (28)	17.7
03	10 (29)	10.2
04	1 (30)	1
05	2	2
21	10	10.2
33	2 (31)	2
35	2	2
72[32]	5	5.1

Table 4.2 Maximum allowable variable load on roof tops.

Lightning

The buildings of KCB are equipped with lightning protection, which is connected to the grounding points, in accordance with NEN 1014. In addition, the main buildings (01/02, 03, 04, 05, 06, 33, 35 and 72) are shielded from lightning because the reinforcing steel grid structure in the roofs (see Figure 4.1) and walls is interconnected by welds at regular intervals, creating so-called 'faraday cages' which are connected to the grounding points. The safety systems located in building 33 and 35 are shielded by a faraday grid of 3 x 3 m. The lightning protection of the containment (building 01/02) and building 05 fullfils 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. Faraday cages are also applied around underground cables.

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[26] It is assumed no additional variable loads are already present on the roof.
[27] The density of fresh snow is assumed to be 100 kg/m^3. Compressed snow can have a density of 200 kg/m^3. Wet snow can have a density of up to 500 kg/m^3 .
[28] Based on the resistance against wind load (17.4 kN/m^2) it can be concluded that building 01/02 is resistant to the maximum load that must be anticipated according to NEN 6702.
[29] The roof of building 03 is divided into different zones with different maximum allowable variable loads, ranging from 10 to 20 kN/m^2. As a conservative approach 10 kN/m^2 has been chosen.
[30] The allowed snow load on building 04 is 50 kg/m^2. In addition to this snow load, the variable design load is 100 kg/m^2 (~1 kN/m^2).
[31] Variable design load is 1 kN/m^2. However, for building 33, a total of variable and fixed loads of 5 kN/m^2 is allowed (design spec). The fixed loads on the roof amount to 3 kN/m^2.
[32] The gangway that connects building 72 to buidling 04 is resistant against a snow load according to Building standard NEN 6702.

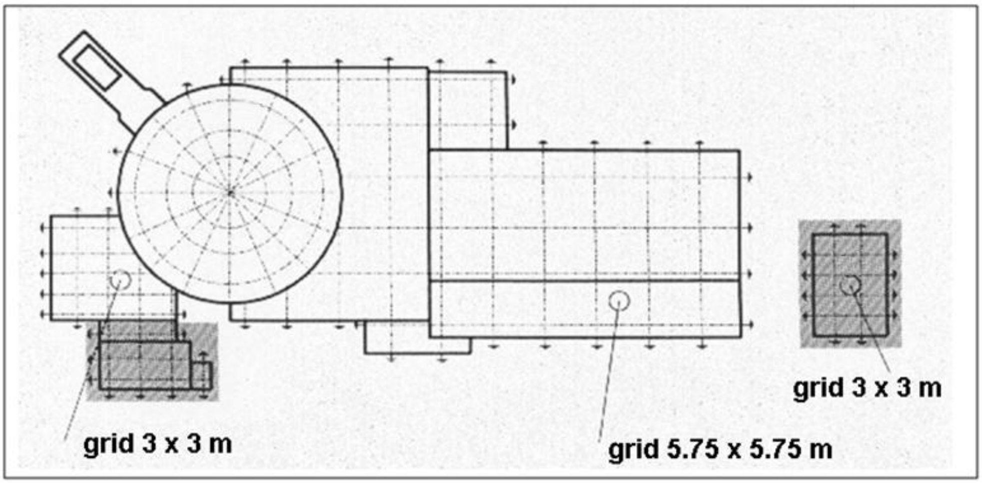


Figure 4.1 Grounding and lightning protection of main buildings

The grids on the roofs meet requirement NEN 1014. Within the electrical installation, additional measures have been taken to prevent a system failure due to potential differences that may occur in case of lightning.

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 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

Besides the weather conditions discussed in 4.1.1.1, 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 first review of the expected frequency of the design basis conditions is treated in 4.1.1.1 for the seperate phenomena. 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. However, most of the phenomena are subject to climate change, which makes a more profound assesment necessary.

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 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 combination of weather conditions

The following credible combinations of extreme weather conditions can be imagined:

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

Ad 1. The combination of high air and high water temperatures is considered a temporary phenomenon because air temperature will decrease at night. The cooling water capacity is not affected by a high air temperature. The allowable air temperature in the containment may be the limiting factor.

Ad 2. The combination of low air and low water temperature is in itself not considered a problem. However, a possible effect at low temperatures is the formation of ice on power lines. The weight of this ice can cause greater tension in the lines, making them break resulting in a loss of offsite power which is discussed in Chapter 5. In addition, black ice may affect logistics. In case of icy roads, a winter maintenance procedure is initiated.

Ad 3. The combination of snow and wind may lead to clogging of air intakes. To eliminate the possiblity of simultaneous clogging, the air intakes of the diesel generators of Emergency Grid 2 in building 33 point in various directions and are aimed downward. Because at least one air intake remains available at all times, no cliff edge effects will occur. High wind combined with heavy snowfall may also cause line galloping and lead to loss of offsite power. In case heavy snowfall threatens to affect logistics, a winter maintenance procedure is initiated.

Ad 4. The combination of high winds, extreme rainfall and lightning can be expected during a thunderstorm. But because the loads caused by these weather conditions are different, they will not reinforce each others effect on the plant.

4.1.1.5 Conclusion on the adequacy of protection against extreme weather conditions

The adequacy of protection for the seperate phenomena is discussed in paragraph 4.1.1.1 and 4.1.1.4. It can be concluded that there are no flaws in the protection, although there is some room for improvement. These points are discussed in the evaluation of the safety margins (see 4.2.1). The recommendations are summarized in 4.2.2.

4.2 Evaluation of safety margins

4.2.1 Estimation of safety margin against extreme weather conditions

This paragraph contains an analysis of the potential impact of different extreme weather conditions 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. In addition, the cliff edge limits that would seriously challenge the reliability of the heat transfer are identified.

Water temperature

The intake temperature of water from the River Westerschelde at which sufficient cooling is fully guaranteed is 25 °C. The system availability at higher water intake temperatures is shown in Figure 4.2.

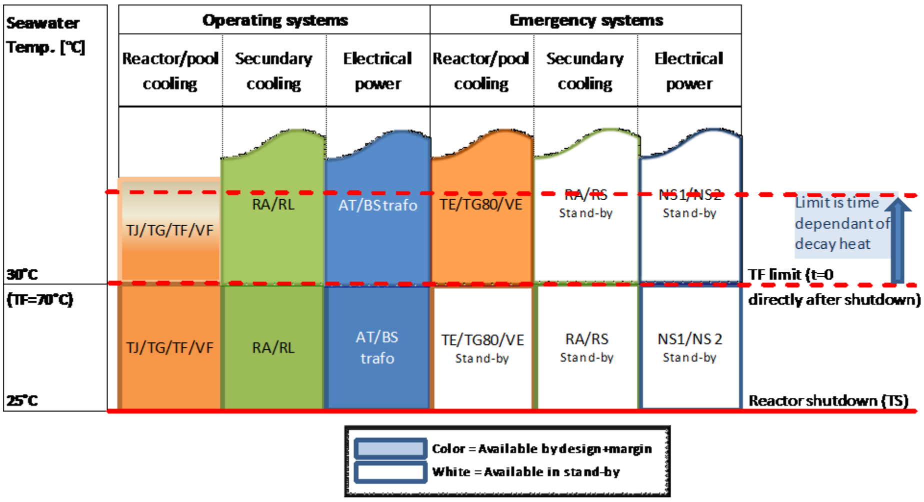


Figure 4.2 System availability as function of the seawater temperature

Figure 4.2 shows that the reactor will be shut down at a daily average seawater temperature of 25 °C due to the requirements in the Technical Specifications. In the exceptional case that this is not done and the seawater temperature rises further, the reactor will be shut down automatically around 27 °C when the reactor coolant pumps shut down because of insufficient cooling to the pump seals. All necessary operating systems are normally available.

After reactor shutdown the seawater temperature will be less limiting because of the decreasing decay heat; furthermore secondary cool down (RA/RS) can be extended. The Component cooling water system (TF) will fail when the TF water temperature exceeds 70°C. As depicted in Figure 4.3, this limits the seawater temperature (VF) directly after reactor shutdown to 30 °C. Because decay heat decreases, the limiting seawater temperature increases in time, since less heat has to be removed from TF to maintain the TF temperature below 70 °C. This means that a further increase in the seawater temperature up to 39 °C during the day after reactor shutdown would be no problem for sufficient cooling of the reactor fuel and the spent fuel pool[33].

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 UHS (which 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, is available as backup.

With regard to normal operating conditions, the temperature inside the biological barrier is kept below 60 °C by the Biological barrier cooling system (TM) in order to prevent accelerated degradation of the concrete. TM is cooled by TF/VF. Therefore, at high water temperatures in the River Westerschelde, the TM cooling capacity may decrease and ultimately shut down could be required. This is, however, only an operational measure and not related to the safety of the plant. The VF system also cools diesel generator EY030. If, at a high seawater temperature, VF is not capable to cool EY030 adequately, still Emergency Grid 1 and 2 are available. Therefore, no cliff edge effects occur at a high VF 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.

[33] 30 °C And 39 °C are calculated values.

Wind

The safety margin for wind resistance of buildings 01/02, 33 and 35 is summed up in Table 4.3 and illustrated in Figure 4.3. Note that the range from 0.1 to 0.3 bar is not to scale.

Building (code nr.)	Required value (bar)	Design value (bar)	Margin (bar)
01/02	0.10 [34]	0.36	0.26
33	0.10	0.30	0.20
35	0.10	0.30	0.20

Table 4.3 Safety margins for wind resistance

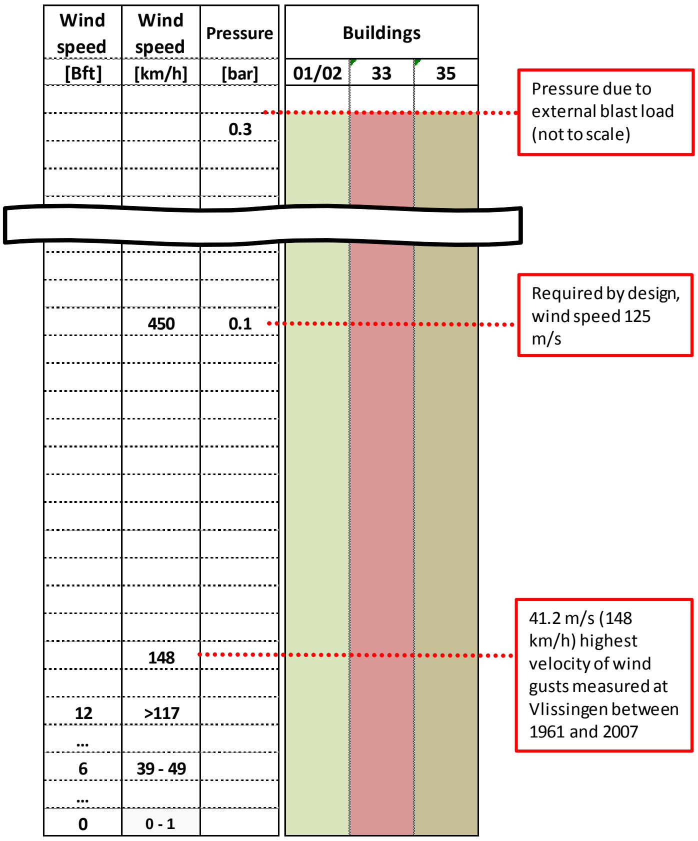


Figure 4.3 Safety margins for wind resistance

For building 03, 04, 05, 21 and 72 no margin is quantified. Extremely high wind speeds may cause fragments of wall cladding to detach in very rare cases, but no cliff edge effects will take place.

[34] The resulting pressure at a wind speed of 450 km/h, see 4.1.1.1.

Formation of ice on the Westerschelde

No quantifiable margin for the ice formation can be given. Formation of ice is no direct problem because normally the VF cooling water will stay available. In case ice does block or damage the cooling water intake, cooling is possible using the VE system, as described in relation with the Loss of Ultimate Heat Sink (LUHS) (see Chapter 5). In the most extreme case, formation of ice on the Westerschelde may lead to unavailability of the cooling water intake, but no cliff edge effects will occur as a result because of the remaining availibility of the VE system.

Rainfall

If after 48 hours rain continues to fall and drainage remains blocked, the water level on top of the building 03 may rise up to a level that exceeds the allowable design load. At this level collapse of the roof is possible. Loss of building 03 will not result in loss of safety systems required for safe shutdown of the plant. Therefore, no cliff edge effects will take place. Collapse is however unlikely given:

- the statistics on expected rainfall as presented by the KNMI;
- the presence of multiple drain pipes.

Operators in and around the plant perform status checks on a daily basis to monitor plant safety. In addition, in case of extreme weather conditions, the civil department inspects building integrity. Blocked drainages will be cleared if necessary.

Snowfall

The safety margin for snowfall resistance of buildings 01/02, 03, 04, 05, 21, 33, 35 and 72 is depicted in Table 4.4 in relation to the maximum load as required by NEN 6702 (0.7 kN/m^2). It should be noted that building 04 has the least resistance to snow and must be monitored first.

Building (code no.)	Height of roof edge (m)	Allowable variable load (design) (kN/m^2)	Margin in relation to NEN 6702 (kN/m^2)
01/02	n/a	17.4	16.7
03 [35]	1.20	10	9.3
04	negligible	1	0.3
05	0.45	2	1.3
21	n/a	10	9.3
33 [36]	0.25	2	1.3
35	0.65	2	1.3
72	0.50	5	4.3

Table 4.4 Roof edge height and safety margins for snowfall resistance

Operators perform status checks in and around the plant on a daily basis to monitor plant safety. In case of conditions involving snow and ice, the civil department inspects building integrity on a daily basis and removes snow if necessary.

When these actions are insufficient, the roof of building 04 will collapse first and will come down on the top floor. This is floor is strong enough to carry this extra load. The major consequence of the collapse can be that the closed loop of secondary cooling via the condensor becomes unavailable. The availability of the Main and auxiliary feedwater system (RL-Main and RL-Emergency) becomes uncertain as the feed water tank is located on the top floor of building 04 and may be damaged by the collapsing roof. Also, the feed water piping is situated against the wall between building 03 and 04 and may be damaged by the collapsing roof. If not, RL-E can be used in combination with the secondary blow down valves (of the Main steam system, RA) to remove decay heat. Water can then be supplied by the Demin water supply system (RZ) or the Backup feed water system (RS). When the RL system is damaged in such a way that this is not possible, the RS system can be used to bring the reactor in a cold undercritical state.

Some parts of the Conventional emergency cooling water system (VF) are located on the bottom floor of building 04. Because the top floor is strong enough, collapsing of the roof of building 04 will not lead to loss of VF.

[35] The roof of building 03 is divided into different zones with different maximum allowable variable loads, ranging from 1,000 to 2,000 kg/m^2. As a conservative approach 1,000 kg/m^2 is chosen.
[36] Variable design load is 100 kg/m^2. However, for building 33, a total of variable and fixed loads of 5 kN/m^2 is allowed (design spec). The fixed loads on the roof amount to 3 kN/m^2.

Lightning

No quantifiable margin for lightning can be given.

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

With regard to the potential increase in the robustness of the plant, the following measure can be envisaged:

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

Annex 4.1. KNMI weather data

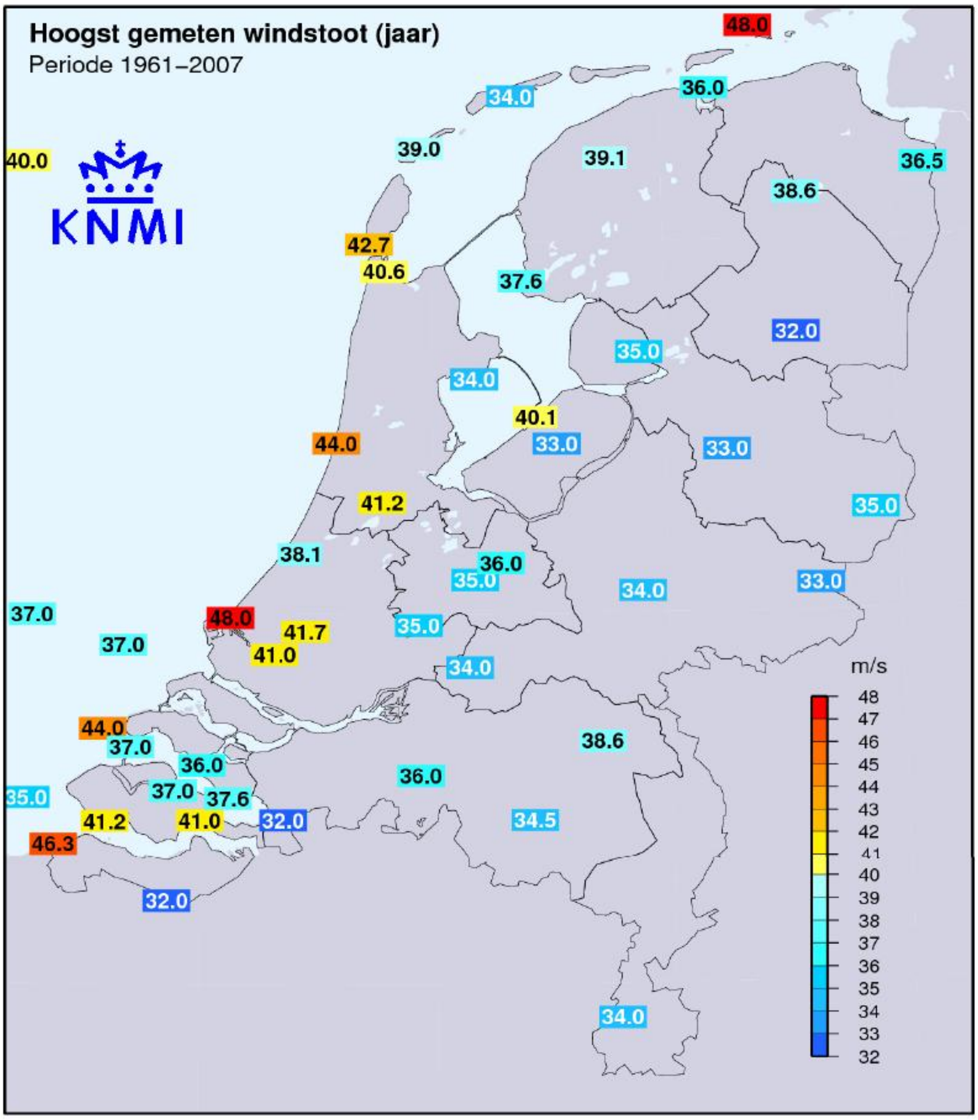
310	Temperatuur(°C)			Relatieve vochtigheid %		Neerslag			Verdamping	Globale Straling	Zonneschijn		Lucht druk	Pot. wind	Gem. wind	Wind-vector		Aantal dagen met windkracht				
	gemiddeld	gemiddeld minimum	gemiddeld maximum	gemiddeld	12.00 UT	duur in uren	in % van de tijd	som in mm	som in mm	som in J/cm2	in uren	in % langst mogelijke duur	in hPa	snelheid in m/s	snelheid in m/s	snelheid in m/s	richting in graden	≥ 4 Bft	≥ 5 Bft	≥ 6 Bft	≥ 7 Bft	≥ 8 Bft
jan	4.0	2.1	6.0	87	86	63.6	9	58.5	9.9	8172	67.8	26	1017.0	6.7	7.5	3.6	218	28	22	14	7	3
feb	4.0	2.0	6.2	85	82	52.7	8	48.0	17.3	14284	89.2	32	1016.9	6.2	6.9	2.1	229	24	18	10	5	1
mrt	6.4	4.1	9.1	83	79	54.6	7	51.1	37.3	28177	134.2	36	1015.4	5.9	6.6	2.5	243	27	20	11	5	2
apr	9.2	6.3	12.5	78	70	39.0	5	38.7	63.5	44083	187.2	45	1014.4	5.2	5.8	0.8	287	26	16	7	2	0
mei	12.9	9.9	16.4	78	70	40.2	5	52.6	88.9	55957	218.3	45	1015.5	5.1	5.6	1.0	295	27	15	6	1	0
jun	15.6	12.6	19.1	78	71	38.9	5	63.2	98.8	58708	215.5	43	1016.5	4.9	5.4	2.0	277	25	15	6	1	0
jul	18.0	15.0	21.5	78	70	31.5	4	64.1	103.9	58860	223.2	45	1016.2	4.9	5.5	2.3	261	26	16	5	1	0
aug	18.2	15.3	21.6	78	70	34.3	5	74.9	88.4	49908	207.4	46	1015.8	4.9	5.4	2.1	256	26	15	6	1	0
sep	15.8	13.2	18.6	80	73	45.3	6	69.4	55.5	32888	152.9	40	1016.1	5.2	5.8	1.9	236	25	16	7	3	0
okt	12.2	9.8	14.6	83	78	53.8	7	76.1	30.7	19812	116.5	35	1014.5	5.9	6.7	2.8	207	27	20	10	4	1
nov	8.1	6.1	10.1	87	84	67.6	9	77.1	13.1	9414	68.7	26	1014.3	6.1	6.9	3.2	211	26	20	11	5	1
dec	4.9	3.0	6.7	88	86	69.6	9	69.0	7.3	5862	52.1	21	1015.6	6.2	7.0	2.9	214	27	20	12	6	2
winter	4.4	2.4	6.3	86	84	186.0	9	176.5	34.5	28303	210.6	27	1016.6	6.4	7.2	2.9	220	79	61	37	19	6
lente	9.5	6.8	12.7	80	73	133.8	6	142.4	189.7	128217	539.7	43	1015.1	5.4	6.0	1.3	263	80	51	24	8	2
zomer	17.3	14.3	20.7	78	70	104.6	5	202.2	291.1	167477	646.0	45	1016.2	4.9	5.5	2.1	264	77	45	17	4	0
herfst	12.0	9.7	14.4	83	78	166.7	8	222.7	99.3	62113	338.2	35	1015.0	5.7	6.5	2.6	215	78	55	28	12	2
jaar	10.8	8.3	13.5	82	77	591.0	7	742.8	614.6	386126	1733.1	39	1015.7	5.6	6.3	2.0	235	314	211	105	43	11

310	Aantal dagen met: Temperatuur maximum					Temperatuur minimum			10 cm	Weersverschijnselen						Neerslag					Zonneschijn		
	≥ 30 °C	≥ 25 °C	≥ 20 °C	≥ 15 °C	< 0 °C	< 0 °C	< -5 °C	< -10 °C	< 0 °C	mist *	regen	sneeuw *	hagel *	onweer *	ijsvorming *	droog	> 0 mm	≥ 0,1 mm	≥ 1 mm	≥ 10 mm	zonloos	≤ 20 %	≥ 80 %
jan	.	.	.	.	2	7	2	0	9	6	21	5	2	0	1	9	22	18	12	1	10	17	3
feb	.	.	.	0	1	7	1	0	9	6	16	5	2	0	1	10	18	14	9	1	6	13	3
mrt	.	.	.	1	0	3	0	.	4	5	19	3	2	1	0	11	20	16	11	1	5	12	4
apr	.	0	1	7	.	0	.	.	1	2	18	2	2	1	.	12	18	13	9	1	2	8	5
mei	.	1	7	18	.	.	.	.	0	2	18	0	1	4	.	14	17	14	10	1	2	9	6
jun	0	2	10	27	.	.	.	.	.	1	18	.	1	4	.	12	18	13	10	2	1	8	4
jul	1	6	18	31	.	.	.	.	.	1	18	.	0	4	.	13	18	13	9	2	1	7	4
aug	1	5	20	31	.	.	.	.	.	1	17	.	0	4	.	14	17	13	10	2	1	7	4
sep	.	1	8	28	.	.	.	.	.	3	18	.	0	3	.	12	18	15	11	2	2	10	3
okt	.	.	1	14	.	0	.	.	0	3	20	0	1	2	.	11	20	16	12	2	4	12	3
nov	.	.	.	1	0	1	.	.	2	5	22	2	2	1	0	8	22	18	13	2	8	17	1
dec	.	.	.	.	1	6	0	.	7	6	21	3	3	1	1	9	22	17	13	2	13	20	2
winter	.	.	.	0	5	20	3	0	24	18	58	13	6	1	3	28	62	49	34	4	29	51	8
lente	.	1	8	26	0	3	0	.	5	9	55	5	5	5	0	37	55	42	29	3	8	28	15
zomer	1	14	48	89	.	.	.	.	.	4	53	.	1	12	.	39	53	39	29	6	3	22	12
herfst	.	1	9	43	0	1	.	.	2	10	60	2	4	6	0	30	61	49	36	6	15	39	8
jaar	1	16	65	158	5	25	3	0	31	42	226	19	16	24	3	134	231	180	128	19	55	139	42

* Fog, snow, hail, thunderstorms and icing are the long-term averages for 1971-2000

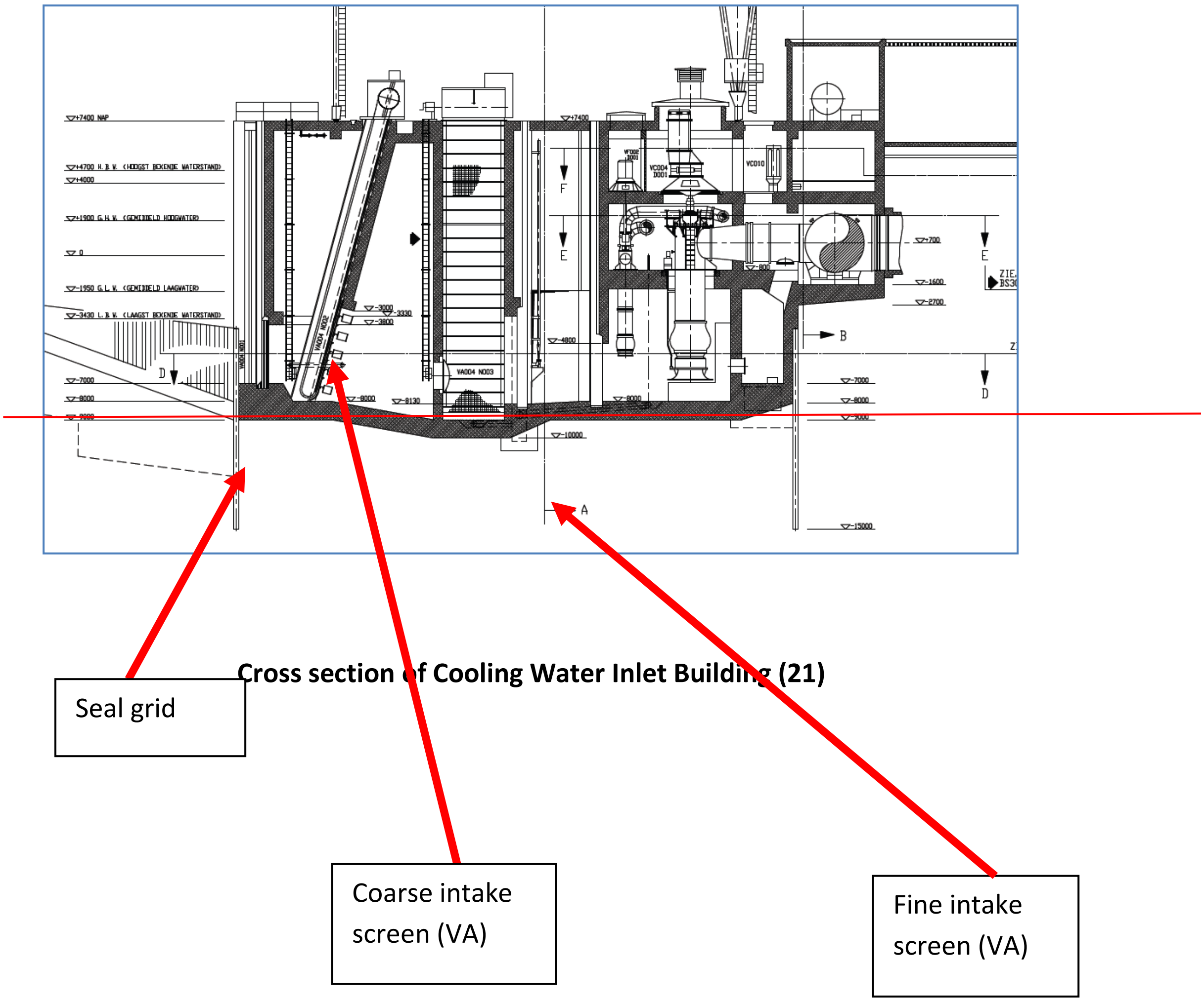
Data for Vlissingen, long-term averages, period 1981-2010

Annex 4.2. KNMI measured extreme wind gusts



Highest wind gusts measured between 1961 and 2007

Annex 4.3. Cooling Water Inlet Building (21)



Cross section of Cooling Water Inlet Building (21)

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

5.1 Nuclear power reactors

5.1.1 Loss of electrical power

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

1. loss of off-site power;
2. loss of off-site power and station black out (referred to as SBO 1);
3. loss of off-site power and total loss of AC-power (referred to as SBO 2).

The assumptions for the separate plant states are as follows::

1. loss of off-site power (LOOP)

   This state is characterized by the unavailability of:

   - supply from the external grids 150 kV and 10 kV and supply from the 6 kV connection between KCB and CCB;

loss of off-site power and station black out (referred to as SBO 1)

   This state is characterized by the unavailability of:

   - supply from the external (direct connected) grids 150 kV and 10 kV and supply from the 6 kV connection between KCB and CCB, plus
   - the emergency grid 1 (NS 1);

loss of off-site power and total loss of AC-power (referred to as SBO 2)

   This state is characterized by the unavailability of:

   - supply from the external (directly connected) grids 150 kV and 10 kV and supply from the 6 kV connection between KCB and CCB, plus
   - emergency grid 1 (NS1), plus
   - emergency grid 2 (NS 2).

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

5.1.1.1 Loss of off-site power

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

Design provisions to prevent LOOP

The electrical energy at a voltage-level of 21 kV generated by the turbine generator is transformed to a voltage level of 150 kV and transported to the 150 kV grid using the generator transformer AT000. The electrical energy required for feeding the systems of the plant is supplied by the house load transformer BT000. This 6 kV house grid exists of two redundant parts, which are operated in parallel and independent of each other. During start-up and shut-down the energy to this grid is supplied by the auxiliary transformers BS001 and BS002 .

The same grid can be fed by CCB via two separate lines connected to the bus bars BA and BB.

In addition, the supply to the back-up feed water system is provided by the 10 kV lines, which in turn are supplied by the 150 kV grid.

LOOP is defined as a loss of supply of electrical power to KCB by:

- loss of grid supply, namely:
    - 150 kV (public line) and,
    - 10 kV (public line) and,
    - 6 kV (CCB – KCB line);
- loss of KCB connections to the 150 kV grid (BT000, BS001 and BS002)
  In this situation it is assumed that the 150 kV grid is still available for CCB as well as the 6 kV connection between CCB and KCB.

Measures to deal with loss of supply related to its originator (loss of grids or loss of connections):

House-load operation

The plant operates in a self-supporting mode, producing electrical power only for its own needs. This means there is a decrease of load of the turbine generator from 100% to approximately 5%. The reactor, for its part, will be operated at approx. 30% of its nominal power. The reactor will be operated in this mode instead of shut-down, when the grid will be restored within a short time-frame. If there is an extended loss of grid, a decision will be made regarding shutdown for which emergency grid 1 (NS 1) will be available. A successful performance of the automatic transition to house-load operation is 81% ,

Redundancy in grid connection

- 150 kV connection
  During normal operation energy to the house grid is supplied by the 21/6 kV house transformer BT000. In case this transformer fails, the connection to the 150 kV grid is provided by the auxiliary transformers BS001 and BS002, which act as redundancy for BT000;
- 10 kV
  The redundant bus bars CX and CW of the emergency grid 2 (NS 2) are independently connected to the 10 kV grid (fed by the 150 kV);
- 6 kV
  Two external redundant connections from CCB bus bars AL19BG/AL19BH to KCB can feed the house grid viz. bus bar BA and BB of KCB to bus bar BG and BH of CCB. The capacity of this feed is limited to 6 MW;

The uninterrupted power system

The uninterrupted power systems (UPS) provide power to the reactor protection system and those components that, in the event of a complete loss of power are necessary to bring the reactor to a safe shut down state.

Internal back-up provisions

The Borssele NPP design currently includes two systems, the emergency grids 1 and 2 (NS 1, NS 2), to deal with LOOP. In addition to this, when the failure of one or both systems deteriorate the plant state, interconnection of both systems or the electrical power supply from the adjacent coal fired plant (CCB) is possible. Ultimately, the uninterrupted power system, is available.

As a last line of defence, one of the two diesel generators of NS 2 can be backed up by the mobile diesel generator EY080. This diesel generator is already on site but (by definition) is restricted available as EPZ needs external support to transport EY080.

The power systems and interconnection possibilities are briefly presented below:

AC power:

- emergency grid 1 (NS 1)

NS 1, basically includes the redundant bus bars BU and BV. These bus bars provide power to the redundant safety-related systems necessary to safely shut down the reactor in case of LOOP . Each redundant train has its own emergency diesel generator (EY010 and EY020). After losing power (voltage level, frequency) on bus bars BU/BV, emergency diesel generators EY010, EY020 and EY030 (3 x 100 %) will start after two seconds. After that, it takes ten seconds to power-up. EY030, which is the standby emergency diesel generator, takes over in case one of the other two emergency diesel generators fails. The diesel generators EY010 and EY020 are cooled by ambient air, while the heat produced by EY030 is transferred to the conventional emergency cooling water system (VF). The diesel generators can also be operated manually from the main control room as well as locally;

- emergency grid 2 (NS 2)

For the plant state where NS 1 also fails following LOOP, relevant safety-related systems to provide a safe reactor state are electrically supplied by NS 2 which consists of the redundant bus bars CW/CX that are normally fed by the external 10 kV connection. When the 10 kV connection is out of service, the diesel generators EY040 and EY050 will start after two seconds. Each redundancy has its dedicated emergency diesel generator. Emergency grid 2 is situated in the backup systems bunkered building, protected against external events like flooding, earthquake and explosion;

- interconnection of NS 1 and NS 2

In the case of LOOP and when NS 2 (additional) is not fed by its own diesel generators, it is possible to connect bus bars BU/BV of the 6 kV system with bus bars CW/CX of NS 2 by using switches and transformers (CT015/016) to supply electrical energy to the users of NS 2 . To implement this, the connections between NS 1 and NS 2 are reset since they are (automatically) locked open in this situation. Procedures are not available for this operation;

- supply by the coal fired power plant (CCB)

CCB is considered to be an on-site, and thus an internal, provider. This provides the following options:

  - 6 kV connection

It is possible that only KCB suffers a loss of off-site power while CCB still has electrical power. In that case the 6 kV bus bars BA/BB at KCB can be connected with the 6 kV bus bars BG/BH at CCB (2 x 6 MW) ;

  - CCB's emergency diesel generators

The two 6 kV emergency diesel generators at CCB can both deliver up to 1 MW to bus bars BA/BB via a fixed connection. This is enough power to feed NS 2 (via coupling NS 1 to NS 2);

- mobile diesel generator EY080

In case of a complete loss of NS 1 and NS 2, the on-site mobile diesel generator EY080 is available. For reasons of physical separation, it is stored at a distance; therefore it has to be transported to backup systems bunker and connected to NS 2. This 1 MW mobile diesel generator has sufficient capacity to supply electrical power to NS 2. Although EY080 is available on-site, EPZ needs external support for its transportation, therefore this option is not considered as a real internal back up option.

DC power

- uninterrupted power supply systems (UPS)

Although the DC power supply is part of the normal operating systems, the batteries are to be considered as the penultimate line of defence. When AC power is available, DC bus bars will be fed by rectifiers (AC/DC). When AC power is lost, the batteries will take over that DC power supply without interruption. These DC-systems provide DC power to the reactor protection system, the instrumentation and those components that, in the event of complete loss of AC power, are needed to bring the reactor to a safe state. The uninterrupted AC busses are powered by the DC/AC convertors fed from the DC busses.

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

The autonomy period is defined to be the time period that systems can provide their back-up function without on-site intervention or internal support e.g. switch over to other systems to supply water or diesel in case the system under consideration runs out of stock. In this chapter only the autonomy period for electrical supply (AC and DC) is considered.

AC power

The autonomy period for the power systems (including connection possibilities) to provide electrical AC power in case of LOOP is presented here:

- emergency grid 1

The emergency diesel generators 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.

According to the Technical Specifications (TS) , both EY010 and EY020 have a fuel supply for 24 hours according to the design requirements. In reallity they have a stock of approx. 95 m^3 and consume 1.2 m^3/h at full load per diesel generator. Therefore they can function for 79 hours). KCB's Technical Information Package (TIP) states that they function for at least 72 hours (design). Diesel generator EY030, which operates as the back-up diesel generator, has a fuel supply of approx. 30 m^3 and also consumes 1.2 m^3/h at full load. Therefore EY030 can function for 25 hours, which is one hour more than the design requirement (24 hours). The control systems of the diesel generators are equipped with battery back-up power sources. In the case this batteries are discharged, the diesel generators can be started manually by opening the valve for starting air;

- emergency grid 2

The diesel generators 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 .

According to the TS , emergency diesel generators EY040 and EY050 both have a diesel tank of approx. 4 m^3 and an extra diesel tank of approx. 9 m^3 is available for both emergency diesel generators. This is enough to operate during the design requirement of 72 hours with expected loads . It has been demonstrated that for both EY040 and EY050, 4 m^3 diesel for each is enough to function for 24 hours with maximum expected loads. It has also been demonstrated that, when the fuel tanks of EY040 and EY050 are both filled with 4 m^3 diesel, an extra fuel tank of 6 m^3 is sufficient to function for 72 hours at maximum expected loads ;

- interconnection of NS 1 and NS 2

It takes around one hour to connect emergency grid 1 with emergency grid 2 to supply electrical energy to the users of emergency power system 2. This time is determined by the switching times and the intervention procedure (on account of reseting thelocked open connections);

- 10 kV connection

In the event that only the 150 kV connections to KCB fail, there will be no limitation in time to supply KCB because 10 kV is in normal operation;

- supply by the coal fired power plant (CCB)

CCB is considered to be an on-site, and thus an internal, provider. This provides the following options:

  - 6 kV connection

In the event that only the 150 kV connections at KCB fail, with regards to electrical power supply, this situation is identical to the 10 kV situation above :

  - emergency diesel generators at CCB

In this situation, the CCB faces LOOP and will therefore be fed by its own emergency power system (two diesel generators). To protect its own equipment during the first 24 hours CCB will need 1 MW. This means that the other diesel generator is available to KCB. After 24 hours, the power supply can be increased up to 1.5 MW. Connecting the first diesel generator to the KCB internal grid will take approx. 4 h.

In case of emergency demands, the supply to CCB can be terminated. It takes approx. 30 min. to start the supply of 2 MW[37].

With a diesel supply (stock) of 4 m^3, and a consumption of 0.43 m^3/h at full load, the run-time of the CCB diesel generators becomes approx. nine hours.

The fuel tank of the mobile diesel generator EY080 contains 3 m^3 diesel: consumption at full load equals 0.3 m^3/h. This allows a power supply for about ten hours.

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[37] Connecting the second CCB diesel generator takes a shorther time then connecting the first one because all the remaining CCB loads in that case can be disconnected in a short time period

This generator is available on-site, but has to be transported to backup systems bunker to be connected to NS 2. The transporting time depends on the availability of a truck, which has to be provided by an external company. It is assumed that this will take some hours.

After EY080 is in position it will take approx. four hours to establish the connection to NS 2

Fuel stocks

The minimum amount of available diesel fuel in stocks is 245 m^3. It will depend on the current situation to get all this stock at the right place during an event.

Nevertheless, depending on the availability and taking fuel consumption related to produced power, the running time of one single diesel generator can be extended to a total of 280 hours (EY010, EY020) or even 1300 hours (EY040, EY050) before all stocks are exhausted.

Within the above-mentioned run times diesel refueling from internal stocks is necessary.

Table 5.1 shows an overview of fuel stocks, running times of the diesel generators and the time necessary to switch from one system to another. NS 1 is available under LOOP conditions. NS 2 and the mobile emergency generator or CCB diesel generator are during SBO 1.No emergency generators are available during SBO. Note that the runtimes in Table 5.1 for the CCB diesel generator and the mobile diesel generator are at full load; for EY010/020/030/040/050 at expected load.

System NS 1	Stock (m^3)	Running time (h)	**Switch/ connecting time (h)**
EY010	95	79	
EY020	95	79	
EY030	30	25	
System NS 2			
EY040	4	22.5	
EY050	4	22.5	
With extra tank	8.8	72	
NS 1 → NS 2			**1**
Mobile diesel generator			
EY080	3	10	
EY080 → NS 2			**6**[38]
CCB diesel generator			
	4	9 (1 x 1 MW)	
CCB → NS 2			**4**

Table 5.1 Stocks and runtimes of emergency diesel generators

DC power

The autonomy periods for the systems to provide electrical DC power in case of LOOP are presented below. It is indicated that for as long as AC power is available, the DC-power system is in operation. As soon as AC power is lost, the DC-system is powered by the batteries.

The DC power-system batteries of ± 24 V/220 V are available when the main power systems, NS 1 and NS 2, are out of service. The batteries deliver electrical power for at least two hours (design requirement) to all DC consumers which need uninterrupted power . In reality, the discharge times are more than two hours. Table 5.2 lists the real discharge times of the batteries of the different systems.

Battery	Discharge time (h)
220 V (NS 1)	2.8
+ 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

Table 5.2 Battery discharge times

The discharge time of the 220 V batteries can be increased to four hours by connecting the EC-bus to the DC/AC-convertors. When the emergency oil pump of the turbine-generator is switched off, the discharge time can increase to 5.7 hours.

The uninterrupted AC-busses are energised by DC/AC-convertors, which are fed by the 220 V DC-busses.

[38] Inclusive transportation time of some hours

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

SBO 1 is defined as loss of off-site power and station blackout. This means that both off-site power and the emergency grid 1 (NS1) are not available. A list of systems needed to provide cooling is shown in Table 5.3; this is explained in more detail in section 5.1.2 and 5.1.3.

System	System code	Connected to
Pressure relief valves	RA	NS 2/EY080 UPS
Emergency feedwater pumps (3)	RL	NS 1 (2) and steam turbine pump (1)
Demineralised water supply system	RZ	NS 1
Demineralised water preparation tank	UA	regular grid
Backuo feed water system	RS	NS 2/EY080
Low-pressure fire-extinguishing system	UJ	Emergency power CCB and diesel pump
Backup residual heat removal system	TE	NS 2/EY080
Component cooling water system	TF	NS 1
Spent fuel pool cooling system	TG	NS 1 (1 pump) NS 2/EY080 (2 pumps)
Safety injection system residual heat removal system	TJ	NS 1
Backup coolant makeup system	TW	NS 2/EY080
Backup cooling water system	VE	NS 2/EY080
Conventional emergency cooling water system	VF	NS 1
Primary pressure system valves	YP	NS 2/EY080 UPS

Table 5.3 Connections of relevant systems to the emergency power systems NS 1 and NS 2

5.1.1.2.1 Design provisions taking into account for 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

Design provisions

In case of a loss of off-site power (LOOP), the emergency grid 1 (NS 1) is designed to provide electrical power to the safety-related systems which are necessary to safely shut down the reactor.

Provisions to prevent failure of NS 1 are:

- two physical separated redundant parts of NS 1 with each part having its own diesel generator and a third diesel generator in a separate building as back-up;
- two external redundant connections from CCB to KCB connected with the main power system.

Internal back-up provisions

To deal with the failure of NS 1 the emergency grid 2 (NS 2) is added to the basic design. In addition to this most of the provisions listed in 5.1.1.1.1 can provide backup in the indicated way. In summary:

AC power:

- emergency grid 2 (NS2)

  NS 2 provides electrical power to relevant safety-related systems to provide a safe reactor state;

- in the event that only 150 kV connections to KCB fail there will be no limitation in time to supply KCB because 10 kV is in normal operation;
- supply by the coal fired power plant CCB is accounted for as being on site, options are:

  - 6 kV connection

    In the event that only the 150 kV coupling to KCB fails, an additional supply to NS 1 can be provided;

  - Emergency diesel generators

An additional supply to KCB, which is considered to be sufficient for NS 2;

- mobile diesel generator EY080.

In case of a total station blackout, mobile diesel generator EY080 is available on site. However, EPZ needs external support for transportation; therefore this option is not considered as a real internal backup option. Instructions for connecting the mobile diesel generator are available .

Table 5.1 includes the relevant data for the diesel supply of the diesel generator systems listed above, which are still available in this plant state. A more detailed explanation is presented in section 5.1.1.1.2.

5.1.1.2.2 Battery capacity, duration and possibilities to recharge batteries

DC power:

The autonomy periods for the systems to provide electrical DC power in case of LOOP are presented below where it is indicated that as long as AC power is available the DC-power system is in operation. As soon as AC power is lost, the DC-system is powered by batteries.

The DC power system batteries of ± 24 V / 220 VDC are available when the main power system, the emergency grid 1 (NS 1) and emergency grid 2 (NS 2) are out of service. The batteries deliver electrical power for at least two hours (design requirement) to all DC-consumers which need uninterrupted DC-power. In reality the discharge times are more than two hours. Table 5.2 lists the real discharge times of the batteries of the different systems.

The discharge time of the 220 V batteries can be increased to four hours by connecting the EC-bus to the DC/AC-convertors[39] and when the emergency oil pump of the turbine-generator is switched off, the discharge time can increase to 5.7 hours.

The uninterrupted AC-busses are energised by DC/AC-convertors, which are fed by the 220 V DC-busses.

The batteries can be recharged when AC-power (normal or emergency) is available again. Recharging will take about 8 hours.

[39] This is an automatic action, under the condition that the control rods are all in the reactor core

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

SBO 2 is a total loss of AC-power, which is characterised by the loss of the off-site power (LOOP, NS 1 and NS 2), including the non-availability of the additional supply of the CCB's emergency diesel generators and the mobile diesel generator EY080. Only the batteries and the uninterrupted (AC) power systems are still in operation.

Design provisions to prevent the plant state

For the event SBO 1, i.e. loss of off-site power (LOOP) and loss of emergency grid 1 (NS 1), the emergency grid 2 (NS 2) is added to provide electrical power to relevant safety-related systems to provide a safe reactor state.

Provisions to prevent the plant state SBO 2 are:

- two physical separated redundant parts of the NS 2 with each part having its dedicated diesel generator (EY040 and EY050);
- separate connection of the 10 kV grid to the redundant bus bars CX and CW of NS 2;
- supply by the CCB emergency power system (maximum 2 x 1 MW);
- installation and connection the mobile diesel generator EY080 to NS 2.

Internal back-up provisions

Most of the provisions that are added to the design to deal with the failure of NS 2 and thus prevent this situation can be considered as backup. These are:.

- supply by the CCB emergency power system (maximum 2 x 1 MW) to the bus bars BA/BB and connected to NS2;
- installation and connection the mobile diesel generator EY080 to NS 2.

However, the application of these backups will turn the SBO 2 situation functionally into SBO 1; this situation is dealt with in section 5.1.1.2

Therefore, it can be concluded that for the true SBO 2 situation only the uninterrupted battery power system remains.

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

The autonomy periods for the systems to provide electrical DC power in case of LOOP are presented below where it is indicated that as long as AC power is available the DC-power system is in operation. As soon as AC power is lost, the DC-system is powered by batteries.

The DC power system batteries of ± 24 V / 220 VDC are available when the main power system, the emergency grid 1 (NS 1) and emergency grid 2 (NS 2) are out of service. The batteries deliver electrical power for at least two hours (design requirement) to all DC-consumers which need uninterrupted DC-power . In reality the discharge times are more than two hours. Table 5.2 lists the real discharge times of the batteries of the different systems.

The discharge time of the 220 V batteries can be increased to four hours by connecting the EC-bus to the DC/AC-convertors and when the emergency oil pump of the turbine-generator is switched off, the discharge time can increase to 5.7 hours.

The uninterrupted AC-busses are energised by DC/AC-convertors, which are fed by the 220 V DC-busses.

The batteries can be recharged when AC-power (normal or emergency) is available again. Recharging will take about 8 hours.

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

Loss of off-site power (LOOP)

<u>Internal support measures</u>

Additional internal support measures in the event of loss of off-site power are the implementation of measures to restore power or enssure replenishment of supplies when the plant’s stocks become exhausted.

Restoration of power supply

A procedure is in place, see Annex 5.1 , which includes actions to restore power in case of loss of power BU/BV.

Replenishment of supplies

The on-site transportation of diesel from remaining stocks can be envisaged to extend the operation time of those emergency diesel generators that are needed and running at that time. Possible on-site stocks listed in Table 5.1 are supplies from those emergency diesel generators that are not in operation. However there are no hardware or software (procedures) measures to facilitate these kinds of diesel transfers (e.g. from diesel generator building 72 (NS 1) to the backup systems bunker (NS 2)).

<u>External support measures</u>

External actions to ensure that the electrical power supply can be continued are identical to those presented for “Internal support measures”. It is assumed that, because the event is only LOOP, a successful external supply of diesel results in an unlimited power supply.

Extra diesel can be transported to the power plant according to .

It should also be indicated that EY080, being the last line of defense, can be installed, when needed. Transport of this diesel generator, by an external contractor, will take some hours; connecting to NS 2 is assessed to take a further four hours.

When extra fuel is transported to the plant within those time periods the runtime is unlimited.

Loss of off-site power and station black out (SBO 1)

Internal support measures

Additional internal support in this SBO 1 event comes from implementing measures to restore power or ensure that diesel fuel supplies are replenished when the plant’s stocks become exhausted. Thes are:

Restoration of power supply

Restart NS 1 in the event that simple actions can reset the system. This will up-grade the plant state to LOOP. This situation has already been covered.

A procedure is in place (see Annex 5.1) which includes actions to restore power in case of loss of power BU/BV.

Replenishment of fuel supplies

As indicated in Table 5.1, stocks can be transferred, especially those from NS 1 which are now available due to NS 1 system failure.

However, there are no hardware or software (procedures) measures to facilitate those kinds of diesel transfer currently available on-site.

External support measures

The actions to ensure that the electrical power supply can be continued are identical to those presented under LOOP conditions. Installation of the diesel generator EY080 has already been indicated.

It is assumed that, because the event is only SBO 1, a successful external supply of diesel fuel will result in an unlimited electrical power supply.

There are arrangements to ensure delivery of diesel in such an event .

Total loss of AC-power (SBO 2)

Internal support measures

Additional internal support in this SBO 2 event comes from implementing measures to restore power or at least put the plant into the SBO 1 state and then ensure replenishment of supplies when the plant's stocks become exhausted. These are:

Restoration of power supply

Restart NS 1 or NS 2 in the event that simple actions can reset the system. This will upgrade the plant state to SBO 1 or LOOP; this situation has already been covered.

Replenishment of diesel supplies

As indicated in Table 5.1, stocks can be transferred to the restored diesel generator(s), especially those from NS 1 and NS 2 that are now available, due to NS 1 and NS 2 system failures. This means that at least SBO 1 is restored.

However, no hardware or software (procedures) measures to facilitate those kinds of transfer are currently available on-site.

External support measures

Basically external actions will include:

- installation (transfer) of the mobile diesel generator;
- transport to KCB and installation of an external diesel generator located near Rotterdam within more than 8 hours;
- a procedure to conclude and then order, the external a diesel generator is nearly complete
- external supply of diesel. It is assumed that because the event is only SBO 2 this supply is unlimited.

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.

The shift staff is competent to make the necessary connections as described in 5.1.1.3.2. Staff is quallified for these tasks. The times needed to arrange the connections are at least four hours to connect the 6 kV emergency diesel generator of CCB and six hours (including transport) for the mobile dieselgenerator, and one hour to connect NS 1 to NS 2. However, all these actions are currently not periodically trained according to a training program.

A specific operational mode on which a SBO may have a large impact is mid-loop operation. This is a short period during the outage period in which the water level in the primary system is lowered to about 2/3 of the primary coolant lines. During mid-loop operation, cooling is provided by the ECCS system (TJ), which can operate on the ordinary backup AC power source (NS 1) during a LOOP. In case of loss of ordinary backup AC power sources, cooling will be provided by water supply by the backup primary supply system (TW) initiated by automatic actions based on low water level or high temperature. The primary system can be filled up with this system and secundary cooling is possible through the steam generators. Feedwater is supplied by the backup feedwater system (RS). Long term cooling can be provided by the alternate ultimate heat sink (TE/VE). These systems (TW/TE/VE) are powered by the diverse backup AC power source (NS 2).

In case of loss of all AC power sources (SBO 2) during the short mid-loop period, the primary water will heat up and start boiling within 15 minutes. After about three hours the upper core will be uncovered, leading to core heat up. To prevent this, water supply is possible by the accumulators (TJ) which operate on nitrogen pressure within the tanks. Water supply is also possible from the ECCS water storage tanks (TJ) by gravity. For both water supplies, manual actions are necessary to open several valves. A procedure is available to perform these actions. The accumulators provide sufficient water for cooling and evaporation during 7 hours and the storage tanks for maximum 36 hours (dependent on the pressure in the primary system caused by boiling). To ensure successful cooling in this way, opening of the valves must be done in a relatively short period of time before the conditions in the containment prevent the necessary manual actions.

In order to make sure the above mentioned actions will be successfully performed the following measures are proposed: 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 timely.

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

See paragraph.5.1.2

5.1.1.4 Conclusion on the adequacy of protection against loss of electrical power

KCB is sufficiently protected against a Loss of Offsite Power. Two independent and redundant (3 x 100% and 2 x 100%) emergency power systems, one of which is protected against external events like flooding, earthquake and explosion, are available to challenge a loss of off site power. As a defence in depth measure the emergency power system of CCB and/or a mobile diesel generator and an external diesel generator are available. All these equipment is adequate in providing electric power to safe shut down, cooling (core and spent fuel) and preventing a radiological release.

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

5.1.1.5.1 Potential cliff-edge effects and measures in case of LOOP

Potential cliff-edge effects

For LOOP events the following potential cliff-edges have been identified:

- failure of house load operation KCB

  When transfer to house-load operation is not successful, the plant must be shut down for which NS 1 is necessary;

- failure of one of the emergency power systems:

    - failure of NS 1 will result in the turnover of plant condition LOOP into SBO 1. This plant state is dealt with in section 5.1.1.5.2. An example of failure of NS 1 are voltage transients;
    - failure of NS 2 will result in no immediate problem as long as NS 1 is in operation and the NS 2 batteries powering the reactor protection system are not exhausted. In addition, NS 2 can be connected to NS 1 when reset conditions are met to unlock connections that are automatically locked open in this situation. Also EY080 or the CCB diesel generator can be applied;
    - failure of all supply, namely NS 1, NS 2, CCB emergency diesel generator and EY080 will cause plant state SBO 2. This situation is dealt with in section 5.1.1.5.3;

- running out of diesel supplies

  Running out of diesel supplies means that one or more emergency power systems will fail; identical to the events mentioned before;

- failure of CCB's supply

  Failure of CCB's supply will consolidate the SBO 1 or SBO 2 situation. These situations are dealt with in sections 5.1.1.5.2 and 5.1.1.5.3 respectively;

- EY080 failure

  Failure to transport and/or to connect the mobile diesel generator results in a complete lack of electrical power identical to the plant state of primary loss of ultimate heat sink combined with SBO 2. Tthis situation is dealt with in section 5.1.1.5.3;

- failure of action

  Failure to replenish the internal diesel supplies, since currently there are no hardware or software (procedures) presentresults in a situation where the running time is limited.

As a result, the only cliff-edge remaining is the failure to replenishes the diesel supplies.

Measures which can be envisaged to increase the robustness of the installation

Possible measures to improve the robustness of the plant, so that it better matches the LOOP conditions are:

- 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;
- 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 2 hours, would increase the margin in case of loss of all AC power supplies including the SBO generators; ;
- Devolp a set of Extensive Damage Management guides (EDMG) and implement a training program. Issues to be addressed:
    - connecting CCB/NS1.

5.1.1.5.2 Potential cliff-edge effects and measures in case of LOOP-SBO 1

Potential cliff-edge effects

In an SBO 1 event, the cliff-edges are the same as those for LOOP, except for the KCB house-load operation and for NS1 because the latter system completely fails. The remaining cliff-edges are:

- failure of supply by all the remaining backup emergency power systems namely NS 2, CCB emergency power system and EY080 results in SBO 2;
- running out of diesel supplies results in SBO 2;
- failure of the CCB supply, which consolidates the SBO 1 situation;
- failure of action (failure to replenish the diesel fuel supply).

As before, the remaining cliff-edge is a failure of action.

Measures which can be envisaged to increase the robustness of the installation

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.

5.1.1.5.3 Potential cliff-edge effects and measures in case of LOOP-SBO 2

Potential cliff-edge effects

In the event of SBO 2, the cliff-edges are similar when compared to those for LOOP and SBO 1, except that the NS 1 and NS 2 systems completely fail. The remaining cliff-edges are:

- failure to deliver an external diesel generator

  The failure to transport and connect an (spare) external diesel generator to NS 2 results in a complete lack of electrical power, which is identical to the plant state of primary loss of ultimate heat sink combined with SBO 2.This situation is dealt with in section 5.1.3;

- running out of diesel supply (if the diesel generators are working)

  Running out of diesel supply means that either NS 1 fails, or NS 2 or both. This is identical to the events mentioned above;

- failure of action

  Failure to implement actions to replenish the diesel supply, since there are no hardware or software (procedures);

- uninterrupted power system

  In the event that the batteries are exhausted, relevant components e.g. relief valves of RA and YP, shall be operated manually. The RA valves can be reached and operated manually.

In conclusion, the cliff-edges for the SBO 2 situation are a failure to deliver the external (spare) diesel generator and/or its diesel fuel supply and a failure of the UPS.

Measures which can be envisaged to increase the robustness of the installation

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 exemple an emergency AC generator could increase the robustness in case of loss of all AC power supplies including the SBO generators.

5.1.2 Loss of the ultimate heat sink

The EPZ nuclear power plant defined the following combinations for loss of the UHS and loss of the UHS combined with SBO (plant states):

1. loss of primary ultimate heat sink (LPUHS);
loss of primary and alternate ultimate heat sink (LPAUHS);
loss of primary ultimate heat sink and station black out (referred to as UHS-SBO 1);
loss of primary ultimate heat sink and total loss of AC-power (referred to as UHS-SBO 2).

Each of the four combinations can be characterised as follows:

1. loss of primary ultimate heat sink (LPUHS)

The following are unavailable due to the loss of its principal supply[40]:

- the Main cooling water system (VC)(needed for bypass operation), plus
- the Conventional and emergency cooling water system (VF);

loss of primary and alternate ultimate heat sink (LPAUHS)

The following are unavailable due to loss of its principal supply:

- the Main cooling water system (VC), plus
- the Conventional and emergency cooling water system (VF), plus
- the Backup cooling water system (VE);

loss of primary ultimate heat sink and station black out (referred to as SBO 1)

The following are unavailable due to loss of its principal supply:

- the Main cooling water system (VC), plus
- the Conventional and emergency cooling water system (VF), plus
- off-site power (LOOP), plus
- the first emergency power system (Emergency Grid 1; NS 1);

loss of primary ultimate heat sink and total loss of AC-power (referred to as SBO 2)

[40] “Due to loss of its principal supply” means that systems are not available due to loss of the principal supply (water or electricity), however there are other means of supply available e.g. LPUHS VF can be supplied by other systems like UJ

The following are unavailable due to loss of its principal supply:

- the Main cooling water system (VC), plus
- the Conventional and emergency cooling water system (VF), plus
- off-site power (LOOP), plus
- Emergency Grid 1, plus
- the second emergency power system (Emergency Grid 2; NS 2), plus
- the emergency power system of the coal-fired power plant (CCB), plus
- the mobile diesel generator EY080.

Because LOOP SBO combinations are dealt with separately, electrical feeding by NS 1 and/or NS 2 to the systems under consideration will be dealt with where relevant. For a complete listing of relevant systems and their connection to NS 1 or NS 2.

Figure 5.1 includes the event tree showing the combinations of system (un)availabilities that result in the defined four plant states.

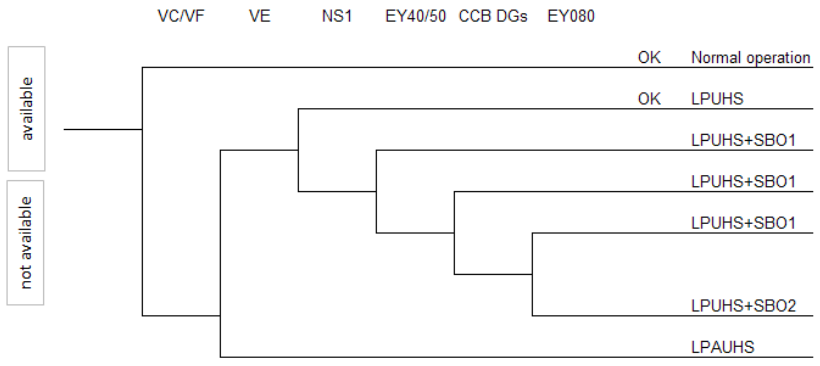

Figure 5.1 Event tree showing the four defined plant states, combining LUHS and SBO situations

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

The primary ultimate heat sink is the water from the River Westerschelde, which is supplied by the main cooling water system VC and the conventional emergency cooling water system VF. The pumps of both systems are located in the cooling water inlet building. Cooling water is supplied through an open channel of which the bottom level is at 5.5 m – NAP, lowered to 8.0 m – NAP in front of the cooling water intake building. By regular dredging it is ensured that the channel has a guaranteed level of 4.7 m – NAP and 7.0 m – NAP at the cooling water intake building, corresponding to the bottom level of the inlet openings .

Further protection of the VC system is delivered by its mussel filters before the condensers and the various filter arrays of the cooling water filtering system VA. Whereas the VC system is not considered safety relevant, the VF system should be functional during normal operation, incidents and accidents. Therefore VC will be switched off-line automatically in case not enough water is available to feed both VC and VF.

The following additional provisions to prevent failure of the VF system are installed:

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

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

Besides the primary ultimate heat sink being the water from the River Westerschelde, which, supplied by the main cooling water system VC and the conventional emergency cooling water system VF, two alternative heat sinks can be identified:

- the atmosphere, in case of steam venting via the main steam relief valves (RA);
- eight deep-water wells (VE).

The relief valves are multiple redundant: two trains with two relief valves each. Feedwater is being supplied by the main and auxiliary feed water system RL or the backup feed water system RS. The latter is twofold redundant and protected against external events.

The deep-water wells are connected to the plant through the Backup cooling water system (VE). Earthquake and flooding are part of its design basis, as well as a n+2 redundancy for the wells and pumps. Of the eight VE pumps available, six are needed for design capacity. The VE pumps are connected to the power supply of NS 2, which is designed for external events. Four pumps are fed from one busbar of NS 2 and four from the other one. The VE system connects to the reactor cooling via the Backup residual heat removal system TE, which has two cooling pumps in parallel and a single heat exchanger. The two pumps are fed from the redundant busses of NS 2. The VE system is connected with the spent fuel pool-cooling via the TG080 heat exchanger. One out of three pumps in parallel provides pool water flow. One of these pumps is connected to the power supply of NS 1, the other two to NS 2, in the redundant mode of NS 2.

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

In case of a loss of the primary ultimate heat sink (LPUHS), the main cooling water system VC and the conventional emergency cooling water system VF will not continue functioning, because of , for example, station blackout. However, various water reserves are available for cooling, some of which are also used also in normal operation, and some which are installed for emergency situations only. See Table 5.4.

System	System code	minimum water volume (m^3)	Remarks	
Feed water tank	RL	185		
Hot well	RM	41		
Demin water supply tank	RZ	268		
Demineralised water preparation tank	UA	814		
Basins of the backup feed water system	RS	900	(= 2 x 450)	
Water tank of the low pressure fire extinguishing system	UJ	1,200	On the premises of the neighbouring coal-fired power station CCB	
Water of the firefighting pond of CCB	UJ	1,600		
Coolant storage and regeneration system	TD	250		
Spent fuel pool cooling system	TG	565	Until the top side of the fuel racks	
Reserve and buffer tanks of the safety injection system & residual heat removal system	TJ	765.2	(= 4 x 170 + 4 x 21.3)	
Borated basins of the backup coolant makeup system	TW	400	(= 2 x 200)	
Backup cooling water system	VE	unlimited	8 deep water wells	
Public water supply system	Delta	unlimited	Normal supply: 50 m^3/h For emergencies: 180 - 200 m^3/h	
River Westerschelde	Fire brigade	unlimited	Arranged by the fire brigade using its firefighting equipment	

Table 5.4 Available water reserves in case of a loss of ultimate heat sink scenario

The various options for cooling of the core are described below.

The design includes systems that can be applied in several (sometimes complementing) combinations to mitigate and control this situation of loss of primary ultimate heat sink. Following these combinations are referred to as options. The most obvious and most logical option from a control point of view which is also part of the emergency procedures, will be presented first. Illustrating the robustness of the plant to cope with the event, alternatives to perform the same cooling function will be listed in order of preference.

It should also be understood that cliff-edges due to failure of one link in the options' cooling chain will be avoided by applying or switching over to one of these alternatives.

For reactor cooling two phases have been identified:

- cooling-down phase and;
- decay heat removal phase.

A switch over is defined by "decay heat removal conditions". Depending on the systems used, these conditions range from 30 bar and 180 °C to 13 bar and 120 °C for combined primary system pressure and temperature, applied to TJ/VF and TE/VE respectively. Heat removal capacity of these systems is such that TJ/VF can start heat removal 3 hours after shut down and TE/VE 13 hours after shut down.

Cooling-down phase

General

When cooling down, the preferred order of options is:

1. the Main and auxiliary feed water system RL, combined with the main steam relief valves (RA);

the Backup feed water system RS, combined with the main steam relief valves (RA);
the secondary bleed & feed procedure;
the primary bleed & feed procedure; this will ultimately be applied when other cooling down options fail and will be initiated by the primary system limit

$$T_{core\ outlet} > 650\ °C.$$

RL and RS may be supplemented by alternative water reserves, starting with the demineralised water supply system RZ, followed by alternatives[41] like the Low pressure fire extinguishing system UJ and ending with the fire brigade. It must be emphasised that of all the options dealing with cooling down by the secondary system other than RL and RS, the operation of the RA relief valves is essential to arrive at (low-pressure) conditions which supply the possible options. Only RL and RS can supply water at high pressure e.g. when the RA relief valves fail.

[41] Small water reserves, like those of the volume control system TA and the chemical control system TB are not taken into account, as well as the on-board water reserves of the fire trucks.

Basically a distinction is made between the preferred or main option and its alternatives. For reasons of transparency all are grouped by the principal system or method providing cooling. This grouping conforms to the listing of options as indicated above, namely:

1. RZ
2. RS
3. secondary bleed & feed
4. primary bleed & feed.

A sub division of these groups is based on supporting or supplying system(s) that provide(s) water supply. Depending on their availability, this supply will be provided by the principal systems' own stock, by a single system and/or by a succession of combined systems. This forms the following sub-division of, for example:

- Group 1: RL:
    a. RZ
    b. RM/RZ/UA and
- Group 2: RS:
    a. RS own stock
    b. RZ
    c. UJ
    d. Fire truck
    e. River Westerschelde

All combinations will be elaborated in the following sections. Figure 5.2 shows the complete grouping and subdivision of the options.

These supporting (supply) systems can start as a separate part of the cooling chain, but can also succeed the sub-option that is listed as preceding one.

Group 1: Cooling by RL

Option 1a: **RL**/RZ (main option)

The main option to provide cooling down is the application of *the main and auxiliary feed water system RL in combination with the main steam relief valves, RA with an, additional supply from the demin water supply system, RZ.*

With regard to the pros and cons of this option, the following applies:

- Water supply

  Pressure and temperature in the secondary system will be lowered by steam venting over the main steam relief station RA. For the purpose of conservatism, the minimum amount of water supply from the RL tank, 185 m^3, is applied . An assessment with regard to the required (cumulative) water amount during the course of the event is presented in Annex 5.2. In this appendix it is shown that the 185 m^3 will be sufficient for cooling down during the first three hours of the event. One of the functions of the demin water supply system RZ is the supplementation of the emergency feedwater pumps in case of a low water level in the feedwater tank. The minimum available water volume in RZ is 268 m^3. It has been shown that the minimum available amount of demin water is sufficient to arrive safely at the decay heat removal phase . From event initiation, this will take a minimum of approx. three hours (cooling down by 100 K/h) , while after less than 13 hours departing from a full power level cooling down with TE/VE can be started .

  It is noted that a supply by RZ direct to the steam generators is possible; a procedure is at hand .

- RL pumpcooling

  The emergency feed water pumps (seals) are normally cooled by the VF system via the conventional component cooling water system VG. The operational water system UK is designed to take over this function automatically in case VF is not available . The UK system retrieves its water supply from its connection with the public water supply system (Delta). If this connection is not available anymore, this option will expire after the UK reserve tank in building 04 has been exhausted (18 hours) [42], except for the situation when UK water is applied elsewhere e.g. for high pressure firefighting. In this case the UK tank can be replennished from the UK tank of the low pressure fire fighting system UJ (on CCB site; 1,200 m^3)

Option 1b: **RL/RM/RZ/UA**

Main and auxiliary feed water system RL supplemented by alternative available water resources. If the RZ reserves are exhausted, the water reserves from the demin water preparation system UA (814 m^3) and the main condensate system RM (41 m^3 capacity hotwell)

[42] For LPUHS it is assumed that replenishment by Delta is provided

could be used as well . The UA reserves are supplied to the RZ system]. As the capacity of UA is more than that of the RZ system itself, there will be sufficient water available to arrive safely at the decay heat removal phase.

Group 2: Cooling by RS

Option 2a: **RS**

Backup feed water system RS.

This system has been designed to take over the feedwater supply to the steam generators, in case the water supply to the steamgenerators is insufficient (SG level low). The available RS water volume is 900 m^3, distributed ove two basins (450 m^2 each).

Option 2b: **RS/RZ**

Backup feed water system RS supplemented by the demin water supply system RZ.

In case of exhaustion of RS, RZ could be used to replenish its basins.

Option 2c: **RS/UJ**

Backup feed water system RS supplemented by the low pressure fire extinguishing system UJ.

RS has a permanent connection to UJ in the backup systems bunker. The capacity of the UJ system (360 m^3/h) is sufficient for decay heat removal after the autarky period of ten hours The autarky period is the time in which no internal actions are necessary (KCB defintion). The UJ system is pressurised by an electrical jockey pump and one electrical and one diesel-powered main pump on the premises of the coal-fired power station CCB, a diversity against possible station black-out . The UJ system is replenished by the public water supply system (Delta).

Option 2d: **RS/fire truck**

Backup feed water system RS supplemented by a fire truck[43],

In case of the RS basins' supply is exhausted, a free-filler opening with fire hose connection has been provided on the RS system, for water to be suppied from a source outside the RS building (i.e. fire truck fed from the fire fighting pond at CCB).

[43] Basically, the fire truck will take a supply of water from the UJ system (hydrants); alternative or additional supplies may be found externally by other sources e.g. the fire fighting pond at CCB, the River Westerschelde by different supply lines or other neighbouring ponds. The following set of possible combinations can be applied:

- UJ is means direct supply from UJ into the referred basic system, possibly replenished by the public water supply system (Delta)
- "fire truck" supposes that only the fire fighting pond supply is used
- Westerschelde means that the fire truck takes its suction by fire hoses from the Westerschelde

Option 2e: **RS/Westerschelde**

Backup feed water system RS supplemented by the River Westerschelde

As with the previous option, the supply will be provided via one or more fire trucks. This time suction is taken from the River Westerschelde using the fire hoses, provided by the fire brigade, constitute the suction lines.

Group 3: Secondary bleed & feed

Option 3a: **secondary b&f/RL/RZ/RM/UA/RS/UJ**

Secondary bleed and feed procedure with the main and auxillary feed water system RL, followed by support system supply.

In this procedure, the heat is removed on the secondary side by venting the steam from one steam generator via the relief system (bleed), while feeding this steam generator (feed) from the feedwater tank, that is being pressurized by the other steam generator . The available water reserves are:

- the RL feedwater tank;
- the basins of the Demin water supply system RZ;
- the hotwell content (RM), the Backup feed water system RS;
- the Demin water preparation system UA ;
- the UJ supply (UK tank).

Due to the depressurisation of both steam generators (especially the driving one) this bleed & feed method is limited in time. Therefore, after this depressurisation, RZ can supply direct water to the steam generator (procedure is at hand) which is additionally fed by UA. With the water volume of RZ and that of RL (the feed water tank), the required time period of 13 hours for the low pressure systems to take over decay heat removal can already be bridged entirely.

Option 3b: **secondary b&f/UJ**

Secondary bleed and feed procedure with the main steam system RA followed by UJ supply via RS.

Before option 3a runs out of water reserves or in case this option is not available, the preferred method is to supply, after depressurisation, additional water through the UJ system, which has access to a 1,200 m^3 on-site water tank . The UJ system is replenished by the public water supply system (Delta). Alternatively, a 1,600 m^3 fire fighting pond located on the premises of the neighbouring coal-fired power station CCB[44] is available. A flexible connection can be established via an additional fire hose connection on the two redundancies of the RS system for direct injection into the steam generators.

[44] For reasons of conservatism, when assessing supply times of systems supplied from the UJ tank is considered to be applied in parallel with supply from the pond. Alternative these two volumes can be considered in succession.

Option 3c: **secondary b&f/ fire truck**

Secondary bleed and feed procedure with the main steam system RA, followed by the fire truck that takes suction from the fire fighting pond at CCB.

In case of the non-availability of internal water supply (basically options 3a and 3b), the afore mentioned additional fire hose connection is available on the connecting duct between the two redundancies of the RS system for direct injection in the steam generators. When secondary feedwater is no longer available, a secondary bleed and feed is first applied. Then after three hours a fire pump can be started .

Option 3d: **secondary b&f/ Westerschelde**

- Secondary bleed and feed procedure, followed by the fire truck that takes suction from the River Westerschelde

In case of no available supply by UJ or the fire fighting pond at CCB, water can ultimately be tapped from the River Westerschelde and transferred to the plant via fire fighting equipment.

Group 4: Primary bleed & feed

Option 4a: **primary b&f/TJ**

Primary bleed and feed procedure with the safety injection system & residual heat removal system TJ.

As some radioactivity release will always take place with primary venting, this option is the least preferred. The procedure will be initiated at a core outlet temperature of > 650 ^{o}C, combined with insufficient secondary heat removal. Primary bleed and feed is initiated by the opening of a pressuriser valve. With the TJ stock (726 m^3), cooling in this way can be performed for approx. 36 hours.

Option 4b: **primary b&f/TJ/TA/TB/TD**

Primary bleed and feed procedure with the safety injection system & residual heat removal system TJ additionally supplied by the coolant storage and regeneration system TD.

In case option 4a runs out of TJ supply when TJ operation is still needed, TD supply can be established through the chemical control system TB and the volume control system TA.

Option 4c: **primary b&f/TW**

Primary bleed and feed procedure with the backup coolant makeup system TW.

This is similar tooption 4a, but with water supplied by the TW system instead of TJ. After about one hour, the two TW pumps are able to remove the decay heat . After about 10 hours, one TW pump is sufficient. The capacity of the TW water supplies (400 m^3) is sufficient to arrive safely at the decay heat removal phase.

Option 4d: **primary b&f/TW/TB/UJ**

Primary bleed and feed procedure with the backup coolant makeup system TW, additionally supplied by the low pressure fire extinguishing system.

In case the TW water stock is exhausted in option 4c and a water supply is still needed, a replenishment of this supply can be performed by UJ through the chemical control system TB. The connection between UJ and TB will be provided by fire hoses through an open door of the backup systems bunker.

Decay heat removal phase

General

Decay heat is normally removed from the RCS by the Safety injection system & residual heat removal system TJ and transferred via the Component cooling water system TF and the Conventional emergency cooling water system VF to its primary ultimate heat sink the Westerschelde. However, VF is assumed not to be available anymore in the loss of UHS scenario, so the options that are provided basically rely on a replacement by the alternative application of the VF system (ducts) or application of the alternate heat sink by the Backup cooling water system (VE).

As with the cooling down phase, one main option and some alternatives can be identified for the decay heat removal phase, which are further grouped and sub divided resulting inthe following listing:

In groups:

1. TJ/TF/VF
2. TE/VE.

In sub division:

   a. UJ
   b. Fire truck
   c. River Westerschelde.

Figure 5.3 shows a complete listing of this grouping and sub division.

All the combinations are elaborated below.

Group 1: TJ/TF/VF

Option 1a: ***TJ/TFVF//UJ (main option)***

The main option to provide decay heat removal is the application of the cooling chain which includes the *safety injection system & residual heat removal system TJ / component cooling water system TF / water supply by the low pressure fire extinguishing system UJ via the conventional emergency cooling water system VF*.

This option[45] is available from RCS pressures that are lower than 30 bars. The cooling water is supplied by UJ to the TF-coolers. The UJ system can be replenished by the public water supply system (Delta).

Water can also be supplied from the neighbouring coal-fired power station CCB, where three locations are connected to the UJ system . This option requires personnel to establish fire hoses to connect UJ with VF. An instruction is available .

In this way the cooling chain TJ/TF/VF, originally designed for this purpose, can be kept operable.

Option 1b: **TJ/TF/VF/fire truck**

Water supply to the conventional emergency cooling water system VF from a fire truck that takes suction from the fire fighting pond of CCB.

VF is equipped with connections for a fire truck. An instruction for this is available . In this way, the cooling chain TJ/TF/VF can be kept operable.

Option 1c: **TJ/TF/VF/Westerschelde**

Water supply to the conventional emergency cooling water system VF from a fire truck that takes suction from the Westerschelde.

In case there is no available of supply via UJ or the fire fighting pond at CCB, water can ultimately be tapped from the River Westerschelde and transferred to the plant via fire fighting equipment.

[45] As decay heat removal conditions are 30 bar and 180 °C with this option this is preferable to the alternate TE/VE options in emergencies.

Group 2: TE/VE

Option 2a: **TE/VE**

Backup residual heat removal system TE / backup cooling water system VE.

This system is designed for the purpose of backup at the loss of primary ultimate heat sink. It can only take over at RCS conditions that are lower than 13 bar and 120 ^{o}C, and it cannot start earlier than 13 hours after reactor shut down, because of its limited heat removal capacity .

Option 2b: **TE/VE/UJ**

Backup residual heat removal system TE / backup cooling water system VE, supplemented by the low pressure fire extinguishing system UJ.

In case the VE system looses its heat sink (the deep water wells or pumps), the low pressure fire extinguishing system UJ can take over . The UJ system is replenished by the public water supply system (Delta).

Option 2c: **TE/VE/fire truck**

Water supply to the backup cooling water system VE from a fire truck that takes suction from the fire fighting pond at CCB.

In the backup systems bunker a fire hose connection is available. In this way the cooling chain TE/VE can stay intact.

Option 2d: **TE/VE/Westerschelde**

Water supply to the backup cooling water system VE from a fire truck that takes suction from the River Westerschelde

In case there is no available supply by UJ or the fire fighting pond at CCB, ultimately water can be tapped from the River Westerschelde, transferred to the plant via fire fighting equipment and supplied to VE through the fire hose connection in the backup systems bunker. In this way the cooling chain TE/VE can stay intact.

5.1.2.2.3 Time available to recover the primary ultimate heat sink or to initiate external actions and to restore core cooling before fuel damage: consideration of various examples of time delay from reactor shutdown to loss of normal reactor cooling condition (e.g., start of water loss from the primary circuit)

The main options and their alternatives to cool the reactor core and the spent fuel pool, described in section5.1.2.2.2, are indicated in Figure 5.2 and Figure 5.3 with the systems used for the respective options. Figure 5.4 presents the situation where UJ and its alternatives are applied for both cooling down and spent fuel pool cooling. Figure 5.5 presents the situation where UJ and its alternatives are applied for both decay heat removal and spent fuel pool cooling.

The introduced distinction of phases, groups and options, as well as the indicated ranking of preferred options and the succession of applied systems is maintained. Stocks (minimum available water stocks in m^3) in the water supply systems for all options are presented in Table 5.5.

System	volume m^3	volume for evaporation m^3
RL	185	
RZ	268	
RM	41	
UA	814	
RS	900	
UJ	1,200	1,200
TJ		640
TN		268
Fire truck	1,600	
Secondary f&b/RL	185	
Primary b&f/TJ	726	
Primary b&f/TW	400	
Primary b&f/TD	250	
TE/VE/UJ	1,200	
TG	730	565
VF/UJ	1,200	
VF/Fire truck	1,600	
TG080/VE/UJ	1,200	
TG080/VE/Fire truck	1,600	
Public water supply	unlimited	
Westerschelde	unlimited	

Table 5.5 Available amount of water per system

Cooling down

In the time period for cooling down it is assumed that heat removal is provided by evaporation of water that is in stock in the systems under consideration and/or water that is additionally supplied by other systems, mainly to the steam generator.

For the cooling down phase, all options meet the 13-hour criterion, posed by the decay heat removal chain TE/VE. This is the most severe criterion for the cooling down options. From Figure 5.2 it can be concluded that the cooling down phase can be extended up to18 hours for the main option (1a) and a maximum of nine days for option 2c based on available on-site water stocks in the systems that are part of the indicated cooling chain, namely. by supplying the UJ stock. Replenishment of UJ by supplies of the public water supply system (Delta) is possible, so water supply through UJ can be extended for of the event' duration.

As regards option 2d, only the CCB pond is accounted for, in parallel to UJ supply; when assuming successive supply by UJ and the CCB pond, the period will extend to more than 11 days. Untill then, a, continuous water supply from the public water supply system (Delta) is possible.

Option 2e also includes an unlimited supply of water by tapping it from the River Westerschelde.

Decay heat removal

With regard to assessing the time period, it is assumed that decay heat removal will basically be performed by water cooling (no evaporation). As cooling water flows for the indicated options do not match the optimal cooling configuration a water spill of 15% is assumed. This means that, for example, from the UJ stock of 1,200 m^3 there will be 1,020 m^3 water used for effective cooling.

For the decay heat removal phase, see Figure 5.3, two points of time are important with regard to the indicated options of three hours, the minimum time to realise cooling down, and 13 hours, the time required to meet decay heat removal conditions for the TE/VE chain (group 2). For these situations this means that decay heat removal by UJ supply (using only on-site stock) will last for six hours and seven hours respectively after shut down. The difference in time for bare decay heat removal by the UJ stock results from the different moments these removals take place and the related heat production.

Replenishment by the public water supply system (Delta) basically provides unlimited cooling. Alternative water supplies can be provided for limited time periods by the fire fighting pond at CCB (option 1b) and unlimited by the River Westerschelde (option 1c), although it will probably be easier to switch to the next (most probable) option, which is TE/VE cooling. This cooling chain is available during the course of the event from the moment that decay heat removal conditions are reached.

Combined cooling

With regard to reactor operation situation, it might be possible that one system will provide cooling of both the reactor and the spent fuel pool at the same time.

Compared to the fuel pool assessment above for this situation the following is assumed, (see **Fout! Verwijzingsbron niet gevonden.**):

- the number of elements stored is 1/3 of a full core;
- storage time to be accounted for is 14 days after shut down (average refuelling period);
- heat removal will be performed:
    - by evaporation for cooling down;
    - by water cooling for decay heat removal;
    - by either water cooling or heating up and evaporation or both of these options for spent fuel pool cooling.

Cooling down + spent fuel pool cooling

With regards to this situation Figure 5.4 is derived from a combination of Figure 5.2 and Figure 5.16 as its scenario indications refer to the scenarios in this latter figures. It is noted from Figure 5.4 that when applying UJ (on-site) supply for both cooling down and spent fuel pool cooling, the water supply will last nine hours. When thesupply is provided by the fire fighting pond at CCB, the duration of water supply will extend for a further two hours. The supply by the public water supply system (Delta) and the River Westerschelde are again unlimited.

Decay heat removal + spent fuel pool cooling

For this situation, Figure 5.5 is derived from a combination of Figure 5.3 and Figure 5.16, as its scenario indications refer to the scenarios in these latter figures. It is noted from Figure 5.5 that when applying UJ (on-site) supply for both decay heat removal and spent fuel pool cooling, the water supply will last approx. 6 hours or 13 hours after shut down depending on when decay heat removal starts after (3 hours and 13 hours after shut down respectively). There is no decay heat removal by UJ while the pool starts heating up. A continuing supply from the public water supply system is considered insufficient because its capacity for cooling water flow is too small. The same accounts for the supply via the fire truck because its capacity is also too small. Therefore, preference should be given to the combinations presented in the second part of Figure 5.5 where decay heat removal by UJ or fire truck and the capacity of the public water supply system and fire truck are adequate for heat removal in these situations.

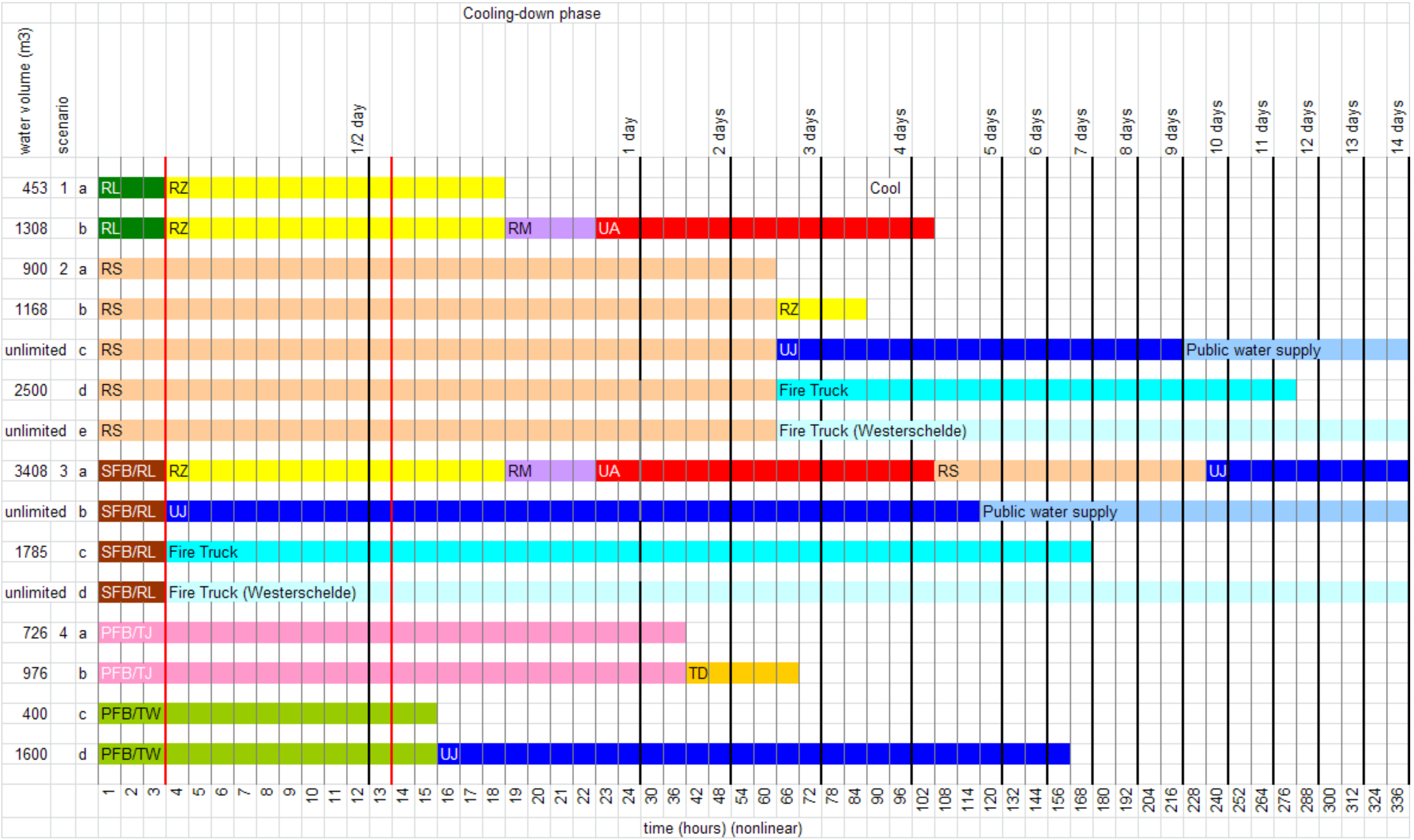

Figure 5.2 Cooling options for cooling down in the LPUHS situation

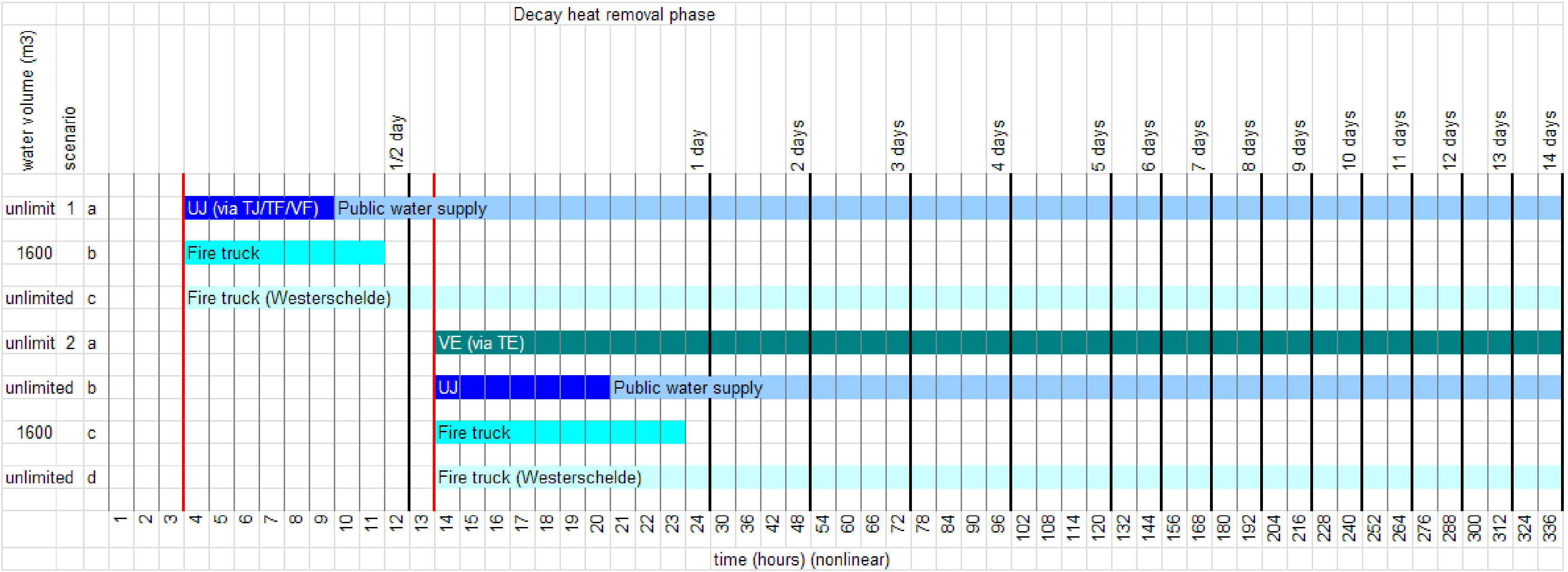

Figure 5.3 Cooling options for decay heat removal in the LPUHS situation

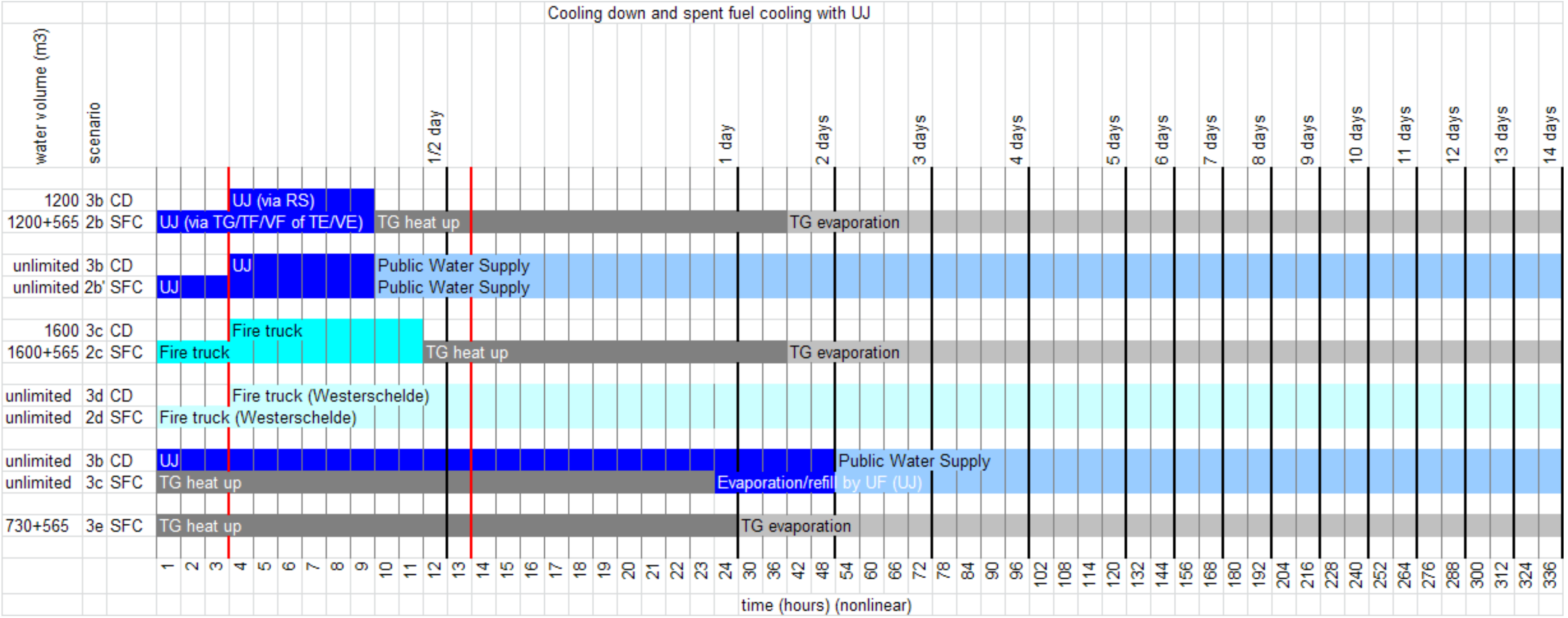

Figure 5.4 Time periods for UJ supply and alternatives for combined cooling down and spent fuel pool cooling in the LPUHS situation

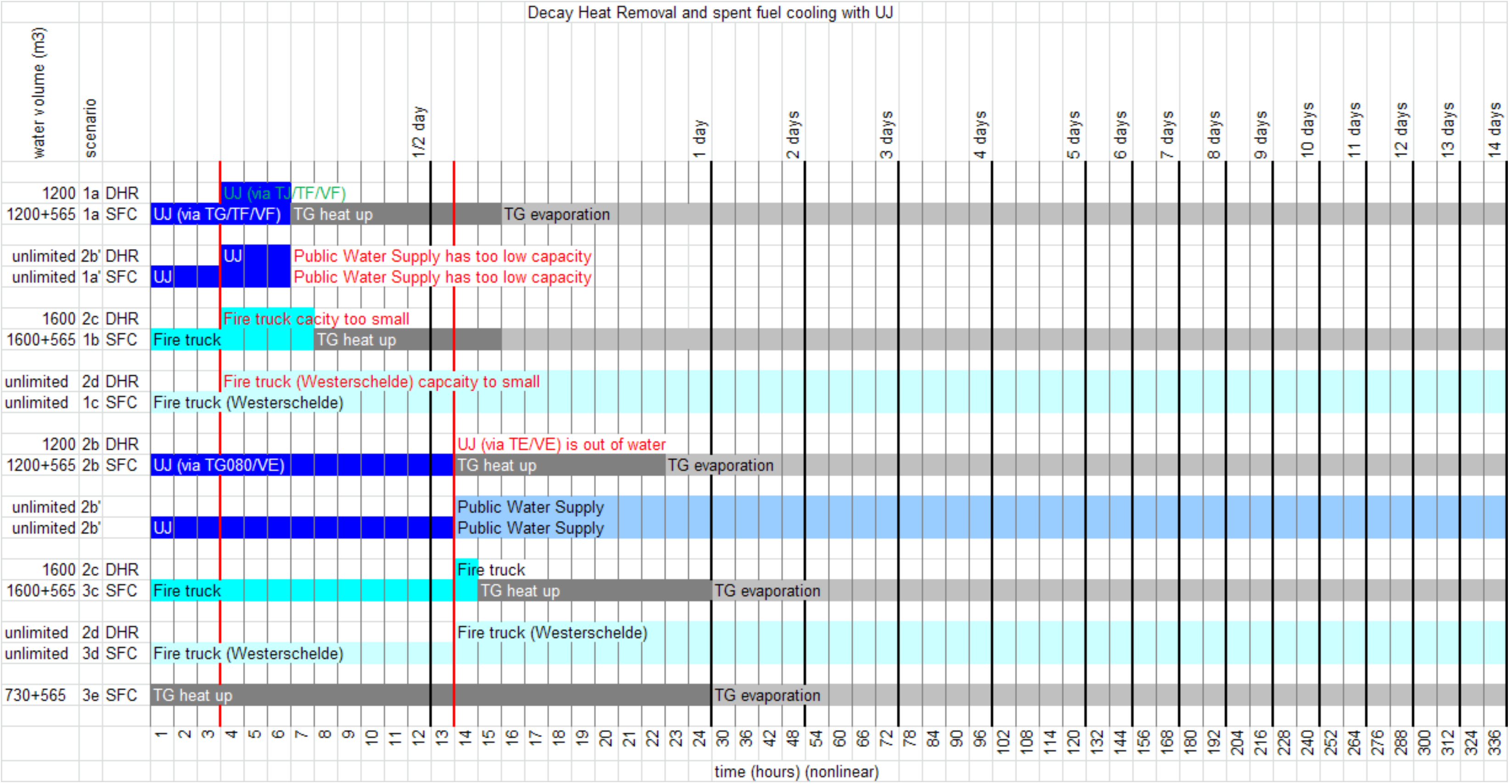

Figure 5.5 Time periods for UJ supply and alternatives for combined decay heat removal and spent fuel pool cooling in the LPUHS situation

Overview of cooling status in case of loss of primary ultimate heat sink

For the situation of loss of primary ultimate heat sink, no fuel damage will occur because by applying the main options presented in section 5.1.2.2.2 this situation is under control (cold shut down); however, alternative options are available. Table 5.6 lists the preferred sequence and the most obvious alternative:

Operational state	Means of cooling	Duration (approx. h)	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 approx. 3 hours
	RS supply	60	
Decay heat removal	Re-establishment of the TJ/TF/VF cooling line by feeding VF by UJ Replenishment of UJ by public water supply system	6, 7[46] (3, 0)[47] unlimited	UJ stock is 1200 m^{3}, replenishment is required
	Switch over to TE/VE cooling	unlimited	

Table 5.6 Cooling status in case of loss of primary ultimate heat sink

5.1.2.2.4 External actions foreseen to prevent fuel degradation

It can be concluded that the plant state LPUHS is controlled by the available on-site systems, so no external actions are necessary.

However, it will be preferable (according to the SAMG) to stretch the cooling down phase for as long as possible because heat removal by heating water (decay heat removal options) instead of evaporation (cooling down options) will require a much greater water inventory (on average 6 times). This is taken into account when assessing the duration of the decay heat removal phase based on available stocks, as indicated in Annex 5.2.

As concluded above external actions are not necessary in this scenario.

However, when deploying the company's own fire brigade, only a small crew will be on-site. An immediate increase in numbers may be required by using off-site volunteers.

Also when water has to be pumped from the CCB's fire fighting pond or from the River Westerschelde to the plant, external support will be necessary. This will be realised in cooperation with the umbrella organisation "Veiligheidsregio Zeeland".

[46] 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 1,020 m^3 that is effective when used for cooling

[47] 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; DHR supplied by UJ starts after three hours and not after 13 hours because UJ would already be empty as a result of spent fuel pool cooling

5.1.2.3 Loss of the primary ultimate heat sink and the alternate heat sink

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 resources as listed in Table 5.4.

As in section 5.1.2.2.2, a cooling-down phase, a decay heat removal phase and the spent fuel pool cooling phase, which all occurs in parallel can be considered separately. The grouping and the division of groups are identical.

Cooling-down phase

All options identified in section 0for the cooling-down phase are also available in the loss of primary and alternate UHS scenario, as listed here:

Group 1: Cooling by RL

Option 1a: **RL/RZ** (main option)

Option 1b: **RL/RM/RZ/UA**

Group 2: Cooling by RS

Option 2a: **RS**

Option 2b: **RS/RZ**

Option 2c: **RS/UJ**; ultimately supplied by the public water supply system (Delta)

Option 2d: **RS/fire truck**

Option 2e: **RS/Westerschelde**

Group 3: Secondary bleed & feed

Option 3a: **secondary b&f/RL/RZ/RM/UA/RS/UJ**

Option 3b: **secondary b&f/UJ**; ultimately supplied by the public water supply system (Delta)

Option 3c: **secondary b&f/ fire truck**

Option 3d: **secondary b&f/ Westerschelde**

Group 4: Primary bleed & feed

Option 4a: **primary b&f/TJ**

Option 4b: **primary b&f/TJ/TA/TB/TD**

Option 4c: **primary b&f/TW**

Option 4d: **primary b&f/TW/TB/UJ**

Decay heat removal phase

VE as an individual system does not fulfill its function when not fed by the deep water wells, so option 2a, is not available. The remaining options, presented in section 5.1.2.2.2 are available and listed here:

Group 1: TJ/TF/VF

Option 1a**: TJ/TF/VF/UJ**; ultimately supplied by the public water supply system (Delta) (main option)

Option 1b: **TJ/TF/VF/fire truck**

Option 1c: **TJ/TF/VF/Westerschelde**

Group 2: TE/VE

Option 2b: **TE/VE/UJ**; ultimately supplied by the public water supply system (Delta)

Option 2c: **TE/VE/fire truck**

Option 2d: **TE/VE/Westerschelde**.

5.1.2.3.1 External actions foreseen to prevent fuel degradation

This is basically not applicable for this scenario, except for those alternatives where the fire brigade is deployed and the water is pumped from the CCB's fire fighting pond or the River Westerschelde.

5.1.2.3.2 Time available to recover one of the lost heat sinks or to initiate external actions and to restore core cooling before fuel damage: consideration of situations with various time delays from reactor shutdown to loss of normal reactor core cooling state (e.g., start of water loss from the primary circuit).

The main options and their alternatives to cool the reactor core, mentioned in section 5.1.2.3, are indicated in Figure 5.6 and Figure 5.7 with the systems used for the respective options. Figure 5.8 presents the situation where UJ and its alternatives are applied for both cooling down and spent fuel pool cooling. Figure 5.9 presents the situation where UJ and its alternatives are applied for both decay heat removal and spent fuel pool cooling. All the options and stocks (in m^3) of the water supply systems are presented in Table 5.5.

Cooling down

For the cooling down phase, Figure 5.6 shows that the situation is identical to the LPUHS situation. This means:

- all options meet the 13 hours criterion set by the decay heat removal chain TE/VE,
- time limits for the cooling down phase from supply are:
    - 18 hours for the main option (1a);
    - a maximum of nine days for option 2c based on available on-site water stocks in the systems that are part of the indicated cooling chain;
    - additional extension of more than 11 days when assuming successive supply by UJ and the CCB pond;
    - unlimited supply of water by the public water supply system (Delta) (extension of option 2c) and the River Westerschelde (option 2e).

Decay heat removal

For the decay heat removal phase (Figure 5.7), the situation is not identical to the LPUHS situation. Two time periods are important with regard to the indicated options i.e. three hours, the minimum time to realise cooling down, and 13 hours, the time required to meet decay heat removal conditions for the TE/VE chain (option 2b). Note that TE/VE (option 2a) is not available due to the non-availability of the deep water wells so VE ducting still can be utilised.

Again, this means that decay heat removal by UJ (on-site) supply in this situations will last until nine hours and 20 hours respectively after shut down. Replenishment of UJ by the public water supply system (Delta) will extend this heat removal for the course of the event (unlimited). An additional supply can be provided for limited time periods from the CCB's fire fighting pond (option 1b) and an unlimited supply from the River Westerschelde (option 1c).

Combined cooling

When the UJ supply is applied for the combined cooling down and spent fuel pool cooling and for decay heat removal and spent fuel pool cooling the situations are identical to the LPUHS situation as shown in Figure 5.8 and Figure 5.9.

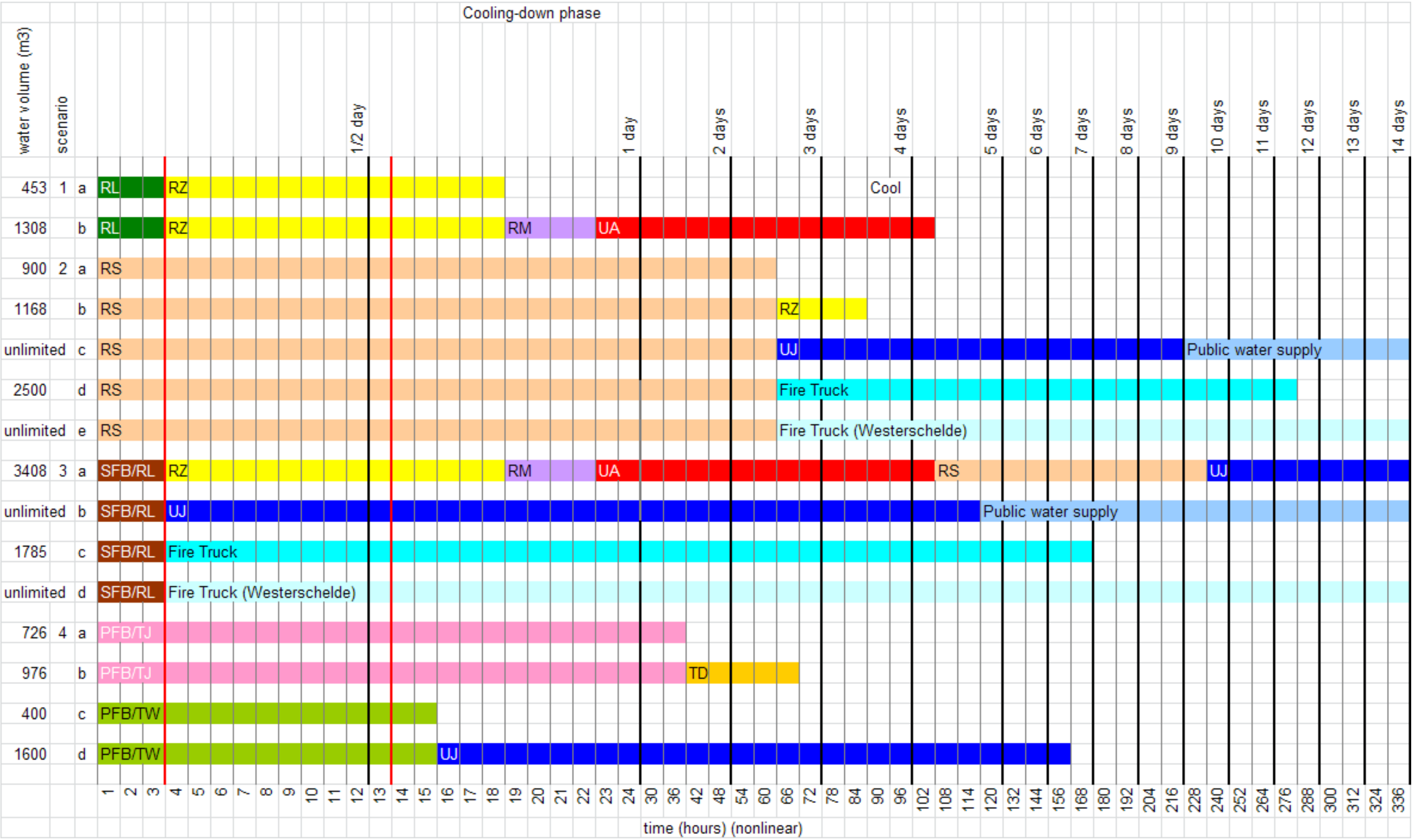

Figure 5.6 Cooling options for cooling down in the LPAUHS situation

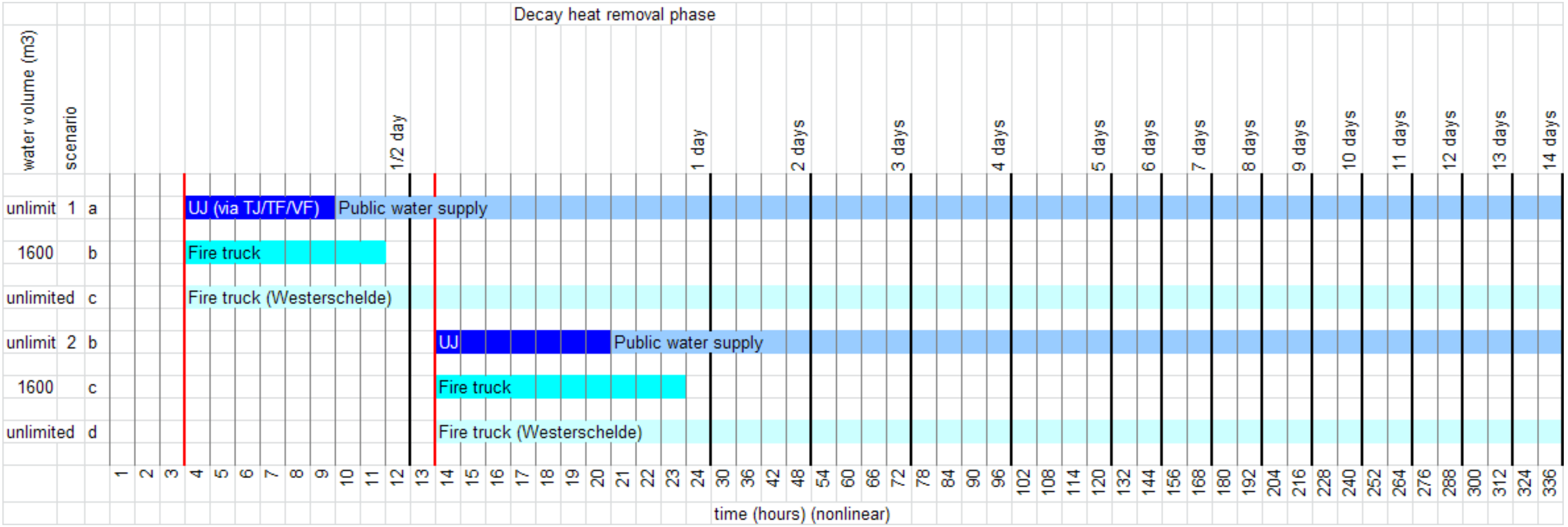

Figure 5.7 Cooling options for decay heat removal in the LPAUHS situation

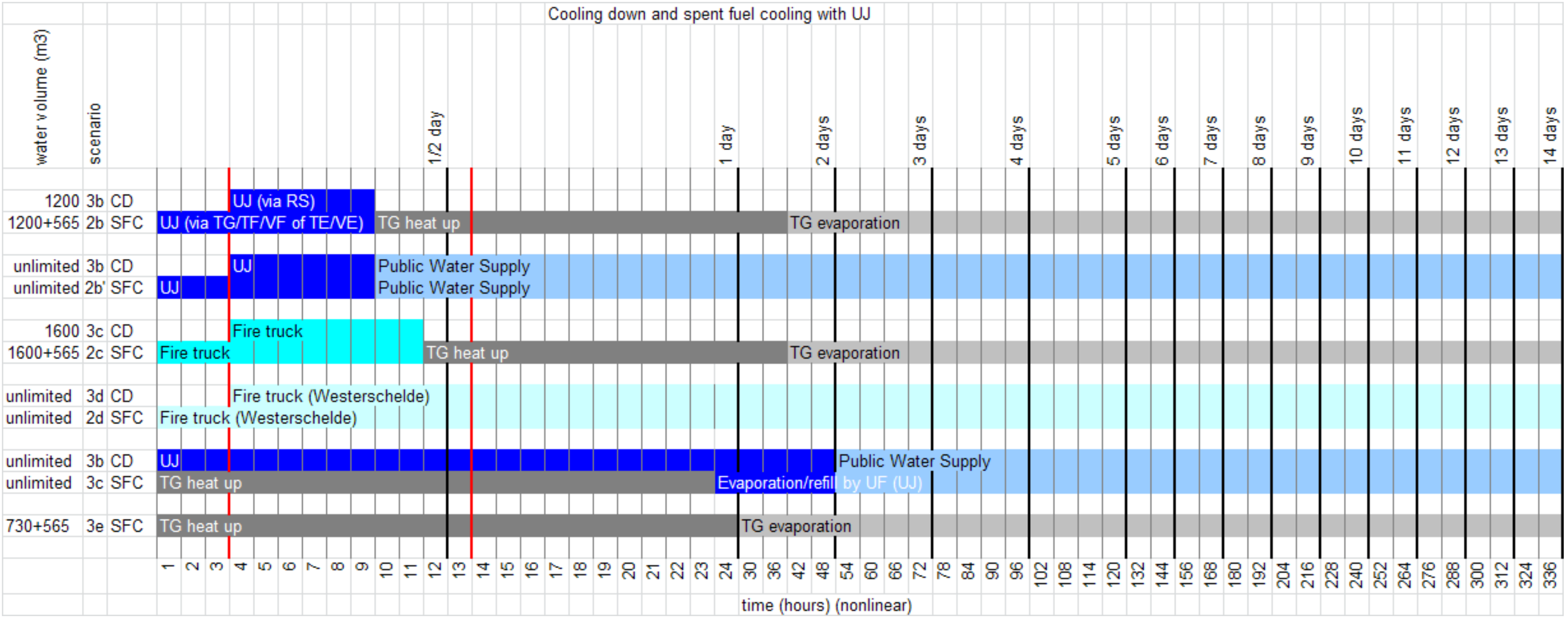

Figure 5.8 Time periods for UJ supply and alternatives for combined cooling down and spent fuel pool cooling in the LPAUHS situation

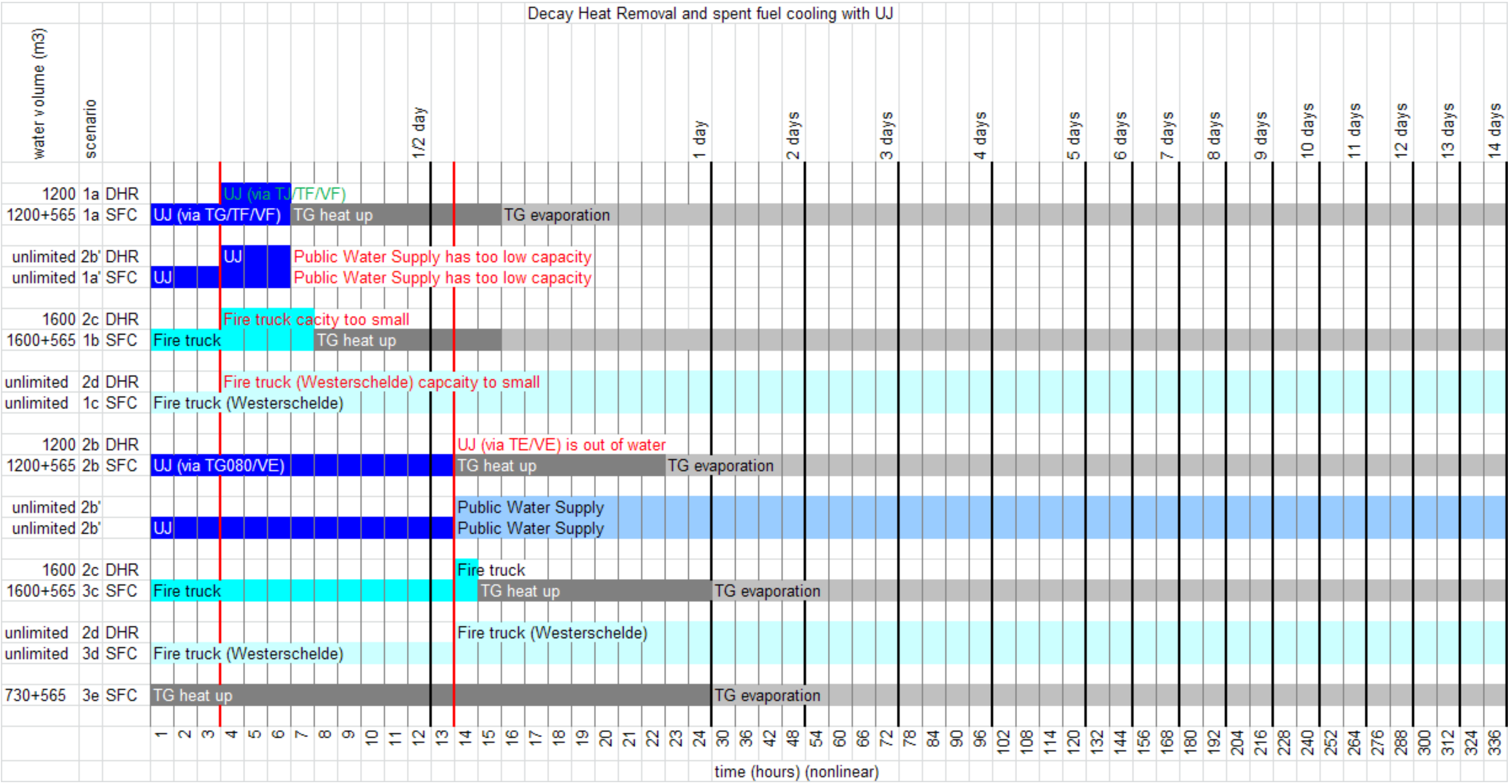

Figure 5.9 Time periods for UJ supply and alternatives for combined decay heat removal and spent fuel pool cooling in the LPAUHS situation

Overview of cooling status in case of loss of primary and alternate ultimate heat sink

When there is a loss of primary and alternate ultimate heat sink, no fuel damage will occur because by applying the main options presented in section 5.1.2.3, this situation is under control (cold shut down); however alternative options are available. Table 5.7 lists the preferred sequence and the most obvious alternatives.

Operational state	Means of cooling	Duration (approx. h)	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 approx. 3 hours
	RS supply	60	
Decay heat removal	Re-establishment of the TJ/TF/VF cooling line by feeding VF by UJ Replenishment of UJ by public water supply system	6, 7[48] (3, 0)[49] unlimited	
	Supply from the CCB's fire fighting pond and the River Westerschelde	8, 10 (-1) unlimited	

Table 5.7 Cooling status in case of loss of primary and alternate ultimate heat sink

[48] It lasts 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 1,020 m^3 That is effective when used for cooling

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

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

It can be concluded that the plant state LPUHS is controlled by the systems that are available on-site, so no external actions are necessary.

However, it will be preferable (according to the SAMG) to stretch the cooling down phase for as long as possible because heat removal by heating water (decay heat removal options) instead of evaporation (cooling down options) will require a much greater water inventory (on average six times as much). This is taken into account when assessing the duration of the decay heat removal phase based on available stocks, as indicated in Annex 5.2.

The plant state LPAUHS can also be controlled provided that an additional supply of water from the public water supply system (Delta) or ultimately from the River Westerschelde can be realised. Otherwise the application of the main options and their alternatives will end when the supplies are exhausted. Time periods for this 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. See the options 1b and 3a (in succession secondary b&f-RL/RZ/RM/AU/RS/UJ) in Figure 5.6;
- the decay heat removal phase only relies on UJ or fire truck supply, which will last 10 hours and 13 hours respectively 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 and free filing as indicated by option 3d (in succession TJ/TN/UF/UJ) occurs.

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

Potential cliff-edge effects

Cliff-edges are characterised by the failure of essential systems in the cooling chain among others caused by a lack of supply. The failure of such systems during a LPUHS event will result in either a switch over to one of the alternative options or in a turn over to the next, more severe, plant state. Ultimately, the plant state will be identical to the UHS-SBO 2 situation.

The following potential cliff-edges have been identified for this scenario when mitigation of the event is performed by the main options for cooling down, decay heat removal and spent fuel pool cooling:

- failure of the relief valves of the main steam system RA;
- failure of RZ system;
- failure of UJ water supply by:
    - failure to establish connections (among others by deploying the fire brigade);
    - premature exhaustion of the water reserves of the low pressure fire extinguishing system UJ. This could be because of it being used for other reasons at the same time, like extinguishing fires.

Taking the alternatives into account, additional potential cliff-edges can be identified:

- failure of more than two pumps of the Backup cooling water system VE;
- failure of the VE duct;
- failure of the heat exchanger of the Backup residual heat removal system TE,
- failure of the heat exchanger of the Spent fuel cooling system TG080;
- failure of water supply by:
    - failure to establish connections to replenish basic stocks (action of operator or the fire brigade);
    - damaged or unavailable fire hose connection on the connecting duct between the two redundancies of the RS system;
    - exhaustion of the water reserves of the low pressure fire extinguishing system UJ. This could be because of it being used for other reasons at the same time, like extinguishing fires.

Potential actions to prevent cliff-edge effects

The handling of the mentioned cliff-edge effects is described below. However, the presence and completeness of the relevant procedures still needs to be checked.

<u>Failure of the relief valves of the main steam system RA</u>

The main steam system consists of two trains. Each train is provided with two relief trains and two ducts with five safety valves each. One relief valve or one safety valve is sufficient for removal of the decay heat The potential action would be to proceed to local manual control of the RA relief valves. Procedures to operate/act in this way are not available; analyses are not performed, however steps to reach the valves have been installed.

In case of a complete failure of the RA relief valves, the secondary pressure will remain high and cooling with the low pressure pumps cannot be realised. However, the safety valves will protect the secondary system from over pressurisation. The reactor can then be brought into the hot shut down condition only. As a result, this is not a cliff-edge.

Failure of RZ water supply

With a failure of RZ water supply due to either system failure or lack of water, alternatives to supply water are at hand, so a switch over to an alternative option can be realised. As a result, this is not a cliff-edge.

Failure of UJ water supply

- Failure to establish connections (e.g. deployment of fire brigade);

  An instruction is available to establish the connection of UJ to VF; deployment of the company's own fire brigade is not a problem;

- premature exhaustion of the water reserves of the Low pressure fire extinguishing system UJ;

  As soon as UJ is anticipated to be used for cooling, the connection to the public water supply system (Delta) should be checked for availability and the additional connection (reserve) should be made ready;

- alternatives for the UJ water supply are the CCB fire fighting pond, the public water supply system and ultimately the Westerschelde.

As a result this is not a cliff-edge, although procedures for this are not readily available and must be made so.

Failure of more than two pumps of the backup cooling water system VE

The VE system is equipped with eight pumps, each with its own deep water well. These pumps are connected to the plant through one single duct. Six pumps are required for proper functioning of the system. When operating with less than six pumps, VE does not remove all the released decay heat immediately and equilibrium will be set at an elevated temperature. This may cause components of the connected systems to be damaged. In case this happens, VE should be considered as not available, thus transferring the event to the scenario of loss of primary and alternate UHS. However, with less then six pumps heat removal is still possible (delayed) and also heat removal from the spent fuel pool is possible.

Failure of the VE duct

The eight pumps of VE system are connected to the plant by one single duct, and failure of this duct will result in loss of the VE system.

Failure of the heat exchanger of the Backup residual heat removal system TE

The TE system is equipped with a single heat exchanger, that is (partly) equipped with plastic seals. It therefore cannot withstand temperatures that are higher than it is designed for. This means that in an LPUHS event with an unanticipated too early operation of TE, when temperatures are not yet low enough, the cooler might be lost. In case this happens, TE should be considered as not available, so the options for decay heat removal using the conventional emergency cooling water ssystem VF over UJ or the fire brigade should remain available.

Failure of the heat exchanger of the Spent fuel cooling system TG080

TG080, the added heat exchanger to the TG system to be cooled by VE, will not be subjected to higher conditions then its design conditions. This means that failure will not occur so this is not a cliff-edge.

Failure of water supply

- failure to establish connections to replenish basic stocks (action of operator or of fire brigade).

  Replenishment of basic stocks by alternatives might fail due to procedures that are not specific to the situation under consideration;

- damaged fire hose connection on the connecting duct between the two redundancies of the RS system.

  This connection can be used in case for the alternative options the RS pumps are not available, but only a single connection exists. If it is damaged, or cannot be reached, no alternative equivalent connection is available so no connection with the secondary system can be made;

- finally, exhaustion of the water reserves of the low pressure fire extinguishing system UJ.

  UJ provides the ultimate water supply that could be exhausted due to it being used for other reasons at the same time, like extinguishing fires.

  As soon as the use of UJ for cooling is anticipated, the connection to the public water supply system (Delta) should be checked for availability and the additional connection (reserve) should be made ready. Alternative (external) supplies can be found by supply from the River Westerschelde via fire fighting equipment which has to be provided by the fire brigade. Procedures for this are not available.

Measures which can be envisaged to increase the robustness of the installation

Potential actions to increase the robustness of the installation are:

- develop a set of Extensive Damage Management Guides (EDMG) and implement a training program. Issues to be addressed:
    - procedure for direct injection of VE by UJ;
    - alternative supplies for UJ.

5.1.3 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)

Design provisions to prevent loss of primary UHS with SBO 1

The primary ultimate heat sink is the water from the River Westerschelde, supplied by the main cooling water system VC and the conventional emergency cooling water system VF.

Loss of off-site power and station black out (referred to as SBO 1) is a plant state characterised by the unavailability of both the supply from the external grids 150 kV and 10 kV (basically non feeding of the BA and BB buses by the CCB 6 kV bus) and the Emergency Grid 1 (NS 1). In this case, the Emergency Grid 2 (NS 2) is still functional, as are the DC (battery) and uninterrupted AC (380V) power systems.

The scenario of loss of primary UHS with SBO 1 is a combination of the scenario described in sections 5.1.2 and 5.1.2.2. The relevant design provisions to prevent this situation are included in these sections. Results with regard to availability (duration without replenishment of diesel supply) of Emergency Grid 2 (NS 2) will be presented.

The relevant systems are connected to the respective emergency grids NS1 and NS 2 as indicated in Table 5.8. Note that the mobile diesel generator EY080 is a back-up of NS 2.

System	System code	Connected to
Pressure relief valves	RA	NS 2/EY080 UPS
Emergency feedwater pumps (3)	RL	NS 1 (2) and 1 steam turbine pump
Demin water supply system	RZ	NS 1
Demin water preparation system	UA	regular grid
Backup feed water system	RS	NS 2 EY080
Low pressure fire extinguishing system	UJ	emergency power CCB and diesel pump
Backup residual heat removal system	TE	NS 2/EY080
Component cooling water system	TF	NS 1
Spent fuel pool cooling system (1+2)	TG	NS 1 (1 pump) NS 2/EY080 (2 pumps)
Safety injection system & residual heat removal system	TJ	NS 1
Backup coolant makeup system	TW	NS 2/EY080
Backup cooling water system	VE	NS 2/EY080
Conventional emergency cooling water system	VF	NS 1
Primary pressure system valves	YP	NS 2/EY080 UPS

Table 5.8 Connection of relevant (cooling) systems by the emergency grids NS 1 and NS 2

Design provisions to prevent fuel damage in case of loss of primary UHS with SBO 1

In the case of a loss of the primary ultimate heat sink, the main cooling water system VC and the conventional emergency cooling water system VF stop functioning, because of, for example, station blackout. In this scenario, this is combined with station black-out type 1, which is characterised by the unavailability of all off-site power and the emergency power system NS 1. As seen in section 5.1.2, a cooling-down phase, a decay heat removal phase and the spent fuel pool cooling phase, which occurs in parallel can be considered separately. The grouping and division of groups is identical. From Table 5.8 it can be concluded which systems are available as they are either operated independently of electrical power or connected to the remaining emergency power supplies.and the table shows which systems are not available in the SBO 1 situation because they are fed by the main electric system or NS 1.

Reactor core cooling

All the options, as described in section 5.1.2, that still are available are indicated here. For generic information per option please refer to section 5.1.2.2.2. Additional information relating to the LPUHS-SBO 1 situation is inserted here. As in the previous chapters, a cooling-down phase and a decay heat removal phase are considered separately.

When comparing the groups and their options for the situation of loss of primary ultimate heat sink with (see section5.1.2) and without (this chapter) NS1, it appears that for cooling down from the “group headers” RL is limited (only one (100%) steam driven pump) available for a limited period of time and that from the sub-dividing support or supply systems only UJ and the fire truck (for “secondary” cooling) and TW (for “primary” cooling) remain available. For decay heat removal TJ and for spent fuel pool cooling TF/VF are not available. Only the TG080/VE (group 2) remains available for spent fuel cooling.

Cooling-down phase

When cooling down, according to the preferred order indicated in section 5.1.2, the limited application of the RL system should be the first and main option. However due to this limitation RS will automatically take over the cooling down function as result of a low steam generator level, supplemented by UJ and/or the fire truck (by action of operators and/or the fire brigade). The remaining options, with additional information, are:

Group 1: Cooling by RL

Option 1a: **RL/RZ**

This option is limited in time, and is only available in this scenario because the emergency RL pump driven by a steam turbine can be applied and the RZ supply is not available. Cooling-down by the RL system in this way and in combination with RA, is limited to the period when the secondary pressure is higher than 6 bar or when the RL tank is exhausted. The estimated/assessed duration of this period is three hours (exhausting the RL tank), provided that this combined RL/RA systems can operate during the course of the event. This means that:

- the seal cooling of this pump can be maintained via the UJ/UK[50] systems. The outlet steam is dumped in the ambient air;
- RL/RA valves remain in operation due to electrical power provided by the uninterrupted electrical power system (no exhaustion of batteries during the said period of three hours);
- RL/RA valves can be operated local and manually.

If the decay heat removal conditions are not met by the time RL fails (e.g. exhaustion of water supply), option 2a will automatically take over (low steam generator level).

[50] 18 hours

Group 2: Cooling by RS

Option 2a: **RS**

Option 2c: **RS/UJ**; ultimately supplied by the public water supply system

The RS system is supplied from emergency power supply system NS 2, and the UJ system can be kept pressurised either by its diesel powered pump or the electrical pumps fed by the CCB's emergency power system. Two options are possible:

1. Supply of UJ to the RS tanks, then the RS pumps will feed the steam generators;
2. Supply by UJ direct to the steam generator via a flexible connection between UJ (hydrant) and the fire hose of the RS system.

Ultimately, the UJ system will be supplied by the public water supply system (which assumes that in spite of SBO, UJ is still fed by the public water supply system; otherwise this option will be limited to when UJ is exhausted).

Option 2d: **RS/fire truck**

Option 2e: **RS/Westerschelde**

Group 3: Secondary bleed & feed

Option 3a: **secondary b&f/RL/RZ/RM/UA/RS/UJ**

When the RZ and UA systems are not available anymore, the RS system is expected to be able to bridge the gap between the secondary bleed and feed and the decay heat removal phase

Option 3b: **secondary b&f/UJ**; ultimately supplied by the public water supply system

Option 3c: **secondary b&f/ fire truck**

Option 3d: **secondary b&f/Westerschelde**

Group 4: Primary bleed & feed

Option 4c: **primary b&f/TW**

Option 4d: **primary b&f/TW/TB/UJ**

As TB is not available, the direct drain of water from UJ into the TW tanks by the fire hoses is possible while the backup systems bunker remains open.

Decay heat removal phase

Decay heat is normally removed by the safety injection system & residual heat removal system TJ or the backup residual heat removal system TE. In the loss of UHS scenario combined with SBO1, the "heading " TJ system, especially the TJ pumps, is not available and therefore the entire group. This also means that the main option for decay heat removal, option 1a TJ/TF/UJ declines. However, TE/VE is still available so the remaining options, in conformity with the listing in section 5.1.2 are available. Option 2a will become the main or preferred one:

Group 2: TE/VE

Option 2a: **TE/VE**

Option 2b: **TE/VE/UJ**; ultimately supplied by the public water supply system

Option 2c: **TE/VE/fire truck**

Option 2d: **TE/VE/Westerschelde**.

Design provisions to prevent fuel damage in case of loss of primary UHS with SBO 2

The primary ultimate heat sink is the water from the River Westerschelde, supplied by the main cooling water system VC and the conventional emergency cooling water system VF.

Total loss of AC-power (referred to as SBO 2) is a plant state characterised by the unavailability of:

- supply from the external grids 150 kV and 10 kV and by the dedicated 6 kV line of CCB;
- Emergency Grid 1 (NS 1);
- Emergency Grid 2 (NS 2);
- the mobile diesel generator EY080[51],
- the emergency power system of the coal fired power plant (CCB); which is a back-up for the diesel generators EY040 and EY050[52].

In this case, the battery and uninterrupted AC (380 V) power systems (UPS) are still functional.

The scenario loss of primary UHS with SBO 2 is a combination of the scenarios described in sections 5.1.2 and 5.1.1.3. The design provisions to prevent this situation are included in 5.1.2 and 5.1.1.3.

---

[51] Maintaining EY080 as back-up for EY040 and EY050 will turn the SBO 2 situation functionally into SBO 1, because the emergency power system 2 will remain in operation while it is fed by another diesel generator Fed by the plant's diesel stock this can operate for 10 hours, then replenishment is needed.See Table 5.10

[52] For this backup by CCB the same applies as for EY080, only this time a replenishment of fuel is needed after nine hours.

The relevant systems are connected to the respective emergency power systems NS 1 and NS 2 as indicated in Table 5.8. Note that the mobile diesel generator is a back-up to NS 2.

Design provisions to prevent fuel damage in case of loss of primary UHS with SBO 2

When there is of a loss of primary ultimate heat sink, the main cooling water system VC and the conventional emergency cooling water system VF stop functioning , because of, for example, station blackout. In this scenario, this is combined with SBO 2, which is characterised by the unavailability of both the off-site power and both NS 1 and NS 2. Therefore, the backup cooling water system VE is also lost, which makes the loss of primary UHS scenario here equivalent to the loss of primary and alternate UHS scenario.

Additionally, it is postulated that the CCB emergency power system is also not available .

As in the previous chapters, a cooling-down phase, a decay heat removal phase and the spent fuel pool cooling phase, which occurs in parallel can be considered separately. The grouping and division of groups are identical.

Reactor core cooling

Due to a complete station black out only those systems that can operate without electrical power or have their own power generator are still available during the SBO 2 situation. Loss of primary ultimate heat sink and loss of alternate ultimate heat sink is already included in this SBO 2 situation. This means that there is no difference between SBO 2 and LP(A)UHS- SBO 2. Therefore, for heat removal only the following are available:

- limited RL ;
- RS duct (this is for the "external water supply", so group 2 disappears);
- UJ;
- fire truck.

With these systems, options for cooling have to be determined.

Cooling-down phase

The first part of the cooling down will be performed by RL which is limited by its supply (only the RL tank), followed by a supply of UJ and its supporting supplies feeding the steamgenerator via the RS connection. The remaining options are as followes:

Group 1: Cooling by RL

Option 1a: **RL**

As long as a RL water supply is available cooling can be performed because a water supply to the steam generator is provided by the steam-turbine driven RL pump. A switch over to alternative options is then necessary. As with group 3 this means that water will be supplied by UJ (plus the public water supply system, fire truck and/or River Westerschelde.

Group 3: Secondary bleed & feed

For the group 3 options, the first part of the cooling down is performed by bleed and feed by RL followed by a successive water supply through the fire hose connection on the connecting duct between the two redundancies of the RS system for direct injection in the steam generators. In order to allow a water supply by the low pressure options e.g. UJ and the fire trucks, the steam generators need to be depressurised after RL-feed.

If RL fails an alternative is to depressurise the steam generators by opening the relief valves (RA) to atmospheric level. An immediate application of these systems then means that , part of the steam generator inventory will be released. This will cause a temperature transient for the steam generator as well as the reactor vessel (temperature decrease > 100 K/h). The options with failed RL are marked with an apostrophe (').

Option 3b: **secondary b&f/UJ**; ultimately supplied by the public water supply system

The UJ system can be kept pressurised by its diesel powered pump and can supply water direct to the steam generator via a flexible connection between UJ (hydrant) and the fire hose of the RS system.

When the on-site stock of UJ is exhausted, the UJ system will be supplied by the public water supply system (assuming that the public water supply system has its own electrical and emergency power system, thus ensuring the supply to UJ even in the SBO situations of KCB, otherwise this option is only available whilst UJ has stock).

Option 3b’: **no secondary b&f/RL/UJ**; ultimately supplied by the public water supply system

There is immediate depressurization of the steam generators within one hour without utilisation of the bleed & feed option. The water supply to the steam generators is by UJ via the fire hose connection on RS.

Option 3c: **secondary b&f/fire truck**

The UJ supply for this option is replaced by a supply by the fire truck that takes water from the CCB’s fire fighting pond.

Option 3c’: **no secondary b&f/RL/fire truck**

Option 3d: **secondary b&f/Westerschelde**

The UJ supply in option 2c is replaced by a supply by the fire truck that has suctioned water from the River Westerschelde.

Option 3d’: **no secondary b&f/Westerschelde**

The options of group 3 can, or preferably will, be operated in succession.

Decay heat removal phase

Decay heat is normally removed by the Safety injection system & Residual heat removal system chain TJ/TF/VF or the ultimate residual heat removal system TE/VE. In the loss of UHS scenario combined with SBO 2, both the regular cooling chain TJ/TF/VF and the reserve cooling chain TJ/TE/VE are not available anymore. This means the “headers” of the decay heat removal groups are not available anymore. Therefore switch over to the decay heat removal phase is not possible, the plant shall be kept in the cooling down phase preferably at hot shut down conditions.

5.1.3.1 Time of autonomy of the site before loss of normal reactor core cooling condition (e.g. start of water loss from the primary circuit)

5.1.3.1.1 Time of autonomy in case of loss of primary UHS with SBO 1

Cooling

The main options and their alternatives to cool the reactor core, described above, are indicated in Figure 5.10 and Figure 5.11, which show the systems that are used for the respective options. Figure 5.12 presents the situation in which UJ and its alternatives are applied for both cooling down and spent fuel pool cooling. Figure 5.13 presents the situation in which UJ and its alternatives are applied for both decay heat removal and spent fuel pool cooling.

The introduced distinction in phases, groups and options, as well as the indicated ranking of preferred options and the succession of applied systems are maintained. Stocks (minimum available water stocks in m^3) of the water supply systems for all the options are presented in Table 5.5.

When comparing the plant states loss of primary ultimate heat sink with and without NS 1, the main options change: cooling down using the first option is limited while the first options are not available for decay heat removal and spent fuel pool cooling. For the cooling down phase this means that cooling down has to be performed for at least 13 hours because the remaining decay heat removal options (TE/VE) requires this length of time.

Cooling down

For the cooling down phase, see Figure 5.10, a switch over to option 2a will be performed because the principle main option (option 1a) is not acceptable on its own. In fact the combination of RL and RS is option 3a. Option 2a, on its own can perform cooling down for approx. 60 hours; therefore the decay heat removal conditions will be met. The duration of this option can be increased by a supply of other systems indicated in the remaining option of group 2. The remaining options in group 3 are more complex to implement but can meet the decay heat conditions. Furthermore secondary bleed & feed combinations with UJ (option 3c) or the fire truck (option 3 d) can run for approx. 114 hours and 168 hours respectively.

These available options last for the same duration as with the preceding events.

Decay heat removal

For the decay heat removal phase (see Figure 5.11), only the TE/VE group applies; therefore the principal main option (option 1a) is replaced by option 2a which will basically supply water during the entire event.

After 13 hours, a switch-over from cooling down can be performed. In case the deep water wells are not available, a supply from the alternative options of group 2 can provide cooling for at least seven (option 2b) to ten hours (option 2c) using on-site stocks, which can extend to infinity when supplies from the public water supply system or the River Westerschelde are used.

Combined cooling

When the UJ supply is applied to the combined cooling down and spent fuel pool cooling and for the combined decay heat removal and spent fuel pool cooling (see Figure 5.12 and Figure 5.13) the cooling down situation is identical to the LPUHS situation. For decay heat removal the preferred options of the LPUHS situation (see section 5.1.2.2.2) remain.

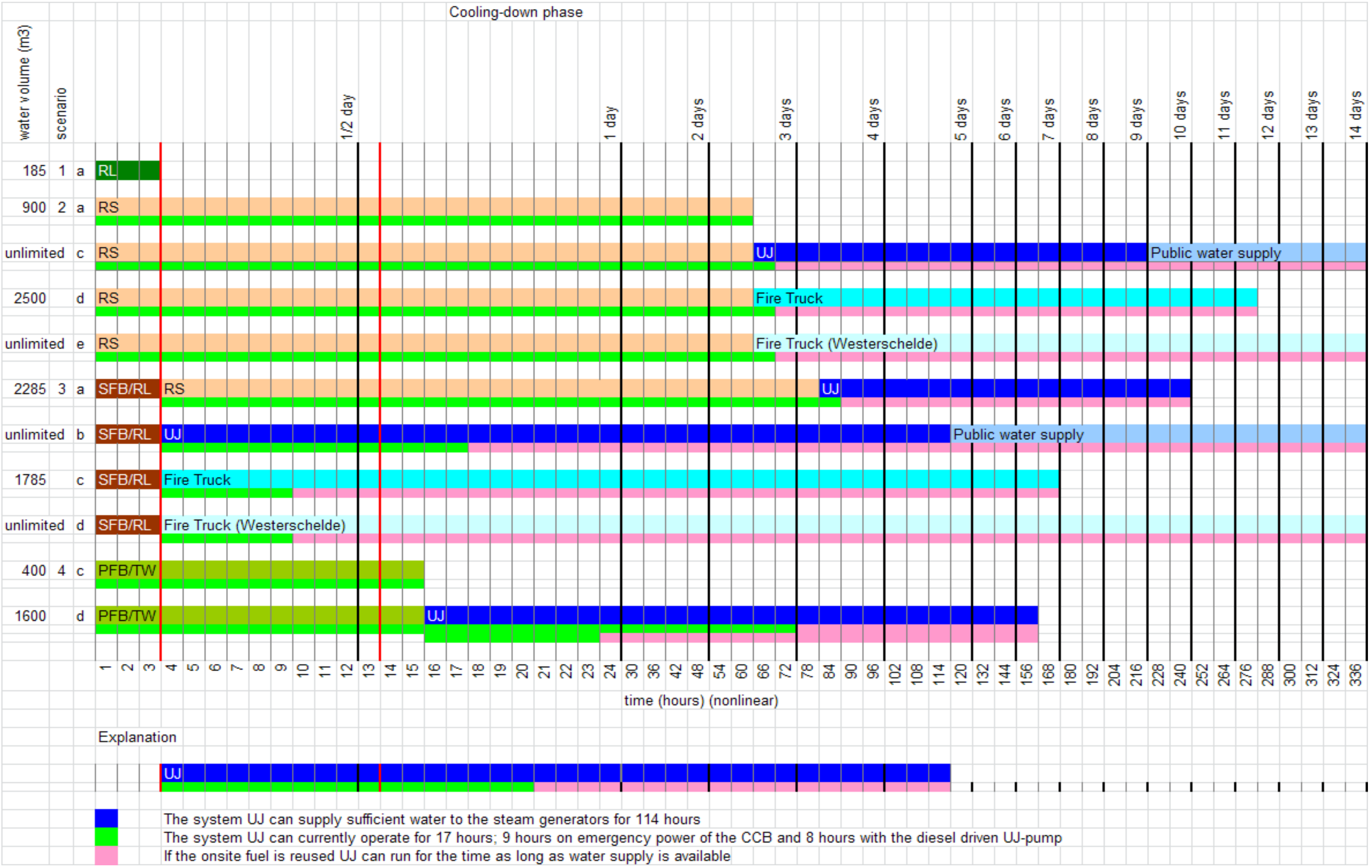

Figure 5.10 Cooling options for cooling down in the LPUHS-SBO 1 situation

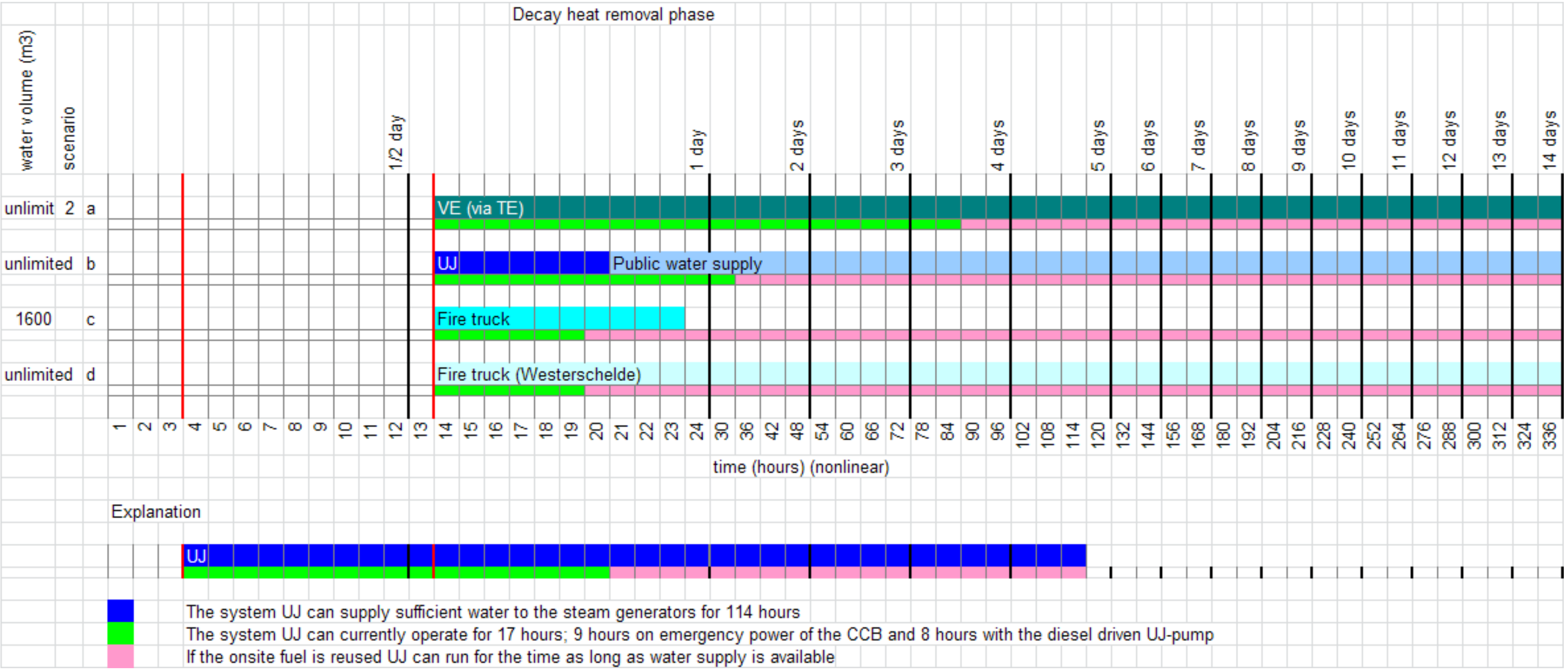

Figure 5.11 Cooling options for decay heat removal in the LPUHS-SBO 1 situation

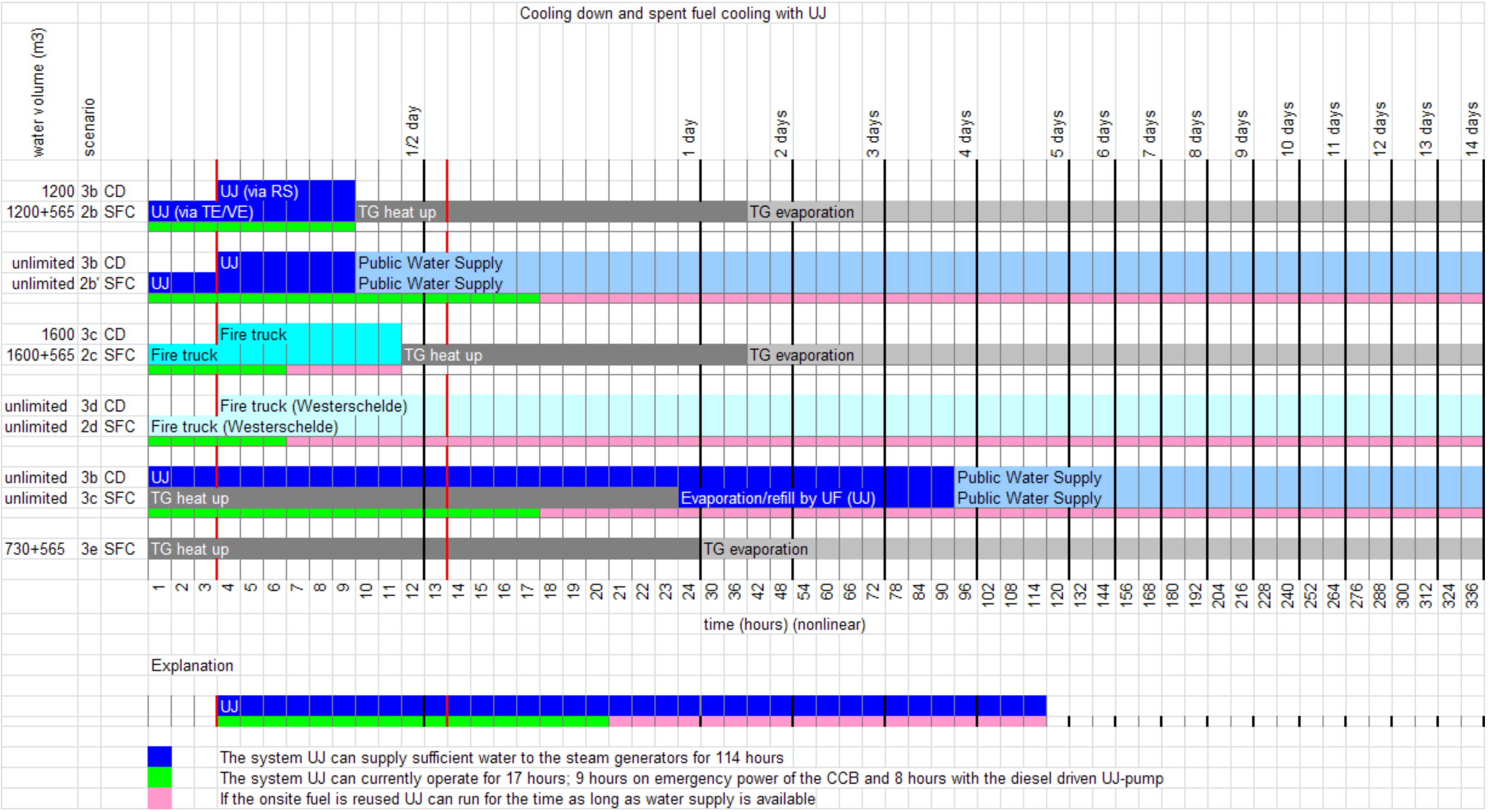

Figure 5.12 Time periods for UJ supplies and alternatives for combined cooling down and spent fuel pool cooling in the LPUHS-SBO 1 situation

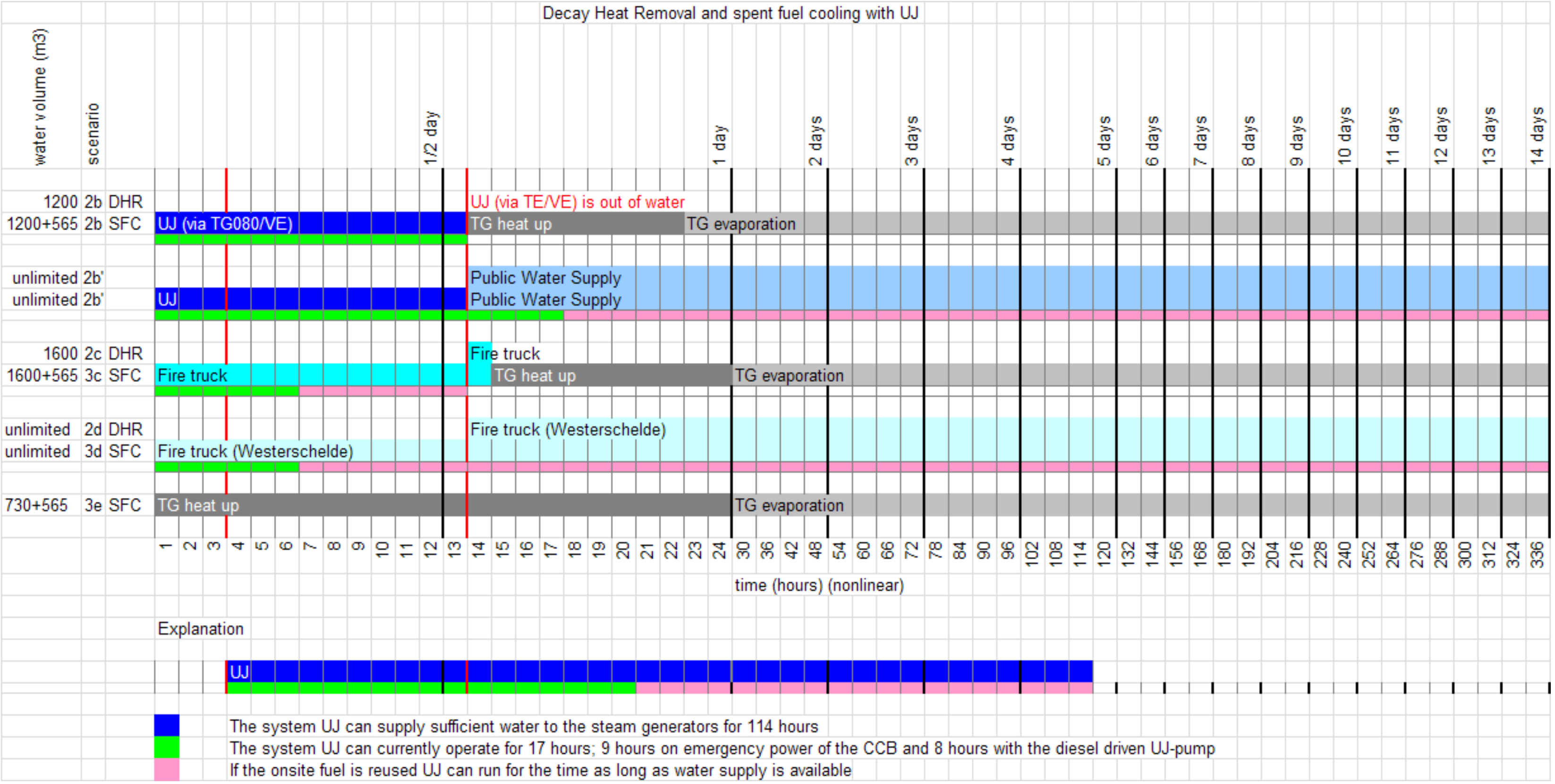

Figure 5.13 Time periods for UJ suppies and alternatives for combined decay heat removal and spent fuel pool cooling in the LPUHS-SBO 1 situation

Overview of cooling status in case of loss of primary heat sink in combination with SBO 1

As regards the situation of loss of primary heat sink in combination with SBO 1, no fuel damage will occur because by applying the options presented above, this situation is under control (cold shut down); however, alternative options are available. Table 5.9 lists the preferred sequence and the most obvious alternatives.

Operational state	System	Duration (approx. h)	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)[53] unlimited	

Table 5.9 Cooling status in case of loss of primary heat sink in combination with SBO 1

[53] For combined decay heat removal and spent fuel pool cooling, the results are presented in brackets. Thewater stock of 1,020 $m^3$is effectively used for cooling

Conclusion

It will be preferable (approved by the SAMG) to make the cooling down phase as long as possible because heat removal by heating water (decay heat removal options) instead of evaporation (cooling down options) will require a much greater water inventory, on average up to six times as much. This is taken into account when assessing the duration of the decay heat removal phase based on available stocks, as indicated in Annex 5.2. However, this situation is basically covered by the TE/VE operation.

Electrical power supply

The internal back up provisions for SBO 1 are summarised below:

For AC power

- emergency Grid 2 (NS 2)

  NS 2 provides electrical power to the relevant safety related system to provide a safe reactor state;

- in the event that only the 150 kV coupling to KCB fails, the supply to NS 2 from the 10 kV is still available;
- supply by the coal fired power plant (CCB)

  CCB is accounted for as on-site. The options are:

  - 6 kV connection

    In the event that only the 150 kV coupling to KCB fails, an additional supply to NS 1 can be provided;

  - emergency diesel generators CCB

    This is an additional supply to KCB; which is considered to be sufficient for NS 2,

- mobile diesel generator EY080

  In case there is a total station blackout, a mobile diesel generator EY080 is available on site. However, EPZ needs external support for its transportation; therefore this option is not considered as a real internal backup option.

For DC power

The autonomy periods for the systems to provide electrical DC power in case of LOOP are presented in the following. It is indicated that as long as AC power is available the DC-power system is in operation. As soon as AC power is lost, the DC-system is powered by the batteries.

The DC power system batteries of ± 24 V/220 VDC, are available when the main power system, emergency grid 1 and emergency grid 2 are out of service. The batteries deliver electrical power for at least two hours (design requirement) to all DC-consumers which need uninterrupted DC-power . In reality the discharge times are more than two hours. Table 5.2 lists the real discharge times of the batteries of the different systems.

The discharge time of the 220 V batteries can be increased to 4.0 hours by connecting the EC-bus to the DC/AC-convertors and when the emergency oil pump of the turbine-generator is switched off, the discharge time can increase to 5.7 hours. The uninterrupted AC-busses are energized by DC/AC-convertors, which are fed by the 220 V DC-busses.

Fuel stocks

The minimum available diesel fuel in stocks is 245 m^3. It will depend on the current situation to get all of the stocks at the right place during an event.

Nevertheless, depending on their availability and the fuel consumption related to produced power, the running time of one single diesel generator can be extended to a total of 280 hours (EY010, EY020) or even 1,300 hours (EY040, EY050) before exhausting of the stocks.

However because not all of the stocks can be transferred, shorter running times have been assumed.

Table 5.10 shows an overview of fuel stocks and running times of the diesel generators and the time necessary to switch from one system to the other system. NS 1 is available under LOOP conditions, NS 2, the mobile emergency generator or the CCB diesel generator are available during SBO 1; no emergency generators are available during SBO 2[54]. Note that the runtimes in Table 5.10 for the CCB diesel generator and the mobile diesel generator are at full load and for EY010/020/030/040/050 at expected load.

It is noticed that in parallel to these provisions, the low pressure fire extinguishing system UJ has one diesel generator driven pump. The diesel consumption is 75 l/h at full power, so with using the plant's own stock of 600 l, the running time is eight hours.

Figure 5.10 to Figure 5.13 show systems that provide cooling water to the reactor. These depend on electrical power for their operation, which is supplied either by the backup systems indicated above or are driven by their own diesel motor. The periods of time these systems supply power (electrical or mechanical) based on supplied diesel fuel is indicated. NS 2 "runs" for 72 hours, the fire trucks need refuelling after three hours, the UJ system has a nine hour principal run time and the CCB emergency power system runs for eight hours on its "own" supply.

[54] While maintaining EY080 and or the CCB's emergency diesel generator as a backup for NS 2 (EY040 or EY050), the SBO 2 situation will functionally turn into SBO 1, because the NS 2 will remain in operation even if it is only fed by its backup.

Run times on “own” stock is indicated by small green bars, replenishment is indicated by small pink bars. These options are combined with the indicated run times.

Table 5.10 shows how long the water supplying system can be in operation.

System NS 1	Stock (m^3)	Running time (h)	Switch/ connecting time (h)
EY010	95	79	
EY020	95	79	
EY030	30	25	
System NS 2			
EY040	4	22.5	
EY050	4	22.5	
With extra tank	8.8	72	
NS 1 → NS 2			1
Mobile diesel generator			
EY080	3	10	
EY080 → NS 2			6[55]
CCB diesel generator			
	4	9 (1 x 1 MW)	
CCB → NS 2			4

Table 5.10 Stock levels and running times of emergency diesel generators

Time period for cooling

The main options listed in section 5.1.3.1.1 will not suffer a lack of water supply during the course of the event, as limiting conditions will stem from the availability of fuel stocks for the diesel generators.

In the SBO 1 situation, only diesel generator EY040 and EY050 have to operate longer than the indicated 72 hours if the event continues. However, the replenishment of diesel by internal, or later by external, means can be provided, ultimately within these 72 hours. It is noted that no means and procedures are available for the performance of this action.

Parallel to this, based on E-power, the UJ supply is ensured for a period of 17 hours provided that the CCB diesel generators and the UJ diesel generator driven pump are operated in succession, and that LOOP-SBO 1 applies for both plants.

[55] Inclusive transportation time of some hours

5.1.3.1.2 Time of autonomy in case of loss of primary UHS with SBO 2

Cooling

The main options and their alternatives to cool the reactor core, described in detail above are indicated in Figure 5.14 which shows the systems used for the respective option. Figure 5.15 presents the situation when UJ and its alternatives are applied both for cooling down and spent fuel pool cooling. The introduced distinction in phases, groups and options, as well as the indicated ranking of preferred options and succession of applied systems are maintained. Stock levels (minimum available water stocks in m^3) of the water supply systems are presented in Table 5.5 for all of the options.

As regards to the loss of primary UHS scenario, all the options using electrical systems connected to the ordinary electricity grid or one of the emergency power supply systems are not available anymore. The RL system remains available but only until its water reserves are exhausted The steam-turbine driven pump can be operated or secondary bleed & feed can be applied for about three hours , and the UJ system with its a diesel driven pump will work for about eight hours.

Cooling down

To cool down the reactor, initially only the limited options 1a/1b are available (in this situation 1a is identical to option 1b) and all the options when “external water“ (i.e. not water from the RS-stock) is supplied to the steam generator via the fire hose connection of RS (i.e. options 3a, 3b, 3c and 3d). These options are combined with secondary bleed & feed to options 3a, 3b, 3c and 3d respectively.

For options 3a, 3b and 3c the RL operation performs the cooling down for the period that the remaining parts of RL (steam driven pump and water supply) are available. This period will last for three hours due to the limited water supply from the RL tank. The next supply will start immediately via the fire hose connection of RS to the steam generator by UJ and/or the fire trucks and will last up to 90 hours and 120 hours respectively. The RS stock is not usable because the RS pumps do not operate. Water supplied by the public water supply system or from the River Westerschelde are available unlimited.

If feed via RL fails, then depressurising of the steam generators is an alternative although less time is available to accomplish the necessary connections (option 3b). This depressurisation (note that this causes the decrease of $T_{primary}$ at more then 100 K/h) lasts approx. one hour which isone third of the RL cooling options. In this situation, cooling continued by UJ or the fire truck supply will also be extended until approx. 90 hours and 132 hours respectively, again followed by the supply from the public water supply system and/or the River Westerschelde.

It has to be noticed that implementing the remaining options includes establishing of connections between the RS system and UJ , the CCB’s fire pond or the River Westerschelde , by the fire brigade which in a worst-case scenario takes less than one hour. This means that, in reality, only the application of the UJ system remains.

Decay heat removal

No options remain for the decay heat removal phase.

Combined cooling

When the UJ supply is applied to the combined cooling down of the reactor and spent fuel pool cooling (see Figure 5.15), the spent fuel pool cooling will be by evaporation after heating up the pool water and refill arranged through UF, as indicated in section 5.2.3. In this situation, the cooling can be achieved in 90 hours.

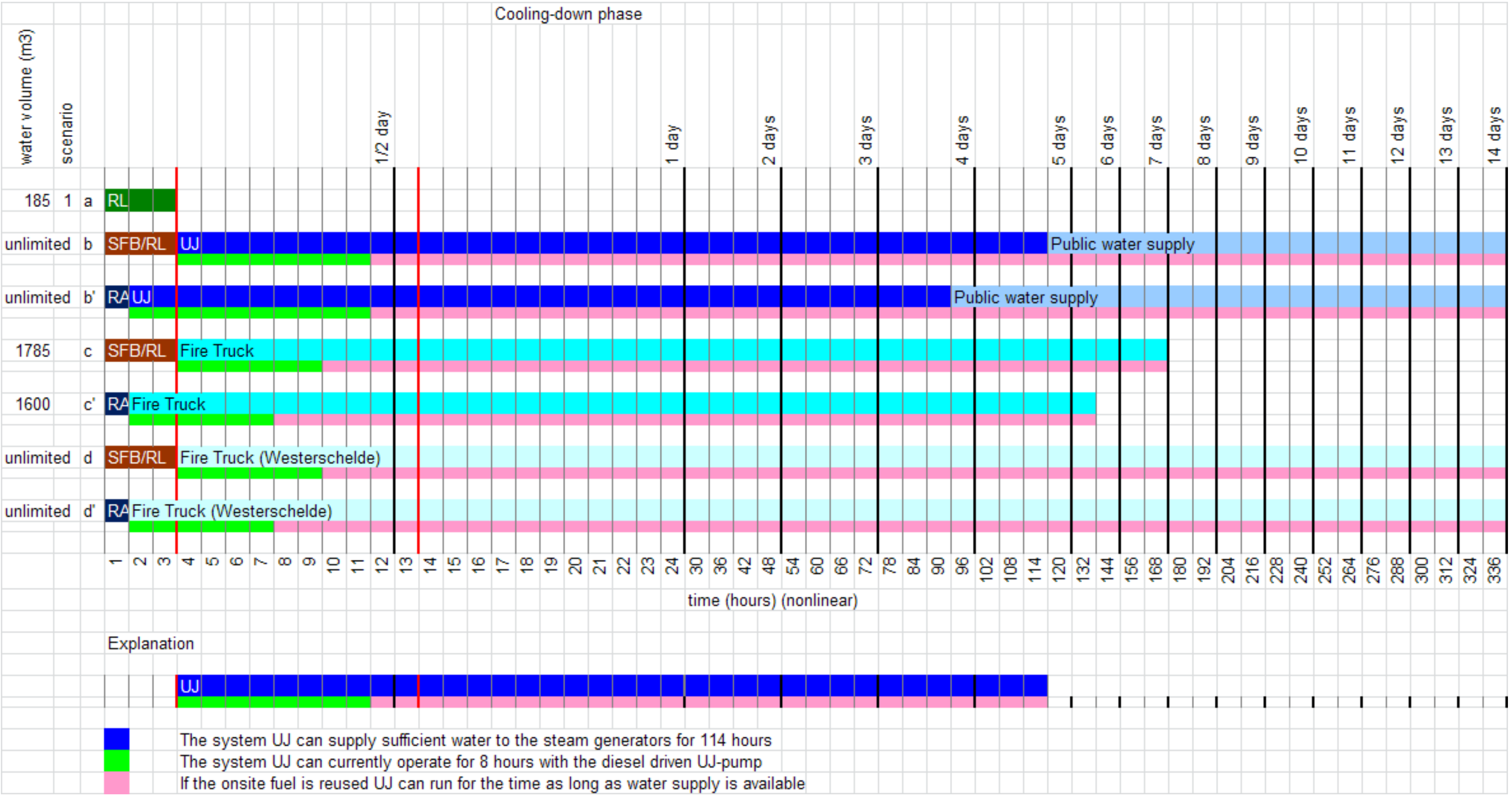

Figure 5.14 Cooling options for cooling down in the LPUHS-SBO 2 situation

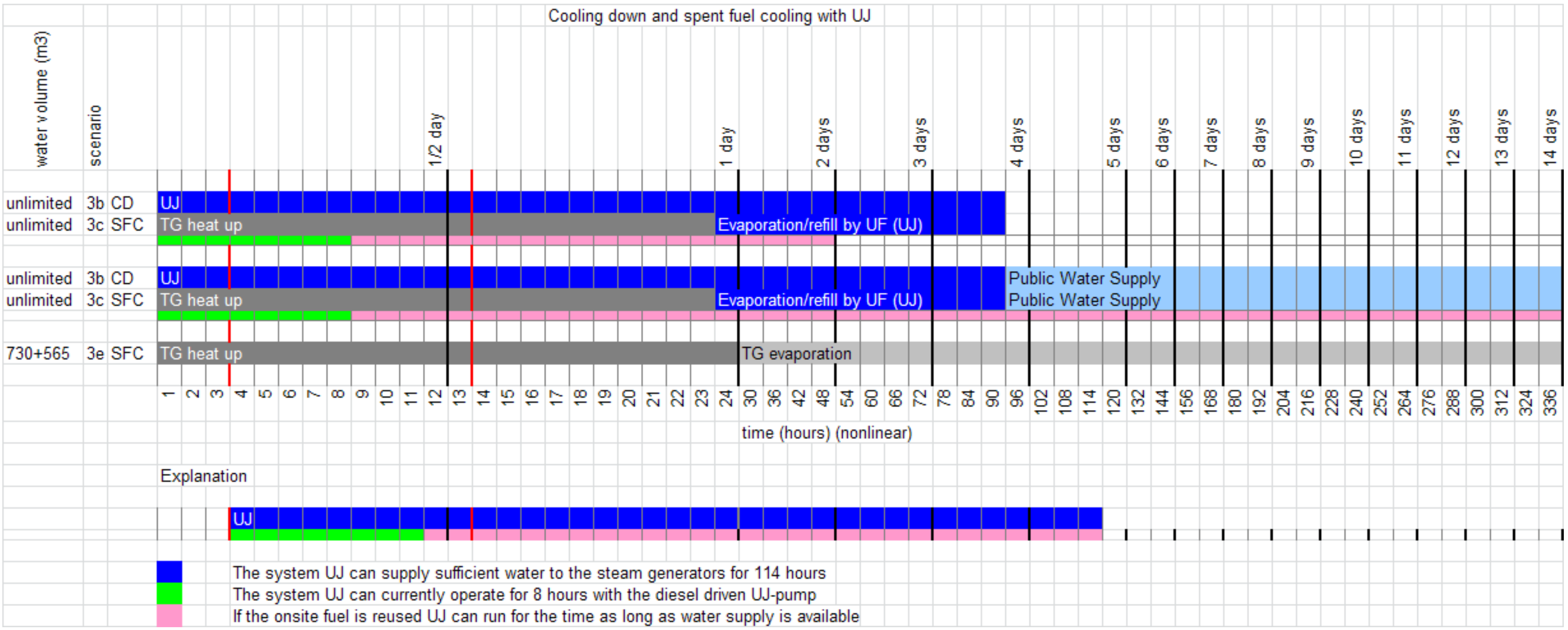

Figure 5.15 Time periods for UJ supply and alternatives for combined cooling down and spent fuel pool cooling in the LPUHS-SBO 2 situation

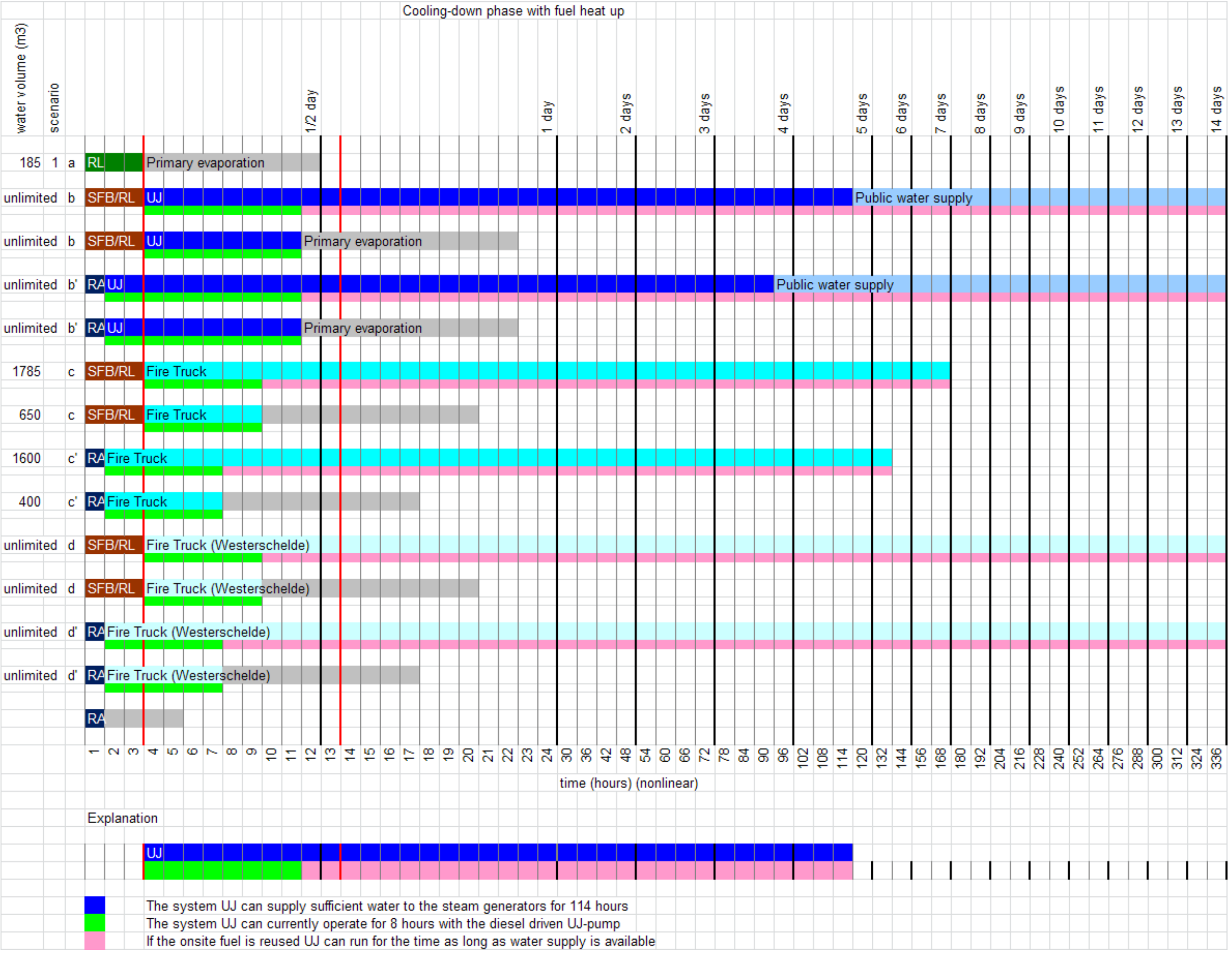

Figure 5.16 Time periods for water supply by solitary systems with and without diesel fuel replenishment

Overview of cooling status in case of loss of primary heat sink in combination with SBO 2

As regards the situation of loss of primary ultimate heat sink combined with complete station black out (SBO 2), no fuel damage will occur while the reactor is kept at hot shut down conditions; however, alternative options are available. Ultimately, the water is supplied by the public water system and/or tapped from the River Westerschelde. Table 5.11 lists the remaining options.

Operational state	System	Duration (approx. h)	Remarks
Cooling down	RL start of cooling down by supply of RL tank	3	Either the supply is provided by the steam driven RL pump or by secondary b&f
	Supply from UJ through RS Public water supply	11 (48)[56] unlimited	
	Depressurize steam generators Supply with fire truck (pond) Use of the Westerschelde	< 1 131 unlimited	
Decay heat removal	No option available		

Table 5.11 Cooling status in case of loss of primary heat sink in combination with SBO 2

Conclusion

This plant state (LPUHS-SBO 2) will, for the time being, achieve a hot shut down condition for the reactor.This situation can be maintained and with the reactor is being cooled by the secondary system so long as an external supply of water exists.

[56] The result in brackets is for the combination of cooling down and spent fuel pool cooling, including pool heating up

Electrical power supply

With the SBO 2 situation, the power is only supplied by the (DC) batteries and the (AC) uninterrupted power system.

The backup for the NS 2 diesel generators are the mobile diesel generator and an external (spare) diesel generator which is located near Rotterdam and thus needs transporting to the plant. It is concluded that this arrangement will take six (mobile diesel generator) to eight (external diesel generator) hours to complete.

Therefore, for the assessment, it is assumed that only the systems that have their own power source can be applied, which are:

- RL, which is driven by its own steam turbine driven pump or operated in the steam generator driven bleed & feed mode; the duration is approx. three hours due to loss of “feed” from steam (pressure);
- UJ, which is operated by its own diesel generator driven pump;
- the fire truck, which relies on its own engine.

The batteries and the uninterrupted power system have the capacity to supply power for at least 2.3 hours. Additional measures like disconnecting of systems are necessary for the following reasons:

- to operate the reactor protection system YZ;
- to control the reactorl;
- to operate valves like , for example, the RL valves or YP relief valves. The RA safety relief valves can be operated manually.

To avoid a loss of systems due to a lack of diesel fuel replenishment will start within:

- six hours for the fire truck;
- eight hours for the UJ system.

Depending on the possibilities of exchanging on-site diesel fuel stocks, the running times of both systems can be extended to weeks.

However it is noticed that no means, either hardware and software (procedures and/or appropriate contracts), exist to deal with these activities. For external supplies, there are contracts at hand for delivery of diesel fuel within certain time frames. There are arrangements to assure delivery of diesel in such an event .

Time period for cooling

So long as either the public water supply system or the tapped water from the River Westerschelde are available the water supply will not be limited. Given that water supplies by external means are arranged, cooling can be provided for during the entire course of the event. In this situation, the hot shut down of the reactor and the cooling of the spent fuel pool by evaporation has to be accepted.

The limiting facts for this situation are:

- limited capacity of batteries (at least 2.3 hours);
- limited basic (own) diesel supply mainly for pumps that operate as stand alone (UJ for eight hours; fire truck for six hours);
- the short time taken for the fire brigade to provide connections, (one hour in a worst-case scenario);
- operability of the relevant valves (either battery supply or manually).

Nevertheless, during an LPUHS-SBO 2 situation it is not preferable to keep the reactor in a hot shut down condition. A long hot shut down situation will result in fatigue of the surgeline of the volume control system TA.

5.1.3.2 External actions foreseen to prevent fuel degradation.

External actions foreseen to prevent fuel degradation in case of loss of primary heat sink in combination with SBO 1

Emergency diesel reserves for the emergency power supply system NS 2 are designed to be available for at least 72 hours, but could be extended to 1,300 hours while exhausting all stocks. After this period, new diesel supplies should be brought to the plant from external suppliers. With external support, the mobile diesel generator can also be put in place to start operation to backup NS 2.

New diesel fuel supplies should be brought to the plant. It would take a few hours to fill up the diesel tank trucks, drive to the power station, and refill the on-site diesel tanks. Contracts for delivering of diesel fuel exist.

Several hours would be needed to put the mobile diesel generator in place and connect it.

External actions foreseen to prevent fuel degradation in case of loss of primary heat sink in combination with SBO 2

The UJ water reserves be exhausted if the public water system (Delta) is not available. Recovery of the public water system is not within the control of EPZ but this system does have its own emergency power system that is not nearby.

Taking suction from the River Westerschelde would be possible as the equipment is already available; however a procedure is lacking.

If the mobile diesel generator EY080, is not available by default, an external diesel is held ready. This is situated near Rotterdam and can be transported and installed within 8 hours. A procedure to conclude and then order, externally, this diesel generator is at hand , which the supplier is contracted to deliver within four hours.

In the SBO 2 situation the UJ diesel pump will need refuelling within eight hours, while the fire trucks need refuelling within approx. six hours. According to the amount of diesel available on site (approx. 240 m^3) there is enough to extend the running times for all the above mentioned users for several weeks. However no procedure is in place to removethe diesel from one system and transfer it to another.

In addition to this, there is only a small crew on site to deploy the companys' own fire brigade; immediate enlargement using off-site volunteers may be required.

External support will also be necessary when water has to be pumped from the CCB fire fighting pond or from the River Westerschelde to the plant,. This will be realised in cooperation with the umbrella organisation "Veiligheidsregio Zeeland".

The establishment of a suction line to the River Westerschelde has not been tested yet, but the duration is estimated to be several hours.

Transportation of the mobile diesel generator will take several hours, time to connect this generator to the NS 2 system is indicated to be four hours.

New diesel supplies should be brought to the plant. It would take a few hours to fill up diesel tank trucks, drive to the power station, and refill the on-site diesel tanks. However it is assumed that this can be performed within the indicated time frame.

Deployment of the fire brigade within the specified one hour is supposed not to be a problem for the small crew on-site, whichmight be able to carry out the indicated UJ-RS connecting activities. Additional and external support may be subject to some delay.

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

5.1.3.3.1 Loss of ultimate heat sink, combined with SBO 1

Potential cliff-edge effects

The following potential cliff-edges have been identified for this scenario when mitigated by the main options for cooling down and decay heat removal:

- failure of the relief valves of the Main steam system RA;
- failure to cool the steam turbine driven RL pump;
- non availability of the steam turbine driven RL pump;
- failure to apply RS after RL application;
- failure of more than two pumps of the Backup cooling water system VE;
- failure of the heat exchanger of the Backup residual heat removal system TE;
- failure of UJ water supply by:
    - failure to establish connections (i.e. deployment of fire brigade);
    - premature exhaustion of the water reserves of the Low pressure fire extinguishing system UJ. This could be because of it being used for other reasons at the same time, like extinguishing fires.

Taking the alternatives into account, then additional potential cliff-edges can then be identified:

- failure of water supply by:
    - failure to establish connections to replenish basic stocks (action of operator or of fire brigade);
    - damaged fire hose connection on the connecting duct between the two redundancies of the RS system;
    - exhaustion of the water reserves of the Low pressure fire extinguishing system UJ. This could because of it being used for other reasons at the same time, like extinguishing fires.

Additionally, the potential cliff-edge effects identified for the SBO 1 (see section5.1.1.3) also apply here:

- failure of supply by all remaining back up emergency power systems namely NS 2, CCB emergency power system and EY080 results in SBO 2;
- running out of diesel supplies results in SBO 2;
- failure of CCB supply, consolidaing the SBO 1 situation;
- failure of action (failure to replenish the diesel fuel supply).

Of these, only the failure of action remains because the other items deteriorate into SBO 2. This plant condition is dealt with in section 5.1.1.3.

Potential actions to prevent cliff-edge effects

Potential cliff-edges as presented above are basically derived from section 5.1.2.5. Therefore, potential actions to prevent these cliff-edges are the same as those presented in that section, related to the cliff-edges under consideration.

The additional ones are:

Failure to cool the steam turbine driven RL pump

With the loss of VG/VF seal cooling of this pump, the UK system will take over this cooling. The gravity driven supply by this system will last approx 18 hours which exceeds by far the period of the RL water supply. In case the cooling fails or the RL water supply is exhausted, a switch over to RS cooling is required.

Non availability of the steam turbine driven RL pump

According to the Technical Specifications, only two out of three emergency feed water (RL) pumps need to be available. This means there is a possibility that the steam turbine driven RL pump is not available due to (allowed) maintenance. An adjustment of the Technical Specifications e.g. no maintenance of that pump during reactor operation might be necessary or other measures shall be taken to prevent this situation like an autonomous mobile high pressure pump for flexible use (e.g. to backup the steam turbine driven RL pump).

Failure to apply RS after RL application

In this situation the alternative options of supply by UJ or fire truck apply.

Potential actions to increase robustness of the installation

Potential actions to increase the robustness of the installation for the LPUHS-SBO 1 scenario are identical to the actions listed in section 5.1.2.5.

Regarding SBO 1, the potential actions are defined in section 5.1.1.5

5.1.3.3.2 Loss of ultimate heat sink, combined with SBO 2

Potential cliff-edge effects

The following potential cliff-edges have been identified for this scenario when using the options for cooling down and spent fuel pool cooling:

- failure of the relief valves of the main steam system RA (availability of relevant valves);
- failure to cool the steam turbine driven RL pump;
- non availability of the steam turbine driven RL pump;
- failure of UJ water supply by:
    - failure to establish connections (i.e. deployment of fire brigade);
    - premature exhaustion of the water reserves of the Low pressure fire extinguishing system UJ. This could be because of it being used for other reasons at the same time, like extinguishing fires,
- failure of water supply by:
    - failure to establish the connections to replenish basic stocks (action of operator or of fire brigade);
    - damaged fire hose connection on the connecting duct between the two redundancies of the RS system;
- running out of on-site diesel supply for the diesel driven UJ pump and for the fire trucks.

Additionally, the cliff-edge effects identified for SBO 2 (see section 5.1.1.5) also apply here: "In conclusion, the cliff-edges for the SBO 2 situation are failure to deliver the external (spare) diesel generator and/or its diesel fuel supply and a failure of the UPS".

Potential actions to prevent cliff-edge effects

Potential actions to prevent the cliff-edge effects of the LPUHS, the LPAUHS and the LPUHS SBO 1 scenarios are described in section 5.1.2.5 and above. The potential actions regarding the cliff-edge effects of SBO 2 are described in section 5.1.1.5.

A potential action to prevent running out of the on-site diesel supply for UJ and for the fire brigade can be an increased number of diesel supplies at separate locations on the plant site.

Measures which can be envisaged to increase the robustness of the installation

The potential actions to increase robustness of the installation for the LPUHS, the LPAUHS and the LPUHS-SBO 1 scenarios are described in section 5.1.2.5 and above.

The potential actions regarding the cliff-edge effects of SBO 2 are described in section 5.1.1.5.

5.2 Spent fuel storage pools

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 electric power

Measures to increase the robustness of the plant in case of loss of electric power are described in 5.1.1.5

5.2.2 Loss of the ultimate heat sink

The various options for cooling of the spent fuel pool are described below.

Loss of primary ultimate heat sink (LPUHS)

General

In case the regular method of spent fuel storage pool cooling over the VF system is not available, the preferred order of options is:

1. apply VF by the alternative application of the VF system (ducts) as they are already applied for the decay heat removal, then
2. use VE over the reserve heat exchanger TG080. Water supplies for VF and VE are identical to those for the decay heat removal.

As with the cooling down phase and the decay heat removal phase, the grouping and sub divisions for the main option and its alternatives are also applied for spent fuel pool cooling.

For the grouping the following distinction is made:

1. TG/TF/VF
2. TG080/VE
3. Evaporation.

For the situation of cooling by evaporation, water will be supplied directly into the pool; it is indicated that a supply of clear, non-borated, water is allowed.

The sub division depends on the supporting systems and their possible combinations in relation to the system that establishes the basis of the group, for example.:

- Group1: TG/TF/VF
    - UJ:
    - Firetruck;
    - Westerschelde
- Group 2: TG080/VE:
    - VE wells,
    - UJ,
    - Fire truck,
    - Westerschelde.
- Group 3: Evaporation
    - TJ,
    - TN,
    - UF / free filling,
    - TJ/TN/UF/ free filling,
    - TG.

These combinations are elaborated below. Figure 5.17 shows a complete listing of this grouping and sub division. It is emphasised that the options that include UJ as a (additional) supply are preferable to the fire truck options.

Group1: TG/TF/VF

*Option 1a: **TG/TF /VF/UJ (main option**)*

The main option to provide spent fuel pool cooling is the application of the cooling chain that includes the *spent fuel pool cooling system TG / the component cooling water system TF /conventional emergency cooling water system VF with water supply by the low pressure fire extinguishing system UJ*.

By re-establishing the VF function through the UJ supply in option 1 for decay heat removal, the regular cooling chain of the fuel storage pool will also be re-established. Again this option requires personnel to establish fire hoses to connect UJ with VF. The UJ system is replenished by the public water supply system (Delta).

Option 1b: **TG/TF/VF/fire truck**

Water supply to the conventional emergency cooling water system VF from a fire truck that takes suction from the fire fighting pond of CCB.

VF is equipped with connections for a fire truck. An instruction for this is available . This is another way, identical to option 1a, to re-establish the regular cooling chain of the fuel storage pool.

Option 1c: **TG/TF/VF/Westerschelde**

Water supply to the conventional emergency cooling water system VF from a fire truck that takes water suction from the River Westerschelde

Water supplied from the fire fighting pond at CCB, option 1b, will be replaced by the supply from the River Westerschelde.

Group 2: TG080/VE

Option 2a: **TG080/VE**

Spen tfuel pool cooling system TG with reserve heat exchanger TG080 / backup cooling water system VE.

As the regular heat exchangers of the TG system are ultimately cooled by the conventional emergency cooling water system VF which lost its heat sink in this scenario, the reserve heat exchanger TG080 will take over. This heat exchanger is cooled by the backup cooling water system VE . It is noted that in case VE is also applied for decay heat removal of the primary system the performance of these two combined cooling functions must be optimised.

Option 2b: **TG080/VE/UJ**

Spentfuel pool cooling system TG with reserve heat exchanger TG080 / backup cooling water system VE, supplemented by the low pressure fire extinguishing system UJ.

In case the VE system loses its heat sink (the deep water wells or pumps), the UJ system can take over. UJ is connected to the VE system and has a water reservoir with a volume of 1,200 m^3 . The UJ system is replenished by the public water supply system (Delta).

Option 2c: **TG080/VE/fire truck**

Water supply to the backup cooling water system VE from a fire truck, that takes suction from the fire fighting pond at CCB.

In the backup systems bunker, a connection on the VE system with a fire hose is available. In this way the cooling chain spent fuel pool cooling system with reserve heat exchanger TG080 / backup cooling water system VE can stay intact.

Option 2d: **TG080/VE/Westerschelde**

Water supply to the backup cooling water system VE from a fire truck that takes suction from the Westerschelde

The water supply from the fire fighting pond at CCB, option 2c, will be replaced by the supply from the River Westerschelde.

Group 3: Evaporation

Option 3a: **Evaporation/TJ**

Replenishment of water of the spent fuel pool by TJ supply

In case the spent fuel cooling method of heating up the water and finally evaporating of the water in the pool, more water can be supplied to the pool by injection from the TJ tanks. The refilling procedure is at hand.

Option 3b: **Evaporation/TN**

Replenishment of water in the spent fuel pool by an alternative supply from TN

By connecting the nuclear fuel storage pool cooling system TG and the water supply system TN (TN10-TN20) with fire hoses (available on site) a supply by TN can be provided to refill the pool in case water is lost due to evaporation. There is no procedure available for this method.

Option 3c: **Evaporation/UF/free filling**

Filling of the spent fuel pool by draining the water directly into the pool through the high pressure fire extinguishing system UF

Using UF to drain water into the pool is possible. However, for this option a fire truck taking suction from UJ or ultimately from the fire fighting pond at CCB and /or the River Westerschelde shall be performed. Fire hoses can be connected to the UF system, but remote control is not possible, while the containment shall be open.

The UJ system is replenished by the public water supply system (Delta).

There is no procedure available for this method.

Option 3d: **Evaporation/TJ/TN/UF/free filling**

Replenishment of the spent fuel pool by all means

By successive application of the preceding option (in the same indicated order) the cooling of the spent fuel pool by this combination of on-site supplies can be extended.

Option 3e: **Evaporation/TG**

No supply to the spent fuel pool

The only possibility here is cooling by heating up and evaporating of the water in the spent fuel pool. This option is limited because no refilling takes place.

Spent fuel pool cooling

For the assessment of the time period for spent fuel pool cooling, it is assumed that heat removal will be performed at first by water cooling and secondly by heating up and evaporating the pool water, which will be replenished.

Furthermore, for spent fuel pool cooling, it is assumed that the core is completely unloaded, so stored in the pool. This means that according to TE/VE design requirements heat production starts at decay heat level of 5.14 MW for a full core (see Annex 5.2).

The spent fuel pool cooling phase is an identical situation to that when for decay heat removal occurs. This means that the first supply, provided by UJ can last for six hours (on-site stock) when only pool cooling is provided (seeFigure 5.17).

Replenishment by the public water supply system (Delta) basically provides unlimited cooling (option 1a’). An aAlternative water supply can be provided for limited time periods from the fire fighting pond at CCB (option 1b) and an unlimited supply by the River Westerschelde (option 1c), although it will probably be easier to switch to the next (most probable) option, which is TG080/VE cooling.

In case neither of these possibilities (via TJ/TF/VF and TG080/VE) are available, heat removal by first heating up and then evaporating of the pool water is taken into account. The resulting time period is based on evaporation until the water level drops to top of the stored fuel elements (indicated as TG evaporation). Figure 5.17 shows this heating up and evaporation and the supply from the public water supply system in parallel for option 1a and 1a’ and 2b and 2b’, while for group 3 only the heating up and evaporation of pool water is accounted for by the amount of water supplied by the indicated system to refill the pool. From group 3 it can be concluded that a combination of the first three options in option 3d will provide a time period of approx. 14 days before the top of the fuel elements is reached. In the situation where no water can be drained to the pool it will take six hours to heating up the pool and another 78 hours of evaporation before the top of the spent fuel elements is reached. After this, rapid heat up and damage to the fuel will occur.

Combined cooling

Starting with the normal reactor operation situation it might be feasable for one system to provide cooling to the reactor and to the spent fuel pool at the same time.

Compared to the fuel pool assessment for this situation above, the following is assumed (see Annex 5.2):

- the number of elements stored is 1/3 of a full core;
- storage time to be accounted for is 14 days after shut down (average refuelling period);
- heat removal will be performed for the following:
    - cooling down by evaporation;
    - decay heat removal by water cooling;
    - spent fuel pool cooling either by water cooling or by heating up and evaporation (or these options in).

Cooling down + spent fuel pool cooling

For details of this situation, please refer to Figure 5.4. It is noted from this figure that when applying UJ (on-site) supply for both cooling down and spent fuel pool cooling, the supply will last nine hours. When the supply is provided by the fire fighting pond at CCB, the duration of the water supply will extend a further two hours when compared to the UJ supply. The supply by the public water supply system (Delta) and the River Westerschelde are unlimited.

Decay heat removal + spent fuel pool cooling

For details of this situation, please refer to Figure 5.5. This figure shows that when applying UJ (on site) supply for both decay heat removal and spent fuel pool cooling, the supply will last approx. six hours or 13 hours after shut down where decay heat removal starts three hours or 13 hours after shut down respectively. This means there is no decay heat removal by UJ while the pool starts heating up. A continuous supply from the public water supply system is considered insufficient because its capacity is too small for cooling water flow. It is the same with supplies via the fire truck where the supply capacity is also too small. This means preference should be given to the combinations presented in the second part of Figure 5.5 which shows that decay heat removal by UJ or fire truck and the capacity of the public water supply system and the fire truck are adequate for heat removal in these situations

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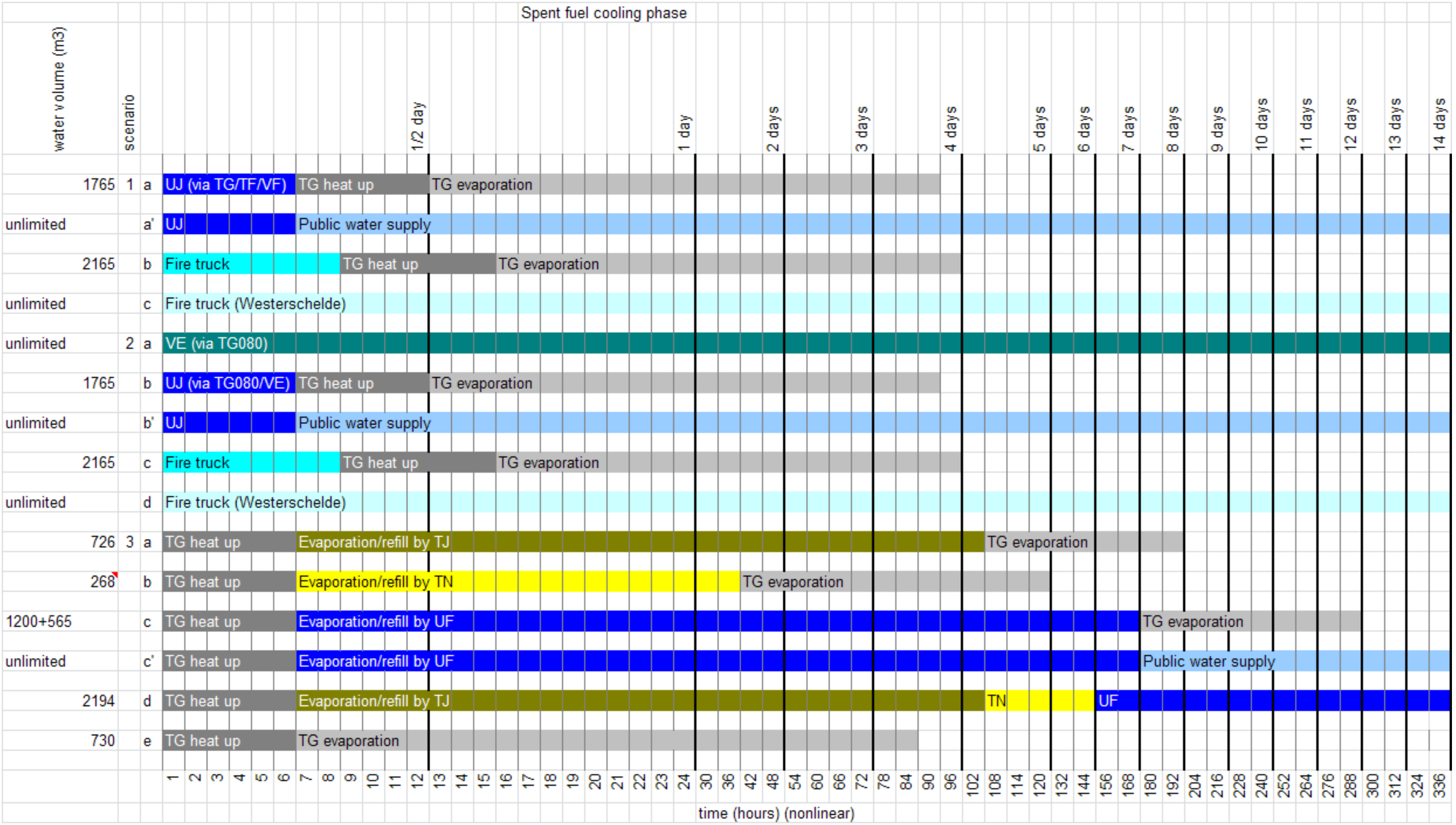

Figure 5.17 Cooling options for spent fuel pool cooling in the LPUHS situation

Loss of primary and alternate ultimate heat sink (LPAUHS)

As with the decay heat removal phase, the option to use VE as individual link, fed by the deep water wells (option 2a), is not available here. The remaining available options, presented above areas followes:

Group1: TG/TF/VF

Option 1a**: TG/TF/VF/UJ**; ultimate supplied by the public water supply system (Delta)

Option 1b: **TG/TF/VF/fire truck**

Option 1c: **TG/TF/VF/Westerschelde.**

Group 2: TG080/VE

Option 2b: **TG080/VE/UJ**; ultimate supplied by the public water supply system (Delta)

Option 2c: **TG080/VE/fire truck**

Option 2d: **TG080/VE/Westerschelde.**

Group 3: Evaporation

Option 3a: **Evaporation/TJ**

Option 3b: **Evaporation/TN**

Option 3c: **Evaporation/UF/free filling**; ultimate supplied by the public water supply system (Delta)

Option 3d: **Evaporation/TJ/TN/UF/free filling**

Option 3e: **Evaporation/TG.**

Spent fuel pool cooling

With the spent fuel pool cooling phase an identical situation to decay heat removal occurs. In other words, first supply to VF or VE is provided by UJ, by its ”own” on-site stock, this can last six hours when only pool cooling is provided (see Figure 5.18). Replenishing UJ from the public water supply system (Delta) will extend this cooling for the course of the event (unlimited). Alternative available supplies are the fire fighting pond and/or the River Westerschelde. In case these alternatives are not available, heat removal by evaporation of the pool water is taken into account. The resulting time periods, based on evaporation until the water level drops to top of the stored fuel elements, are identical to the periods for the LPUHS situation.

Combined cooling

When the UJ supply is applied to the combined cooling down and spent fuel pool cooling and the decay heat removal and spent fuel pool cooling (see Figure 5.8 and Figure 5.9), the situation is identical to the LPUHS situation.

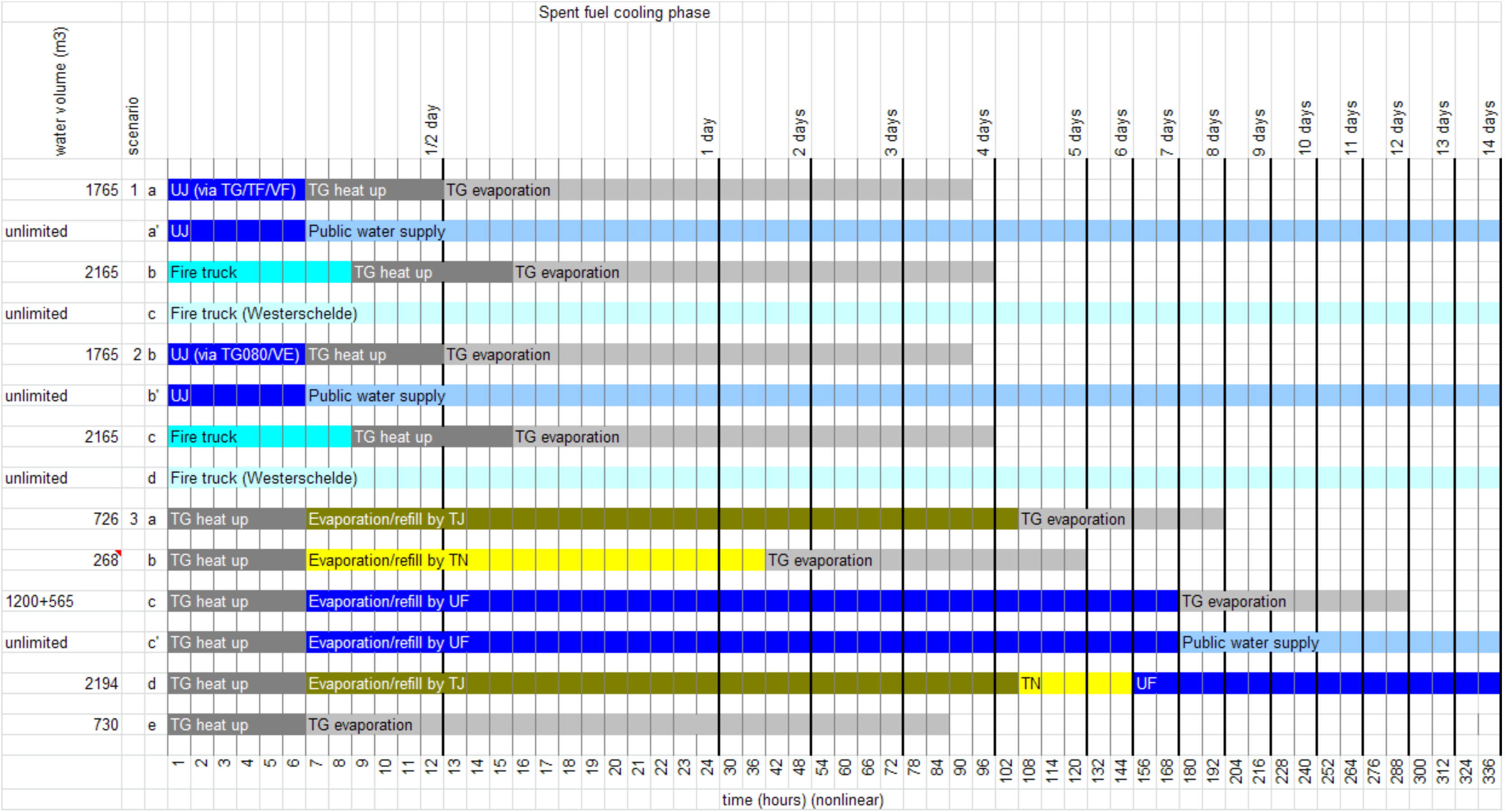

Figure 5.18 Cooling options for spent fuel pool cooling in the LPAUHS situation

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

The following measures can be envisaged to increase robustness of the plant in case of loss of ultimate heat sink:

- 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 prolongued 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 (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.

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)

The various options for cooling of the spent fuel pool are described below.

Loss of primary ultimate heat sink with SBO 1 (UHS-SBO 1)

As with the decay heat removal phase, group 1 declines because the component cooling water system TF, which is the intermediate between TG and VF loses electrical power. The TG08/VE combinations in group 2 remain available and these are:

Group 2: TG080/VE

Option 2a: **TG080/VE**

Option 2b: **TG080/VE/UJ**; ultimately supplied by the public water supply system

Option 2c: **TG080/VE/fire truck**

Option 2d: **TG080/VE/Westerschelde**

Group 3: Evaporation

Option 3c: **Evaporation/UF/free filling**; ultimately supplied by the public water supply system.

UJ operates on CCB emergency power or its own diesel driven pump while a fire truck establishes the connection between UJ and UF; note that the containment is open.

Option 3e: **Evaporation/TG**.

If no pool cooling system is operating, the water in the spent fuel pool heats up to the boiling temperature. Next, the water will start evaporating, which decreases the water level in the spent fuel pool. This causes, the water level above the spent fuel to decrease as well, which results in increasing the radiation levels in the containment.

Spent fuel pool cooling

With regard to the spent fuel pool cooling phase (seeFigure 5.19) an identical situation to the decay heat removal occurs. A direct switch over to TG080/VE cooling occurs butthe water supply is also available here during the course of the event. An additional supply is accessible in case of the deep water wells are not available.

This means that the first supply is provided by UJ or the fire truck and can last for six hours and eight hours respectively when only pool cooling is provided. Furthermore, replenishments from the public water supply system and /or the Westerschelde will be provided during the remaining course of the event. Ultimately the heating up and evaporation option can be applied, because refilling the pool by the UJ stock via UF (option 3c) will last for approx six days. This resulting time period is based on evaporation until the water level drops to the top of the stored fuel elements.

Combined cooling

When the UJ supply is applied to the combined cooling down and spent fuel pool cooling and for the combined decay heat removal and spent fuel pool cooling (see Figure 5.12 and Figure 5.13), cooling down situation is identical to the LPUHS situation. As regards the decay heat removal the preferred options of the LPUHS situation (see section 5.1.2.2.2) remain

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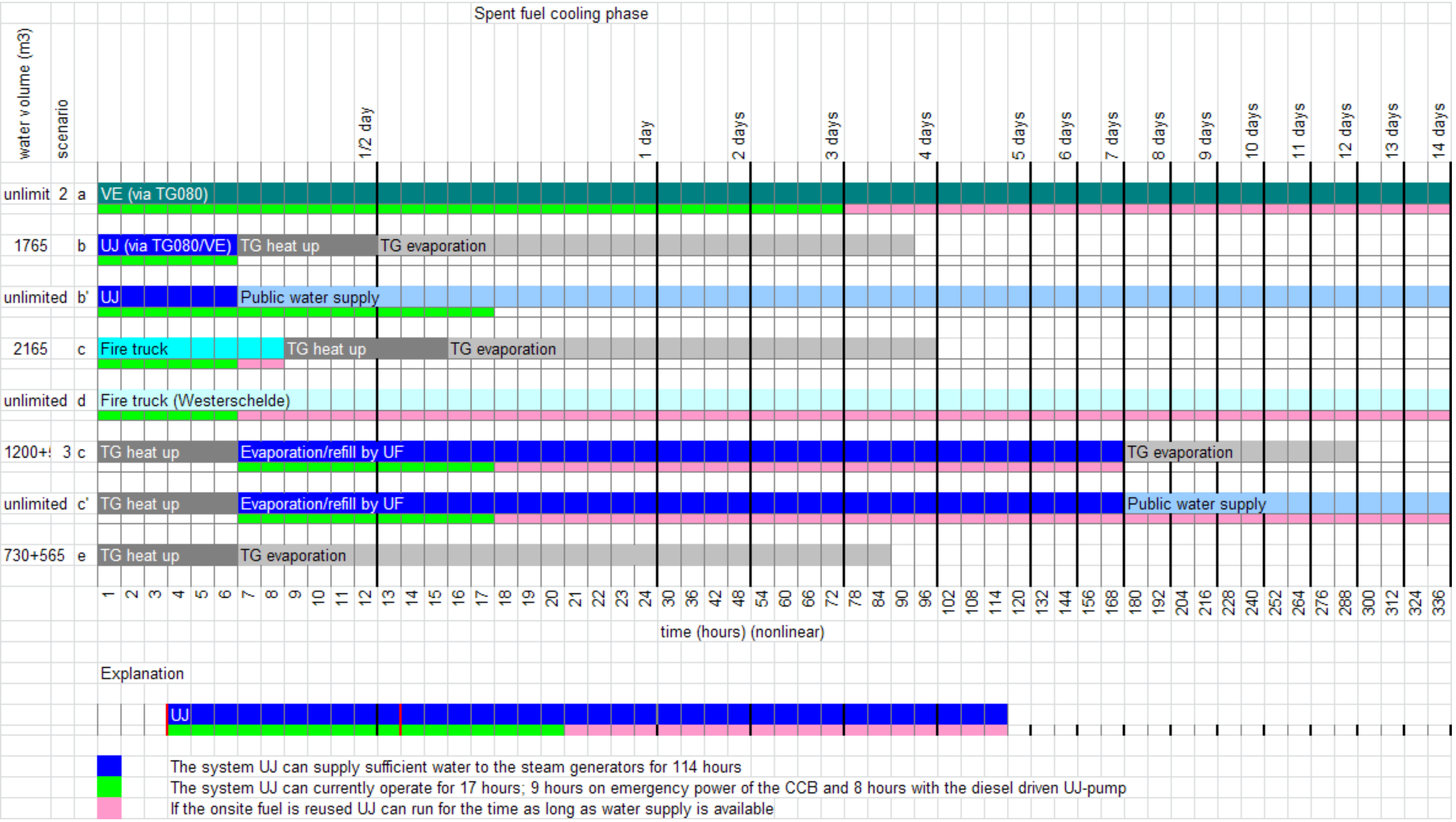


Figure 5.19 Cooling options for spent fuel pool cooling in the UHS-SBO 1 situation

Loss of primary ultimate heat sink with SBO 2 (UHS-SBO 2)

As with the decay heat removal phase, the normal chain TG/TF/VG and emergency chain TG080/VE are not available for fuel pool cooling. Therefore only the evaporation options remain.

Group 3: Evaporation

Option 3c: **Evaporation/UF/free filling**

From the high pressure fire extinguishing system UF water can be tapped to drain directly into the pool. However it needs to have a fire truck and UJ available and the provisions to install flexible (fire hoses) connections between UJ, the fire truck, UF and the pool. The equipment (preferably remotely controlled) and the procedures to perform these actions are not available.

Option 3c’: **Evaporation/UF/free filling**; ultimate supplied by the public water supply system.

The public water supply system will ultimately feed option 3c when UJ stock is exhausted

Option 3e: **Evaporation/TG**.

If there is no pool cooling system operating, the water in the spent fuel pool heats up to the boiling temperature. The water will then start evaporating, which results in a lowering water level in the spent fuel pool. This causes the water level above the spent fuel to drop as well, which results in increasing the radiation levels in the containment.

Spent fuel pool cooling

With regard to the spent fuel pool cooling, only the evaporation options 3c/c’ and 3e apply (see Figure 5.20). Evaporation wilst applying draining from UF is applied (3c), can be maintained during the course of the event providing the following is in place::

- UJ connection and feed to UF by the fire truck is established via an open backup systems bunker. Currently no features for this are available;
- supply from the public water supply system (Delta) is ensured.

Combined cooling

When the UJ supply is applied to the combined cooling down of the reactor and spent fuel pool cooling (see Figure 5.15), the spent fuel pool cooling will be by evaporation after the pool water has been heated up. The refill has to be arranged through UF, as indicated above. In this situation, cooling can continue for 90 hours.

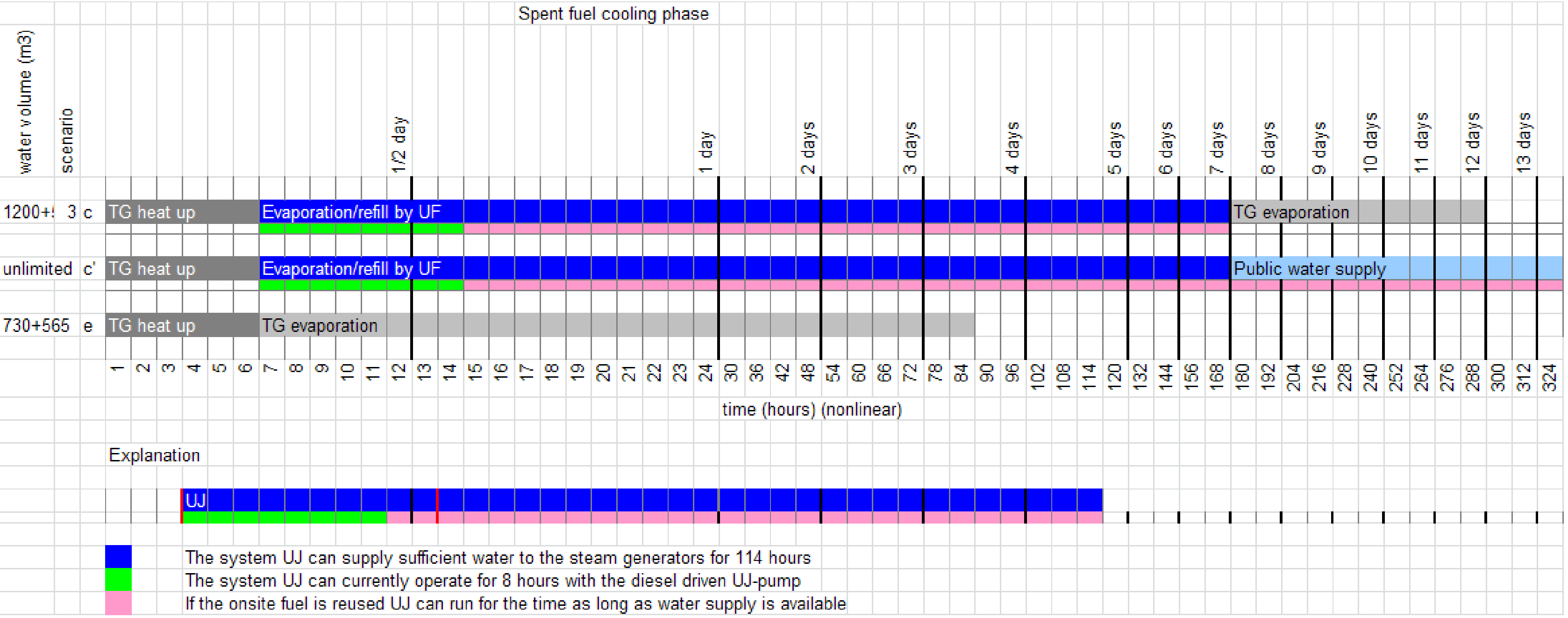


Figure 5.20 Cooling options for spent fuel pool cooling in the UHS-SBO 2 situation

As regards the four determined plant states of loss of the UHS and loss of the UHS combined with SBO the options for cooling of the spent fuel pool presented in sections 5.2.2 and 5.2.3 are listed in Table 5.12. However, alternative options are available. Table 5.12 lists the preferred sequence and the most obvious alternatives.

For the combinations LPUHS, LPAUHS and LPUHS-SBO 1, no fuel damage will occur because by applying the options presented in Table 5.12, these situations are under control. Regarding the combination LPUHS-SBO 2, no fuel damage will occur here either while the spent fuel pool is cooled by evaporating pool water. Ultimately, the water is supplied by the public water system and/or tapped water from the Westerschelde (external action).

Plant state	Means of cooling	Duration (approx. h)	Remarks
Loss of primary ultimate heat sink (LPUHS)	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	Remind that UJ stock is limited
	Switch over to TG080/VE cooling	unlimited	
Loss of primary and alternate heat sink (LPAUHS)	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	Remind that UJ stock is limited
	Supply from the fire fighting pond at CCB and the River Westerschelde	8 (0, 14)[57] unlimited	
Loss of primary ultimate heat sink with station black-out type 1 (LPUHS-SBO 1)	TG080/VE combination provides sufficient cooling	Basically unlimited	
	UJ Public water supply system	6 (13)[58] 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 Public water supply system	180 (48) unlimited	
	Heat up and evaporation of pool water until top of fuel reached	84	

Table 5.12 Cooling status for the spent fuel pool cooling for the four determined plant states of loss of the UHS and loss of the UHS combined with SBO

[57] The results for combined decay heat removal and spent fuel pool cooling are presented in brackets. The variant results from the differences in pool loading and start ans expiring moment of the several activities

[58] 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

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

Potential cliff-edge effects

- failure to drain water from UJ via the fire truck and UF to the spent fuel pool.

Potential actions to prevent cliff-edge effects

To refill the spent fuel pool from the UF system the water supply should be ensured. UF is supplied by UJ which has a diesel driven pump. The high pressure pumps of UF do not operate in SBO 2. Therefore these pumps should be bypassed. To achieve this, both hard-ware (equipment) and soft-ware (procedures) should be established.

Potential actions to increase robustness of the installation

The following measures can be envisaged to increase robustness of the plant in case of loss of ultimate heat sink:

- 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;
- 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.

Annex 5.1. NBP-ECA-0-0, Actions in case of loss of power BU/BV

This procedure is entered when no voltage is available on both of the 6 kV auxiliary power busses (BU & BV). This implies the loss of external power combined with the loss of the emergency power diesel generators.

The procedure moves through three stages:

1. checks to be performed immediately;
actions to restore emergency power;
actions to be performed when emergency power is not restored.

1. Checks to be performed immediately

- Reactor scram (RESA). If this has not happened give the manual reactor shutdown command. If subcriticality is not achieved, **FHP-S-1** is initiated;
- Turbine stop (TUSA). Alternatives to stop the turbine can be found in FHP-S-1.

2. Actions to restore emergency power

These actions are from procedure **S-EY-01.**

The options to restore power to one of the emergency buses directly:

- connect the emergency power bus to the (active) main power bus;
- start one of the emergency diesel generators (EY010 to 030).

If not successful, test for both main power busses in turn (BA & BB) as to whether the bus has no overcurrent protection activated or the overcurrent protection of a branch of the bus is activated and that branch can be isolated. If so:

- try to connect the main power bus to the external 150 kV grid power;
- try to connect the main power bus to an active power bus of the neighbouring coal plant.

If successful, connect the activated main power bus to an auxiliary power bus.

If power is restored to either emergency power bus, NBP-ECA-0-0 exits to the other relevant procedures.

3. Actions to be performed when emergency power is not restored

- check the 400V power system (CW & CX). The 400 V system powers the systems used in the next steps. Options to restore power from procedure **S-EY-02** is to be attempted for both 400 V busses in turn by doing the following:
    - start diesel generator (EY040 & 050);
    - connect to the 6 kV system of the neighbouring coal plant;
    - connect to the emergency power bus of KCB;
    - connect to the external mobile power generator.
- ensure cooling of the power systems by using the secondary bunkered water supply system RS. The bunkered water pools RS need eventually to be cooled through either:
    - fire water supply system UJ;
    - backup cooling water from ground water pumps VE.
- maintain sufficient water level in at least one steam generator. The following measures are available:
    - turbine-driven auxiliary feedwater pump (RL023);
    - bunkered feedwater system RS. Bunkered feedwater tanks will be kept filled by the fire water supply system UJ or by tanker trucks.
- ensure primary volume control through the primary backup coolant makeup system TW. Bunkered injection water needs to be resupplied;
- isolate the primary system. All isolation valves of the primary system need to be checked and closed;
- Start the spent fuel pool cooling system TG.

If at this point the core exit temperature is high and rising, the pressurizer tandem safety valves are opened and the procedure exits to **SACRG-1** (Severe Accident Control Room Guideline 1).

If not, extra measures are to be contemplated and attempts to restore emergency power have to continue.

Annex 5.2. Assessment of heat to be removed

This appendix lists assumptions, boundary conditions and results of calculations to determine the amounts of water necessary to remove the residual and decay heat of the fuel in the reactor core and/or the spent fuel pool. Based on the assumptions and boundary conditions, the resulting calculations are presented in graphs.

The heat to be removed from the primary system is split into decay heat and residual heat in the primary system.

Decay heat

100 % power

100 % decay heat and 0 sigma

Core with 4.45% ^{235}U: licensing calculation in NGPS8/2003/de/0095 Rev. A

Core with MOX: licensing calculation in NESS-G/2008/de/0088 Rev. A

For decay heat, the maximum value of the above mentioned cores is taken at a specific time.

Therefore the used decay heat envelopes a core with 4.45% ^{235}U as well as with MOX.

Heat in the primary system

Sources of heat are :

- heat in the primary water;
- heat in the primary components (steel);
- heat from the main cooling water pumps.

Heat in primary water

Initial conditions: 307 °C and 155 bar

Cool down condition: 120 °C and 13 bar

Heat in primary water = 98.650 MJ

Heat in the primary components (steel)

Initial conditions: 307 °C

Cool down condition: 120 °C

Heat in primary components = 120.428 MJ

Heat from the main cooling water pumps

Continued operation is assumed for 0.5 hour

Heat from the main cooling water pumps = 15.051 MJ

The total heat in the primary system = 98.650 + 120.428 + 15.052 = 234.129 MJ

This total heat in the primary system (excluding the decay heat) is assumed to be removed in three hours, because decay heat removal with TJ can start after three hours.

Cooling down phase

For cooling down, the heat is removed by evaporation in the steam generators. The amount of water required to remove the total heat from the primary system including the decay heat is calculated by dividing this heat by the evaporation heat (2,250 KJ/kg ~ 2,250 MJ/m^3).

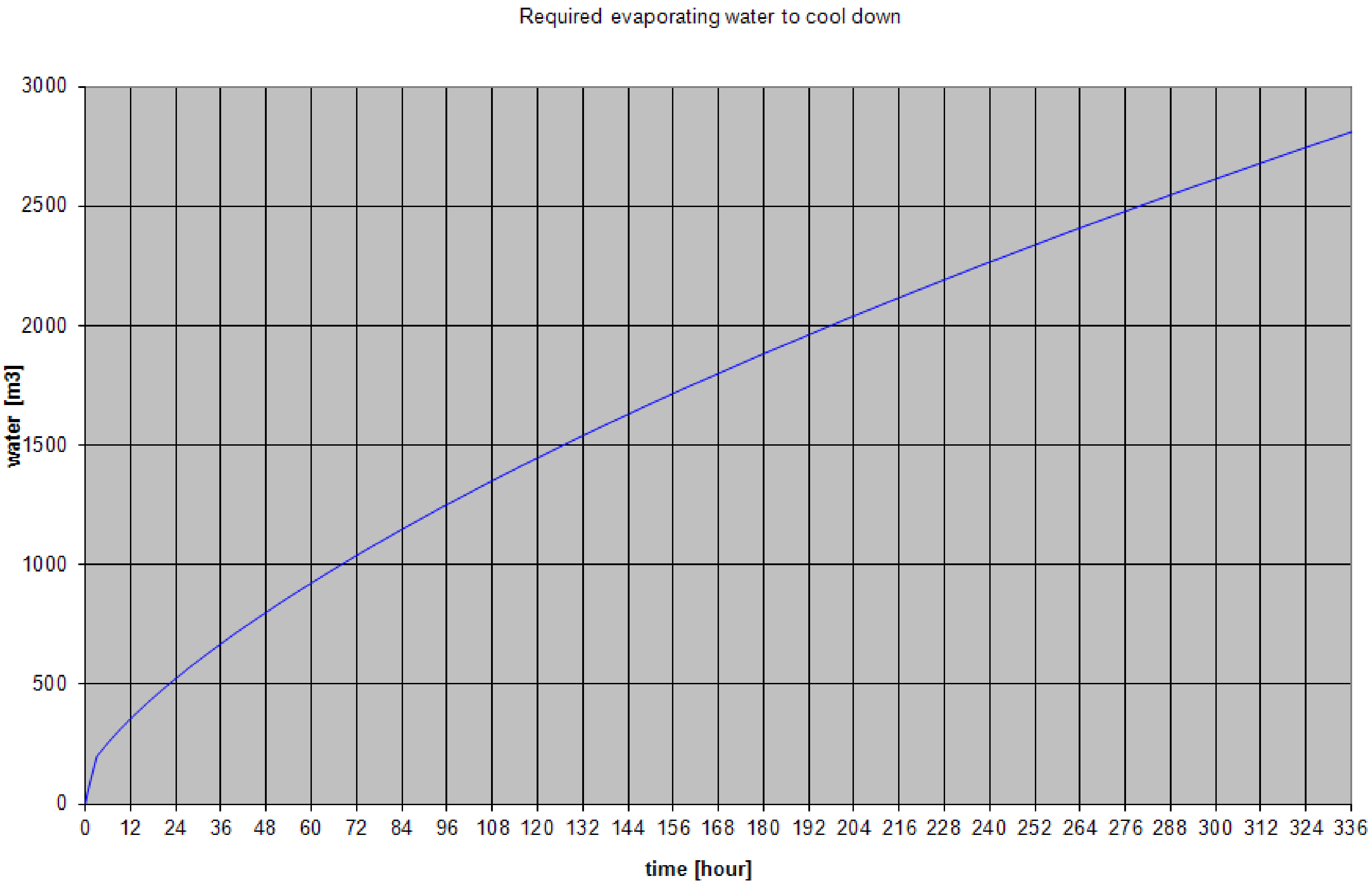
Remarks:

1. the cooling down condition is set at 120 °C and 13 bar. This means that scenario's with decay heat removal by TJ remove too much heat, because of the higher primary process conditions (180 °C and 30 bar). In the TJ-case this heat is normally removed in the decay heat removal phase. So for the TJ cases the assessment in the cooling down phase is somewhat conservative, which is compensated for in the decay heat removal phase.

2. the continued operation of the main cooling pumps is set at a conservative half- hour.

3. the heating up of the supplied water to the steam generators is not taken into account. This is a conservative assumption. Since the cooling options have different feed-water temperatures this would complicate the assessment. The results can now be applied to all options.

The graph below shows the amount of water necessary for evaporation to remove the heat from the primary system over two weeks (336 hours). The first part of the line (three hours) is steeper because of removing the heat from the primary system is in addition to the decay heat. After three hours the line follows the decay heat curve.


Required evaporating water to cool down
water [m3]
3000
2500
2000
1500
1000
500
0
0 12 24 36 48 60 72 84 96 108 120 132 144 156 168 180 192 204 216 228 240 252 264 276 288 300 312 324 336
time [hour]

Decay heat removal phase

The same enveloping decay heat data is used for the cooling down phase.

The heat is now removed by heating up thesupplied water.

Assumptions:

The design characteristics of the TE/VE cooler are also generally applied to cooling via TF/VF.

Cooling water heat exchanger in: 34 °C

Cooling water heat exchanger out: 90 °C

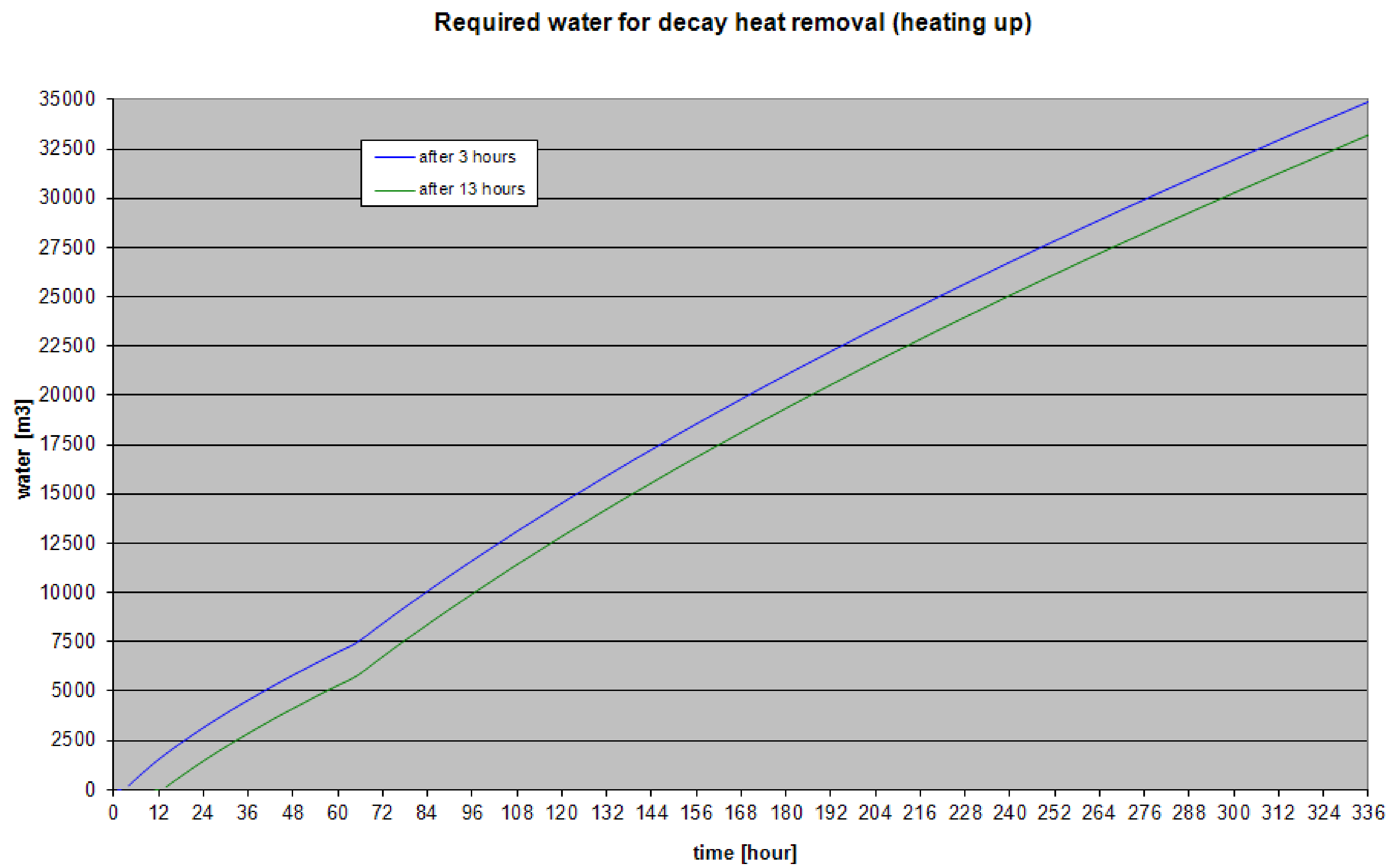
After 72 hours cooling water out: 70 °C

Heat capacity of water: 4.186 KJ/kg.K ~ 4.186 MJ/m^3

For the assessment, these two situations are identified as:

- decay heat removal after three hours and
- decay heat removal after 13 hours

The graphs presented below showing the amounts of water that need to be heated up to remoel the heat from the primary system over a two week period (336 hours) starting from three hours and from 13 hours.


Required water for decay heat removal (heating up)
after 3 hours
after 13 hours
water [m3]
35000
32500
30000
27500
25000
22500
20000
17500
15000
12500
10000
7500
5000
2500
0
0 12 24 36 48 60 72 84 96 108 120 132 144 156 168 180 192 204 216 228 240 252 264 276 288 300 312 324 336
time [hour]

Spent fuel pool cooling

The same enveloping decay heat data is used for spent fuel pool cooling.

A complete core load is present in the pool.

The spent fuel pool cooler TG080 is the most important heat remover for the spent fuel pool.

The characteristics of this cooler are also generally applied for cooling via TF/TG.

The heat removal capacity of TG080 is 5.15 MW.

The enveloping decay heat is 5.14 MW at 105 hours.

The spent fuel pool cooling starts at 110 hour after reactor shutdown.

The heat is removed by heating up supplied water.

<u>Assumptions:</u>

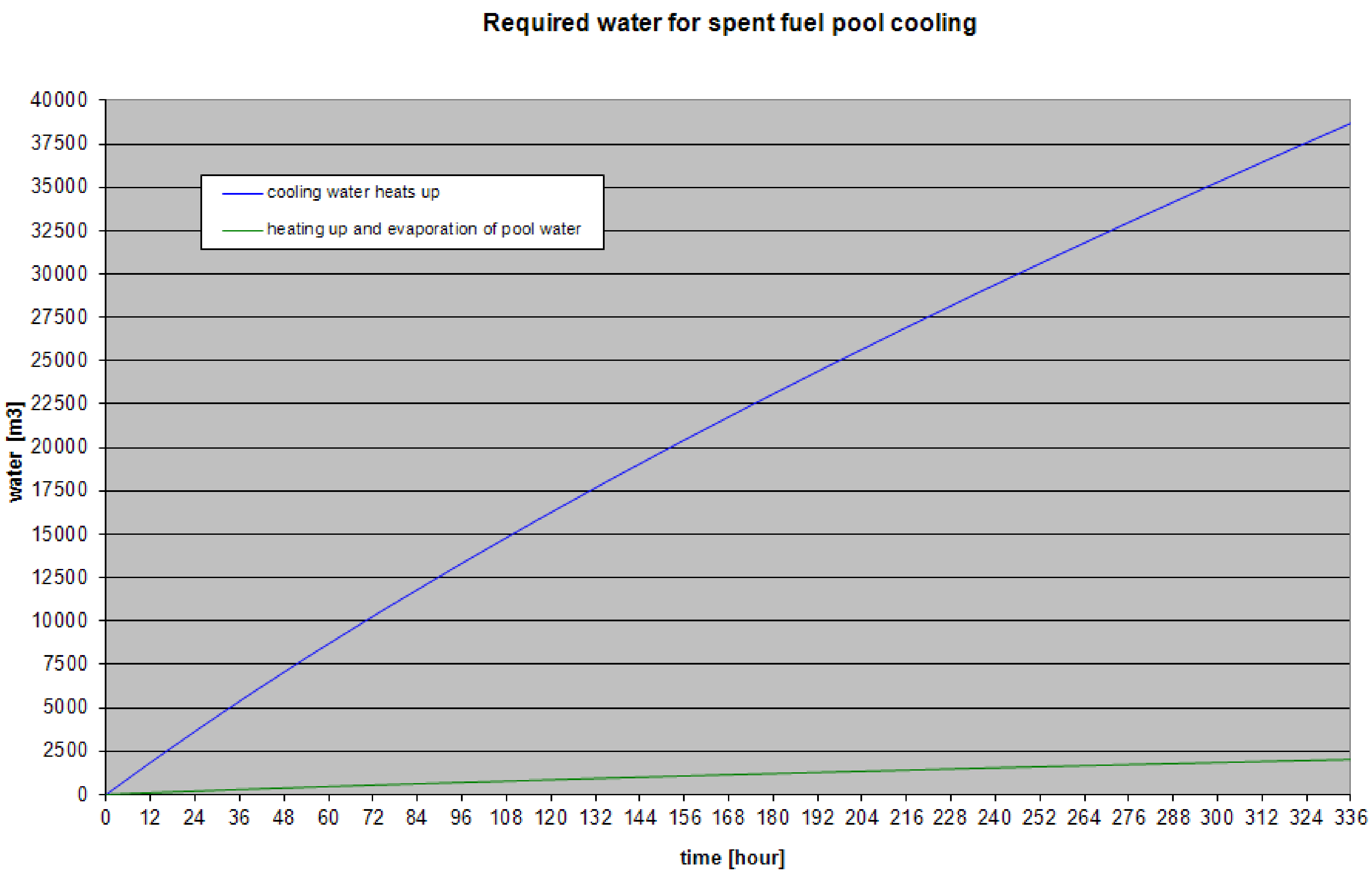
Cooling water heat exchanger in: 20 °C

Cooling water heat exchanger out: 48 °C

Heat capacity of water: 4.186 KJ/kg.K ~ 4.186 MJ/m^3

The heat can also be removed by evaporation of the pool water followed by refilling the pool to compensate for the decreased level. In that case, the amount of water required to remove the heat from the fuel in the spent fuel pool is calculated by dividing the heat evaporation heat (2,250 KJ/kg ~ 2,250 MJ/m^3).

The graph below shows the amount of water that has to be heated up and then evaporated to remove the heat from the spent fuel pool over a two week period (336 hours).


Required water for spent fuel pool cooling
cooling water heats up
heating up and evaporation of pool water
water [m3]
40000
37500
35000
32500
30000
27500
25000
22500
20000
17500
15000
12500
10000
7500
5000
2500
0
0 12 24 36 48 60 72 84 96 108 120 132 144 156 168 180 192 204 216 228 240 252 264 276 288 300 312 324 336
time [hour]

Combined cooling with UJ

The UJ supply can be applied to combined cooling down and spent fuel pool cooling as well as for combined decay heat removal and spent fuel pool cooling. For this situation the following assumptions are made.

The reactor contains a normal fuel load. The spent fuel pool contains one quarter of the fuel load of the earlier cycle. These elements are in the pool for ten days. This is the minimum time of a refuelling outage. There are probably also some elements from earlier cycles, but these produce only a small amount of heat. To cover this, the produced decay heat in the pool is set on 1/3 of the decay heat of core in the reactor.

Therefor the heat to be removed heat = ¼ of core decay heat (refuelling) + 1/12 of core decay heat (old elements).

<u>Assumptions:</u>

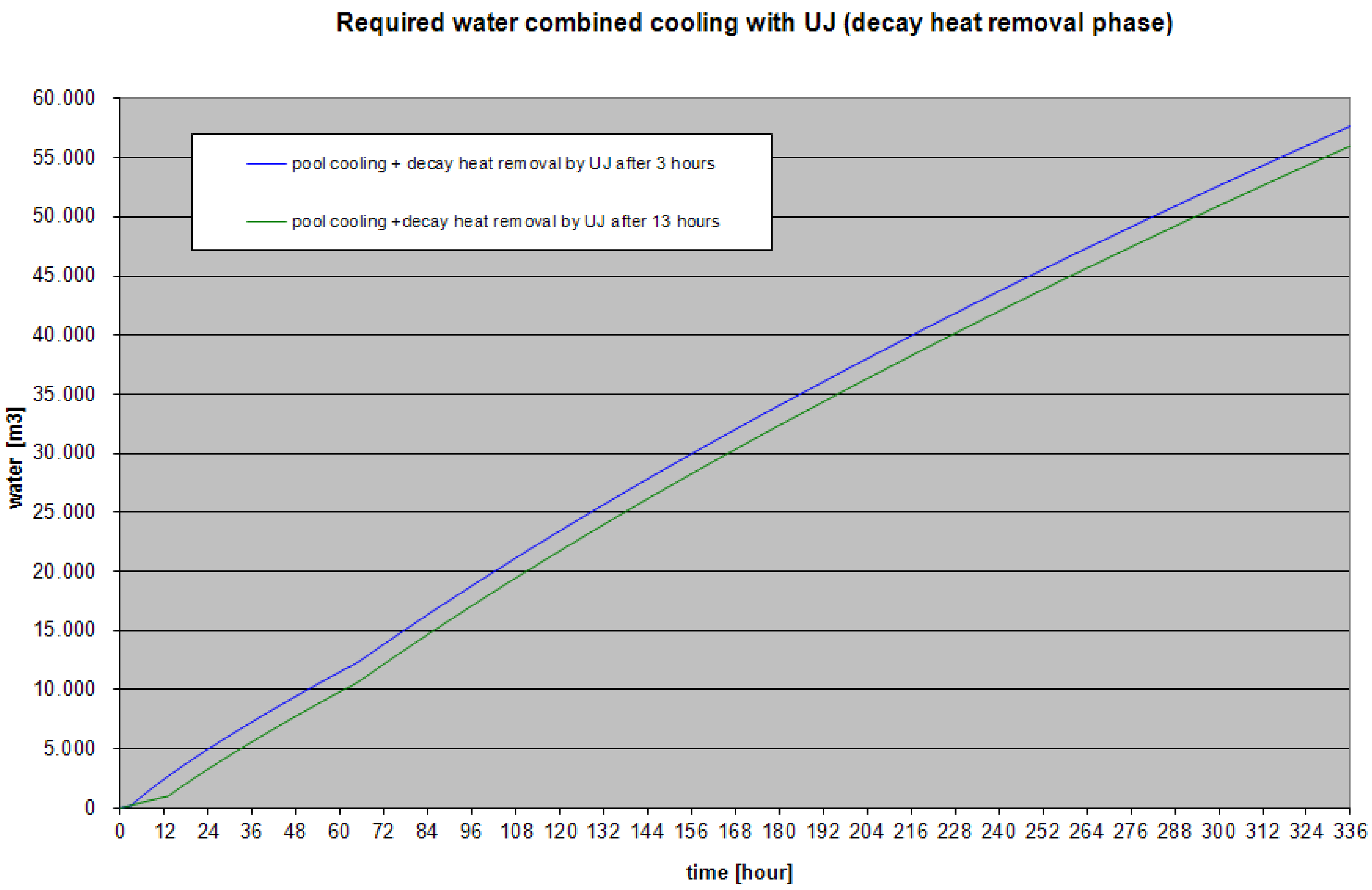
Cooling water heat exchanger spent fuel pool in:	20 °C
Cooling water heat exchanger spent fuel pool out:	34 °C
Cooling water heat exchanger decay heat removal in:	34 °C
Cooling water heat exchanger decay heat removal out:	90 °C
Cooling water heat exchanger decay heat removal out after 72 h:	90 °C

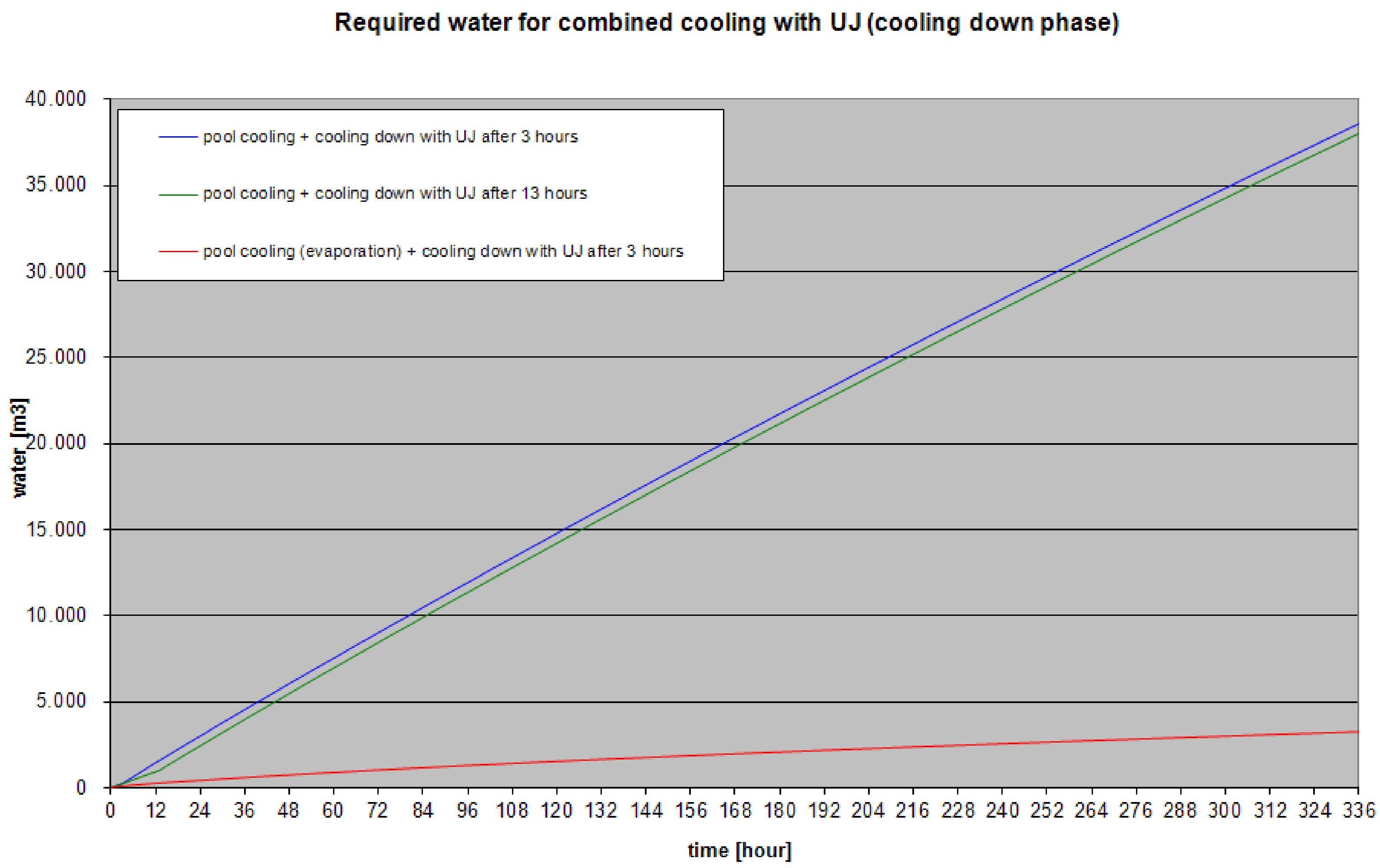
Heat capacity of water: 4.186 KJ/kg.K ~ 4.186 MJ/m^3

Three UJ combinations are presented:

1. decay heat removal with UJ + pool cooling (both heating up the supplied water)
2. cooling down with UJ (evaporation of water) + pool cooling (heating up the supplied water)
3. cooling down with UJ + pool cooling (both evaporating the supplied water)

The combination of decay heat removal and cooling down are presented in the following graphs.


Required water combined cooling with UJ (decay heat removal phase)
pool cooling + decay heat removal by UJ after 3 hours
pool cooling +decay heat removal by UJ after 13 hours
water [m3]
60.000
55.000
50.000
45.000
40.000
35.000
30.000
25.000
20.000
15.000
10.000
5.000
0
0 12 24 36 48 60 72 84 96 108 120 132 144 156 168 180 192 204 216 228 240 252 264 276 288 300 312 324 336
time [hour]


Required water for combined cooling with UJ (cooling down phase)
pool cooling + cooling down with UJ after 3 hours
pool cooling + cooling down with UJ after 13 hours
pool cooling (evaporation) + cooling down with UJ after 3 hours
water [m3]
40.000
35.000
30.000
25.000
20.000
15.000
10.000
5.000
0
0 12 24 36 48 60 72 84 96 108 120 132 144 156 168 180 192 204 216 228 240 252 264 276 288 300 312 324 336
time [hour]

Chapter 6 Severe accident management

6.1 Organisation and arrangements of the licensee to manage accidents

6.1.1 Organisation of the licensee to manage the accident

The basic requirements for the station's emergency preparedness are given in the Nuclear Energy Act operating licence, in particular licence condition B.23 and FSAR 13.1.4.

The NPP emergency planning and organisation have been synchronised with the national crisis organisation as defined in the National Plan for Nuclear Emergency Planning and Response (NPK). They have been submitted to the regulatory body prior to implementation.

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.

6.1.1.1 Staffing and shift management in normal operation

There are 7 operations shift teams, each managed by a shift supervisor.

A shift team is composed of a minimum of 8 operators: 1 shift supervisor, 1 deputy shift supervisor, 2 control room operators, 3 field operators and a electrician/field operator. The deputy shift supervisor can functional be replaced by a senior control room operator.

Shift supervisor

The shift supervisor is responsible for maintaining the nuclear safety for the production process and producing electricity in an economic way, by operating and testing the plant and guarding the process, by which instructions are demonstrably followed

Deputy shift supervisor

The deputy shift supervisor is responsible for verifying the plants nuclear and conventional safety and the way of operation. Control of calamities which require extra coordination, inside and outside the control room area. In case the shift supervisor is temporary absent, the deputy shift supervisor has command over the shift team. The deputy shift supervisor has the same license as the shift supervisor and, when assigned by the manager operations, can replace the shift supervisor completely.

Senior control room operator

The senior control room operator has the same responsibility as the control room operator. Because of the extra training, the senior control room operator can replace the deputy shift supervisor with the related responsibilities. However, the senior control room operator has no license as shift supervisor and can only replace the shift supervisor for short periods, for instance when the shift supervisor has left the control room area to conduct his housekeeping rounds.

Control room operator

The control room operator is responsible for maintaining nuclear safety for the production process and operating the plant from the control room in the most safe and efficient way, as well as testing and checking the plant from the control room and guarding the process, demonstrably following the valid instructions. The control room operator resorts under the shift supervisor.

Field operator

The field operator is responsible for checking, operating and testing the plant outside the control room area on the control room operators responsibility.

Field operator/electrician

The field operator/electrician is, as the field operator, responsible for checking operating and testing the plant outside the control room area on the control room operators responsibility, both on operational and electro technical field. The field operator/electrician also is responsible for starting an electro technical investigation in case of a failure.

Shift requirements

A shift is manned with sufficient capable personnel to secure safe operations at all times. The shift time table is based on a five week period with six shifts creating time for a dayshift week for (refresher) training.

6.1.1.2 Plans for strengthening the site organisation for accident management

The first response group of the fire response organisation is led by the deputy shift supervisor. This means that in case of an event including fire the deputy shift supervisor must leave the control room to lead the response team until his position can be taken over by a colleague from the fire fighting organization. In the mean time he is not available to assist the shift in the control room to manage the emergency situation in the plant. In the near future the team leader function of the fire response team will be transferred to the security department, this will leave the deputy shift supervisor in the control room in case of an event including fire. A new group of eight fire chiefs will be introduced as part of the security department.

In case of physical isolation caused by an external hazard, e.g. dike failure with flooding, additional personnel will be mobilized (operators and maintenance crew). Whereas the infrastructure will still be intact, 2 additional shifts will be called on site to occupy both the emergency control room and the emergency response centre (ACC: Alarm Coördinatie Centrum), for the sake of emergency preparedness.

6.1.1.3 Measures taken to enable optimum intervention by personnel

The organisation to manage accidents is described in the emergency plan and includes:

- conditions where the emergency plan is applicable;
- emergency response organisation, including alerting the authorities;
- possible measures;
- overview of emergency centres, equipment, etc. and contains four planning sections:
    - personnel safety drills;
    - fire safety drills;
    - process safety drills;
    - security drills.

The planning sections can be executed simultaneously.

The Emergency Response Organisation (ERO) supports plant operation in accident and severe accident conditions.

Abnormal conditions are classified by the shift supervisor, who initially decides on the extent of the emergency response organisation to be activated. An overview of the full emergency response organisation is given in Figure 6.1

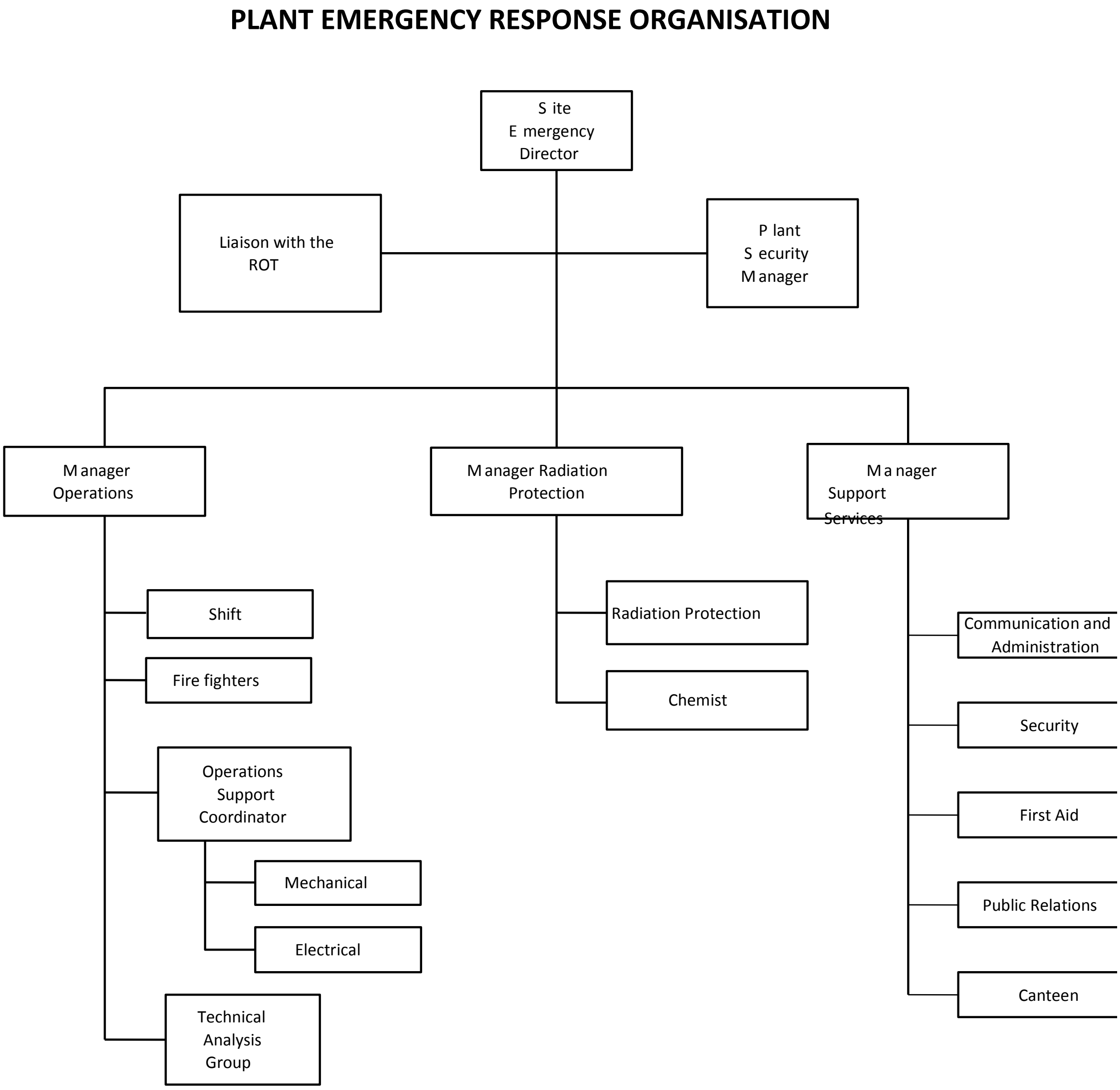


Figure 6.1 Plant emergency response organisation

The emergency response organisation of the plant is an integrated, separate organisation that incorporates industrial and nuclear safety, first aid, fire-fighting and site security functions. This means that the emergency response organisation is actually a combination of an industrial safety and a nuclear emergency organisation. It has the following tasks:

- 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, in the event of an emergency, responsible for the emergency response until the ERO is operational. From that moment the Site Emergency Director (SED) takes on this responsibility from the shift supervisor. The SED, as head of the plant's ERO, is responsible for all decisions and actions taken by the emergency facilities.

The SED is chairman of the Emergency Management Team that consists of: the SED, the Operations Manager (MB), the Radiation Protection Manager (MSB) and the Support Services Manager (MOD). The SED can be advised by the Plant Security Manager (PSM) in case of a plant security issue. The liaison officer, who liaises with the Regional Operational Team (ROT) in Middelburg of the so called Veiligheidsregio, is an EPZ staff member who will be sent to the ROT to provide them with technical explanations on the plant's status and the proposed actions.

Each of the three managers is responsible for a specific area of the plant's ERO and its corresponding tasks. The MB and his group are responsible for the plant process and interventions at the plant. This group is formed by:

- the MB, who is responsible for managing his group and advising and informing the SED;
- the on-duty shift personnel;
- the fire-fighters, both the first response group (shift personnel) and the voluntary fire fighters;
- the Operations Support Group under the Operations Support Coordinator, which can perform mechanical and electrical repair work;
- the Technical Analysis Group (TAG) which provides engineering support (handling of the Severe Accident Management Guidelines).

The MSB and his group are responsible for actions to protect the workers, the public and the environment from the radiological consequences of an accident. This group consists of:

- the MSB, who is responsible for managing his group, advising and informing the SED, and for drafting the advice to the ROT on the external actions to be taken;
- the Radiation Protection Group, which is responsible for radiological analysis and support;
- the Chemistry Group, which takes samples and performs chemical analysis.

The MOD and his group support the emergency organisation with communication and administration, site security, first aid, public relations and logistics. This group consists of:

- the MOD who is responsible for managing his group, and advising and supporting the SED;
- the Communication and Administration Group;
- security personnel;
- first aid personnel;
- the Public Relations Officer;
- canteen personnel.

A major proportion of the plant's ERO is on call via a pager (and in some cases a cell phone). Exceptions are: the liaison officer, the Communication and Administration Group, the Public Relations Officer and the canteen personnel who will be called when needed. The shift personnel, site security personnel and first aid personnel are always on duty. There are enough qualified persons for each function in the plant's ERO to guarantee the availability of the emergency organisation throughout the year. The on-call duty cycle is one week (Friday to Friday). Members of the plant's ERO should be on the site and functioning within one hour.

There is one scalable emergency response organisation and this covers all types of emergency: personal injuries, conventional or nuclear incidents etc. This means that the number of emergency staff alerted in case of an incident will be determined by its scale. It is important that the SED is always in charge of the emergency response organisation, independent of the scale of the incident and the number of staff involved.

6.1.1.4 Use of off-site technical support for accident management

The NPK prescribes two protocols between the plant's ERO and the authorities that will be used to communicate during an emergency situation, one at local level and one at national level. 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. EPZ sends a liaison officer to the ROT to give them extra information on the plant status and to explain the SITRAPs to them when necessary. At national level information is distributed via the KFD. The KFD has a process computer station in The Hague and is therefore able to monitor plant parameters online.

The KFD can also communicate with the SED and the MSB about the classification of the event and the (expected) releases to the environment and with the TAG about the process and actions to mitigate the event.

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 given by the so called 'Krisenstab' (crisis staff) which is a group of engineers who will come when requested by the power plant.

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 (PPS) of the plant. This connection is not working during normal operation, it has to be switched on by the plant personnel. The contacts between the plant's ERO and the Krisenstab are via the TAG.

The Plant Security Manager can communicate about safety actions with the Department of Nuclear Safety and Safeguards at the Ministry of Economic Affairs, Agriculture & Innovation (EL&I) in case of a security problem on site.

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.

Providing information or issuing warnings and instructions to the public in the event of a nuclear accident is the responsibility of the authorities. The EPZ will cooperate with the authorities when asked to provide information for the public and the media.

6.1.1.5 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).
  Procedures for normal, undisturbed operation include plant and system operating procedures, checklists, surveillance requirement execution procedures, etc.;
- abnormal conditions (all operational states).
  Procedures for likely deviations are the so-called S-instructions;
- accident conditions (from hot-steaming through to full power states).
  Emergency Operating Procedures (EOP) are entered upon SCRAM and/or safety injection and consist of:

  - Nood Bedienings Procedures (NBP), which are based on the Westinghouse Owner's Group (WOG) generic Optimal Recovery Guidelines.
    NBPs prescribe verification of automatic actions in accordance with the design, accident management actions for optimal recovery and accident management actions for beyond design situations;
  - Functie Herstel Procedures (FHP), which are based on the WOG Functional Restoration Guidelines (FRG).
    FHPs are entered upon the detection of a threat to or a loss of a critical safety function, which are independently monitored by the deputy shift leader (with backup from the computerised process information system). FHPs prescribe actions to regain and ensure the critical safety functions.

- Severe accident conditions (all operational states).
  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. They consist of:

    - Severe Accident Control Room Guidelines (SACRG);
    - Severe Accident Guidelines (SAG).
      A SAG is activated when plant parameters exceed the level for controlled, stable operation as indicated in the Diagnostic Flow Chart (DFC);
    - Severe Challenge Guideline (SCG).
      A SCG is activated when an immediate and severe challenge to containment fission product boundaries occurs as indicated in the Severe Challenge Status Tree (SCST);
    - Severe Accident Exit Guidelines (SAEG).
      SAEGs describe actions to ensure long-term operation after a controlled, stable operation has been received;
    - Computational Aids (CA).
      These are used to assist diagnostics and decision-making.

Once the Emergency Operation Procedures (EOPs) have been exited and the SAMGs activated, plant conditions must be diagnosed and strategies to mitigate the accident must be evaluated. Summaries of the strategies to mitigate the accident as described in the SAGs, SCGs and SAEGs are given in Annex 6.1.

Restoring the installation after an incident to a stable safe condition is the primary task of the control room personnel, under the supervision of the Operations Manager (MB). Control room personnel and relevant technical staff get periodical re-training in handling these emergency procedures (NBPs, FHPs, SAMGs). For more specific information about the SAMG (re)training see Annex 6.2.

The implementation of EPZ's own full-scope simulator at the simulator centre in Essen (Germany) in 1997 has greatly improved the quality of training for 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 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. This use of the simulator has significantly enhanced the sense of realism of the integrated emergency exercises.

Application of the emergency procedures by the control room operators is the main tool to meet the operational challenges of an emergency situation. This allows the plant's ERO in the ACC to focus its attention on coordination of non process-related actions, such as communications with local and national authorities. The main focus of drills and exercises for the plant's ERO has changed over the years: from support to the shift supervisor to internal and external communications. Training has gradually shifted from predominantly technical training, towards the skills required for the organisational and communicative performance of the plant's ERO. This approach has started to pay off during the emergency exercises which are held annually in cooperation with local and national authorities.

The timely delegation of an expert from the power plant to the ROT of the Veiligheidsregio in Middelburg (the liaison officer with the ROT), has proven to be very useful. This expert is able to explain technical data to the ROT whenever necessary. Communication within the emergency organisation of EPZ is another point of attention during exercises. EPZ personnel have to be informed about an emergency situation on a regular basis, in order to avoid uncertainty among the staff.

The results of the integrated exercises are evaluated and the evaluation reports, with proposals for improvement, are distributed to the managers of the plant's ERO. The progress of approved actions for improvement is followed in the work order system until completion.

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 modular design of the training programme contributes to a significant improvement of skills, as well as greater efficiency. While groups and officials improve their skills in separate exercises (e.g. table-top exercises and instruction), the integrated exercises are the ultimate test for evaluating of overall emergency preparedness. The group managers are responsible for determining the training requirements of their group. These requirements form part of a document that describes all aspects and relevant procedures of training, drills and exercises. Every participant in a training session, drill or exercise is registered in a database by the Training Department. Deviations between training requirements and training results are reported to the responsible group manager on a quarterly basis. This enables managers to take appropriate measures, if necessary, to have training requirements met before the end of the year.

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.

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)

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 truck is needed to transport the onsite diesel generator to the connection point. That means that in both cases whether the onsite or offsite diesel generator is demanded, external support is needed to bring the mobile generator(s) to the connection point. Procedures to connect the mobile diesel generators to the emergency busses and to put them into operation are available. For the onsite diesel generator approximately 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 crashtender as used in airports. Accessory equipment like fire hoses is available too, also as a last-resort option of cooling water transportation.

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

Extensive water and diesel reserves are available at the KCB premises. A description of these can be found in Chapter 5.

Additionally, there are contracts for delivery of chemicals and fuel for the diesel generators. These contracts specify that diesel supply should be made within 8 hours after a request by the plant. There is no special statement in the contract that deals with emergency situations.

EPZ has two warehouses on the site which contain several equipment and spare parts to be used during outages and also during emergency situations. All available parts are registered. On basis of this registration all emergency parts are checked periodically.

6.1.2.3 Management of radioactive releases, provisions to limit them

The strategies used to limit radioactive releases after core melt are provided by the SAMGs. In particular the guidelines SAG-5, SAG-6 and SCG-1 give strategies to mitigate radioactive releases. Annex 6.1 gives a short description of these guidelines and the systems and components that are used to limit the releases.

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.

6.1.2.4 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 the 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.

For the ERO at Borssele, the use of telephones is important for both external and internal communications. In addition to the normal telephone lines, there is an emergency telephone network that serves as backup.

Furthermore, the national emergency telephone network can be used (an independent redundant telephone line), which represents direct communication lines between two locations on the plant or between a location on the plant and an external organisation. This method of communication is mostly used for communicating between the control room, the plant’s ERO, the regional police and the authorities. The national emergency network is used for communicating with the government. 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.

The following communication methods of communication can be used between the control room and the ACC bunker:

- the PPS computer;
- the normal fixed and mobile telephone networks;
- the emergency telephone network (both internal and external emergency networks independent from the normal telephone networks);
- fax (via normal or emergency telephone network);
- e-mail via the internal computer server net;
- walkie-talkies (portofoon);
- a direct telephone line between the telephone exchange in the former site in Goes and the control room at KCB. This means a telephone connection can be made between TAG and the control room by calling this ‘outside number’;
- standard forms to communicate between the control room and the ACC bunker have been developed, which include, amongst others, the SAMG long-term monitoring parameters;
- the C2000 communication network.

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.

Furthermore the communication by telefax becomes a back-up status for communication by e-mail.

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

In case of extensive destruction of the infrastructure around the plant, including the communications facilities, it is possible to use the accident management measures as described in the EOPs and the SAMGs. The plant is normally attainable from 3 directions which lead to the main or sub gate(s) of the site. If all roads are destroyed it will be difficult to enter the site. The shift personnel is always on site and relief of the shift personnel is possible, for example, by using helicopters. The use of mobile resources will depend on the destruction of the infrastructure as to how efficient the staff can carry out the accident management measures. Currently only a limited number of arrangements have been made for off-site support measures. Note that external resources cannot be guaranteed at short notice after the start of an accident as there may be extensive damage to the infrastructure around the plant which could include the communications facilities.

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.

6.1.3.2 Loss of communication facilities/systems

At the moment, there are no ready solutions to deal with situations when all means of communication mentioned in section 6.1.2.4 are lost. The emergency response centre as proposed in section 6.1.3.1 will however retain communications facilities during external hazards.

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

Impairment of work performance due to high local dose rates is to be expected in case of a severe accident. Dose measurement, shielding, protective clothing, respirators and limitation of the exposure time will be used to keep the dose of the workers within the limitation. Combined with the destruction of some facilities this can lead to a poor work performance.

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

The accessibility and habitability of the vital areas of the plant essential to manage the situation (the main control room, emergency control room, ACC bunker, local control and sampling points, workshops) are in general possible before and after the occurrence of fuel damage, except in the case of a (prolonged) external flooding. For this case, an external hazard-proof emergency response centre has been proposed (section 6.1.3).

In case operation from the main control room is not reliable or possible (after external events), the operation will take place from the emergency control room in the bunkered buildings in order to transfer the plant to a safe shutdown state. Emergency procedures regarding operation are available from the emergency control room.

The permissible levels of exposure during emergencies are laid down in instructions from the KCB[59], which are part of the emergency planning and preparedness and are given in table 6.1.

	KCB limit (mSv)
To save human lifes	500
To save important material interests	100
Execution or support of measurements, evacuation, iodine distribution, keeping public order and safety	100

Table 6.1 Exposure limits used by the KCB during emergencies

Dose rates for some relevant locations have been calculated for a representative core melt scenario. This radiation results from airborne radioactivity in the containment atmosphere, with successful containment isolation. The scenario assumes core melt leading to 100% fuel damage and release to the containment atmosphere of

[59] The KCB limits are far below the regular limits laid down in the Dutch “Besluit Stralingsbescherming”

- 100% of noble gases;
- 10% of halogens;
- 10% of Cs, Rb, Te, Se, Ba and Ru;
- 1% of other, non-volatile solids.

Additional radioactivity is assumed to be released from the fuel into the containment water.

Calculated dose rates for some relevant locations as a function of the time after reactor scram are given in table 6.2.

Dose rate in mSv/h				
	5 h	9 h	28 h	30 d
Main control room (05.513)	5.2	2.1	0.3	-
Alarm staff rooms in ACC bunker (15.121- 15.123)	0.03	0.009	0.005	-
Bunker used for gas sampling of containment atmosphere (03.101)	15	-	<2	<1
Chemical laboratory in ACC bunker	0.25	0.10	0.05	-
Measurement point at security lodge (XQ014)	42	17	7	0.04

Table 6.2 Calculated dose rates for some relevant locations as a function of time after the reactor scram

No dose rates were calculated for the emergency control room (building 35). These dose rates for the emergency control room are generally not expected to exceed those in the main control room, depending on the specific details of an accident scenario.

Workers and alarm staff will be continuously monitored to make sure they stay below the limits. Based on the calculated dose rates, a continuous presence in the alarm staffrooms in the ACC bunker will be possible, even during a severe core melt event. Depending on the actual accident sequence, the maximal duration in the control room of individual workers may have to be limited in order to stay below the maximum total dose limits. This is especially true for the first day of an accident event.

In case of a filtered release from the containment, the TL003 containment filtered venting system to the environment will be used. This filter system is qualified for severe accidents and is highly efficient for fission product releases, except noble gases. As a result, the staff can work in the ACC bunker (building 15), the main control room (building 05) and the emergency control room (building 35).

For unfiltered releases from the containment, a relatively large spreading in the consequences exists, depending on the details of the release (such as amount, composition, duration and location), weather conditions, shielding of the buildings and radiation shielding of staff. Note that the ventilation and air conditioning systems of the main control room have an improved design in order to increase the habitability in case of radioactive or toxic contamination of the environment. In case of a radioactive release to the environment the ventilation of the control room will be switched over to internal circulation with active coal filtering. In the control room is also a sufficient number of respirators with compressed air available. Furthermore, the ACC bunker has gas-tight doors and the air supply is via an active coal filter. How long the staff can work in the ACC bunker, the main control room and emergency control room depends on the details of the release (such as amount, composition, duration and location), weather conditions, shielding and a possible destruction of infrastructure.

At the moment, the shift on duty must initiate the first actions in case of a calamity, together with the plant emergency response organisation. In some emergency situations like flooding, more people will be needed. Following the current procedure, people will be called by order of the Site Emergency Director to be available on-site or to relieve shift personnel on duty.

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

The emergency control room is designed to withstand external initiators and is equipped with the controls that are necessary to bring and keep the plant in a stable situation. Due to its design, the impact of an accident on the emergency control room will be very limited. 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. 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, see section 6.1.3.1.

Depending on the nature of the external event, various buildings will still be available to house the incoming crisis teams. Directly behind the plant there is some flat farmland/grassland to locate mobile housing and equipment if necessary.

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

The accident management measures are feasible under conditions of external hazards like earthquakes or floods. The SAMGs provide guidance to deal with these situations and provide a number of different, redundant alternatives to reach specific goals. Part of the plant is protected against earthquakes and floods. Even when some alternatives are not available anymore other alternatives in the protected zones will work. The emergency response centre proposed in section 6.1.3.1 will enhance emergency preparedness in case of the mentioned external hazards.

6.1.3.7 Unavailability of power supply

The provisions in case of unavailability of power supply are described in Chapter 5. For the autonomy period of the systems, see Chapter 5.

The connection of an external power supply to the bus bars CX and CW will be made according to the instructions WNE-CX-003 and WNE-CW-003.

Transport for the emergency diesel generator EY080 is currently not mentioned or described in a separate procedure.

A delivery contract, calamity plan and procedures to put the mobile diesel generator in operation are available. Transportation by means of an available truck of the mobile diesel generator is not foreseen at the moment. 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.

6.1.3.8 Potential failure of instrumentation

During severe accidents the plant diagnostics and the need to consider severe accident management strategies is keyed to a limited number of plant parameters. These are the key parameters that are used in the SAMGs. The important instrumentation for this purpose is qualified for both (severe) accident conditions and a harsh environment. The power is supplied by the emergency power supply systems and by the batteries.

It is noted that the SAMGs are developed on the basis that any instrument that is believed to provide useful information should be used, even if not ‘qualified’ for beyond design basis conditions. Therefore instrumentation without a qualification for severe accident conditions can still be used.

Additionally, environmental conditions during a severe accident may not exceed the conditions for which the instrument is qualified. For example, in most severe accident scenarios, the containment pressure and temperature conditions do not exceed those for design basis accidents for many hours after core damage.

Furthermore, even for conditions beyond the design basis, instrumentation which is not qualified will not fail immediately. Information on survivability of the instrument sensor or transmitter can, for example, be found from the environmental testing data where the actual conditions to which the instrumentation was exposed were well beyond the qualification limits. This extra margin up to the failure point is useful in severe accidents. Note, however, that during severe accidents the operator cannot rely on all instrumentation.

Also note that in the phase before occurrence of fuel damage, the environmental conditions are, in general, within the design basis of the instrumentation. Therefore the required instrumentation is available, even from instrumentation which is not harsh environment qualified.

In case of an assumed failure of instrumentation, the accident management measures for restoring core cooling, protecting the integrity of the containment function and mitigating the consequences of the accident will proceed. This is also a part of severe accident management training at the plant. Therefore despite an assumed failure of instrumentation, accident management measures for cooling the fuel and protecting the integrity of the containment function will still be taken. The six computational aids can be used to check the credibility of a measured parameter against other measurements.

In case of potential failure of instrumentation, procedures for operating or stabilising 'by hand' are foreseen. In case of in corrective measurements, reserve equipment (instruments) for the nuclear-safety related systems is available. The maintenance department manages the stock levels of available instrumentation for nuclear safety-related instrumentation.

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

As KCB is a single-unit plant, typical multi-unit plant effects are not applicable here. However, the coal fired power plant CCB is neighbouring KCB. This can be useful when strong current equipment or knowledgeable personnel is needed. Also an additional diesel generator is availble on the CCB premises.

A potential negative effect from the neighboring coal-fired power plant could be missiles from the turbines. The equipment which is needed to transfer the nuclear plant to the safe shutdown state is however protected by the structure of the buildings at the Borssele plant. Furthermore, note that the undesirable external effects are mentioned and foreseen according the applicable KCB instructions.

6.1.4 Conclusion on the adequacy of organisational issues for accident management

The organisation and arrangements of the KCB to manage accidents is considered to be sufficiently adequate. Nevertheless some recommendations are given to enhance emergency preparedness. These are elaborated in section 6.1.5.

Legal arrangements like the Nuclear Energy Act and the National Plan for Nuclear Emergency Planning and Response are in place. Responsibilities for emergency response and provision of information are clearly distributed over licensee and public authorities. An emergency plan is available and an ERO is in place. This 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 licensing authority KFD, the local authorities (ROT Veiligheidsregio), the crisis staff of the plant vendor and the local hospitals have been made for off-site support. Extensive attention has been paid to the availability and training of procedures, both on-site and in the full-scope simulator located in Germany. On-site available equipment has been examined and found to be sufficient. Some measures to enhance the accident management capabilities have been defined, as elaborated in section 6.1.5.

6.1.5 Measures which can be envisaged to enhance accident management capabilities

The following measures have been identified to enhance accident management capabilities:

- 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 (see section 6.1.2.4);
- 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 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;
    - develop check-lists for plant walk-downs and needed actions after various levels of the foreseeable hazards;
    - connecting CCB/NS1;
    - uncouple lower rails in time in case of flooding;
    - alternative supplies for UJ.

- 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.

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

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

The last-resort accident management measures to prevent fuel damage are described in the following procedures:

- 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

Note that the accident management measures mentioned in the procedures ECA-0-0, S-EY-01 and S-EY-02 apply to a LOOP or SBO situation. Procedure ECA-0-0 transits into the Severe Accident Management Guidelines (SAMGs) when the core exit temperature reaches 650°C and is still increasing. For a more extensive evaluation of the LOOP-SBO scenario for KCB, please refer to Chapter 5.

6.2.2 After occurrence of fuel damage in the reactor pressure vessel

The accident management measures after the occurrence of fuel damage are described in the following guidelines (see Annex 6.1):

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

Note that more than one SAG may be evaluated at a time and the implementation of strategies should follow the priorities dictated. Other SAGs which might be important after the occurrence of fuel damage with respect to core cooling are:

- 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

After the failure of the reactor vessel core debris will leave the primary system. The accident management measures currently in place for cooling ex-vessel core debris outside the cavity and for scrubbing fission product releases of ex-vessel core debris outside the cavity are described in Severe Accident Management Guideline SAG-4:

1. Injection of water from the TJ storage tanks by the containment spray pumps;
2. Injection of water from the TJ storage tanks by gravity drain.

These accident management measures are explained in more detail in Annex 6.1.

With respect to cooling core debris inside the cavity, detailed investigations performed have concluded 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 already addressed in Severe Accident Management Guideline SAG-3. SAG-3 is summarised in Annex 6.1.

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

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

6.3.1.2 Operational provisions

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:

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 Annex 6.1.

6.3.2 Management of hydrogen risk inside the containment

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

The Borssele NPP has a hydrogen control system consisting of passive automatic catalytic recombiners (PARs) located in the containment. These recombiners do not require electrical energy for their operation. Gas mixtures containing hydrogen and oxygen are combined upon contact with the catalyst. The recombiner consists of a metal housing designed to promote flow with gas entering on the bottom and gas exiting at the top. The system is sized and designed for operation during a severe accident. The recombination capacity is such that:

- the system can recombine hydrogen faster than it is generated during the molten core concrete interaction phase of a severe accident;
- during the initial (in-vessel) phase of the accident, the hydrogen concentration in containment is limited to approximately 10 vol% at any location.

As the spent fuel pool is located in the containment hydrogen produced by a zirconium water reaction in the spent fuel pool will also be recombined by the installed PARs.

The PARs are specifically designed for severe accident mitigation and therefore expected to be available and to function during a severe accident. In the unlikely event that the recombiner system malfunctions or does not operate, hydrogen concentrations can increase to a level which could represent a challenge to the containment. Two SAMGs deal with this unlikely situation: SAG-6, (see Annex 6.1) and SCG-3, (see Annex 6.1). These guidelines give guidance on how to measure the hydrogen concentration within the containment and to manage its flammability or prevent deflagration or detonation.

The ‘in-vessel’ hydrogen production determines for the concentration of hydrogen in the containment because this hydrogen is produced on a short time scale at a rather high rate (order of magnitude 100 g/s). The ‘ex-vessel’ hydrogen production (mainly MCCI) occurs later during the accident and at a much slower rate (order of magnitude 1 g/s). Therefore, with the installation of the passive autocatalytic recombiners (recombination rate in the order of 10 - 100 g/s), the “ex-vessel’ hydrogen source can be effectively encountered, while the ‘in-vessel’ hydrogen concentration can be limited.

Hydrogen production caused by molten core-concrete interaction is a smaller problem for KCB than for other nuclear power plants because of the relatively high content of carbonates in the KCB concrete compared to, for instance, most German PWRs. This leads to a relatively higher production of carbon dioxide, which has an inerting effect.

The KCB plant has a containment hydrogen measurement system (TS090) with sample points in the operational area and in the installation area. The hydrogen measurement instrumentation is harsh environment qualified. Besides this, the use of hydrogen measurement instrumentation, backup systems such as TV090 hand sampling and TV061/062 systems is included in the SAMG strategies.

6.3.2.2 Operational provisions

With respect to the prevention of H_2 deflagration or H_2 detonation in the containment the following accident management measures are applicable:

- active opening of relief hatches between the installation area and the operations area of 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.

One of the accident management measures in SCG-3 is filtered venting the containment if other strategies are not successful. However, as mentioned in the guideline SCG-3 (see Annex 6.1), a long-term concern to take into consideration is a possible return to the hydrogen severe challenge area in the case of very high hydrogen concentrations (above 12%, dry measurement). In order to stay outside the hydrogen severe challenge area, the actions to take are inertising the containment by steam addition or by nitrogen injection from the accumulators.

6.3.3 Prevention of overpressure of the containment

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

6.3.3.2 Operational and organisational provisions

The following accident management measures are applicable:

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

6.3.4 Prevention of re-criticality

6.3.4.1 Design provisions

To prevent recriticality, boron can be injected in the primary system by the use of the operational boron suppletion system in combination with the volume control system, or by the safety injection systems.

6.3.4.2 Operational provisions

Function Restoration Procedure FHP-S-1 (see Annex 6.5) gives guidance to restore sub-criticality.

6.3.5 Prevention of basemat melt through

Prevention of basemat melt though is divided in two complementary strategies; each focus on preserving one barrier: in-vessel retention and in-containment retention (after vessel failure).

Several international research programs focus on the debris cooling strategies (in-and ex-vessel) and basemat melt though prevention. Lessons, results and possible ameliorations from these research programs are actually being studied in the current periodic safety review 10EVA13.

6.3.5.1 Potential design arrangements for retention of the corium in the pressure vessel

KCB has no core catcher or any other design arrangement for retention of corium in the pressure vessel. The SAMGs however provide a strategy to cool the debris within the reactor vessel in order to provide a way for retention of the corium in the pressure vessel.

Severe accident management procedures are available to enhance / restore the corium cooling in the reactor vessel: SAMG-SAG-3 (see Annex 6.1, injection in the RCS) gives guidance to the possibilities, arrangements and systems that can be used to restore corium cooling.

In-vessel melt retention is an accident management strategy to cool the reactor vessel preventing vessel failure and relocation of the core in the containment.

Important strategies are:

- debris cooling by restoring primary circuit reflooding: this strategy is used in the SAMGs of KCB;
- external vessel cooling: this is a potential design arrangement that guarantees a cooling water/steam flow at the lower part of the reactor vessel. The narrow gap between the vessel, the 'isolation cylinder' and the concrete wall (biological shield) could be used for water and steam flows. Due to the KCB layout this is almost not feasible.

6.3.5.2 Potential arrangements to cool the corium inside the containment after reactor pressure vessel rupture

After failure of the reactor pressure vessel debris cooling in the cavity is more difficult due to the specific layout of KCB. Cavity flooding is the most adopted strategy and is being investigated in the current periodic safety review 10EVA13. Potential arrangements are early partial or complete flooding of the cavity and use of the findings of the Molten Core Concrete Interactions (MCCI) research programs.

With respect to cooling core debris inside the cavity, detailed investigations performed have concluded 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 SAMG-SAG-3 (see Annex 6.1).

SAMG-SAG-4 (see Annex 6.1) can be used to control sump water level (i.e. level outside the cavity) to ensure cooling of any debris which may escape the cavity and scrubbing of fission product releases from ex-vessel core debris outside the cavity.

The other Accident Management measures applicable are:

1. Inject into the containment: Severe Accident Management Guideline SAG-4 (see Annex 6.1);
2. Control the containment conditions: Severe Accident Management Guideline SAG-6 (see Annex 6.1);

3. Reduce the containment pressure: Function Restoration Procedure FHP-Z-1 (see Annex 6.7);

4. Reduce the containment pressure: Severe Accident Management Guideline SCG-2(see Annex 6.1).

6.3.5.3 Cliff edge effects related to time delay between reactor shutdown and core meltdown

In 6.2 the actions c.q. procedures that are followed to prevent core melt and used after the occurrence of a core melt are described. It can be concluded that the Emergency Operation Procedures (EOPs) and SAMGs provide strategies to mitigate the accident for all possible scenarios. A specific timeline cannot be given because every scenario differs, but entering SAG-1 to SAG-4 of the SAMGs (see Annex 6.1) can be regarded as a cliff-edge, i.e core meltdown will happen when core cooling failure remains.

The cliff edge effects and the time before occurrence of fuel damage in case of external flooding are described in Chapter 3.

The cliff edge effects in case of earthquake are described in Chapter 2.

The cliff edge effects in case of loss of primary UHS and total loss of AC-power (referred to as loss of primary UHS with SBO-2) are presented in Chapter 5.

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

For ensuring the containment isolation function it is important that the containment isolation valves are closed. 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.

Note that when using the TL003 containment filtered venting system the isolation valves of the TL003 containment filtered venting system to the environment will be opened. These valves can also be operated manually from outside the buildings without electrical power, thus this accident management measure can be used without any electrical power supply.

6.3.6.2 Operational provisions

The following strategies can be performed if electrical AC power is lost:

- increase the steam concentration in the containment by opening the pressuriser tandem relief valves (this strategy is used in SCG-3 (see Annex 6.1), reducing containment hydrogen). The required power for opening the valves is supplied by the batteries,
- stop the heat removal from the containment by stopping the following components (this strategy is used in SCG-3 , controlling hydrogen flammability). In case of loss of AC power this will happen automatically because the following cooling systems are AC powered:
    - containment spray;
    - air coolers TL030-032;
    - biological barrier coolers TM001/002;
    - coolers TL040-045 and TL111-114;
    - annulus coolers TL050-053, TL055-058, TL060-TL064, TL066/TL068.
- active opening of the relief hatches in the containment building between the installation area and the operations area (to reach a more even distribution of hydrogen in the containment). This strategy is still possible with the loss of electrical AC power. The hatches are opened with compressed air from little pressure tanks in the containment and the electrical power for the actuation valves comes from the batteries.

With respect to protecting the containment integrity during severe accidents the other accident management measures described in the sections 6.3.2 to 6.3.5 require electrical AC and DC power. These are the accident management measures described in the following procedures/guidelines, with the exception of the measures which are mentioned above in this section:

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

Furthermore, note that a mobile diesel generator EY080 is available and supply of fuel is foreseen. The connection of an external power supply to the bus bars CX and CW will be made according to the instructions WNE-CX-003 and WNE-CW-003.

6.3.7 Measuring and control instrumentation needed for protecting containment integrity

Diverse instruments for measuring the containment temperature and pressure in different ranges are available. The readings of these instruments are available in the main control room, process computer, emergency control room and for alarm annunciation.

There are two instruments for measuring the H_2-concentration (0 - 10 %) in the containment.

The radiation level in the containment is also measured (0.01 – 1.10^5 Sv/h).

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.

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

The KCB is well equipped with accident management systems to protect the containment integrity. The automatic catalytic recombiners and the filtered venting system are effective design provisions that prevent against high hydrogen concentrations and over- pressurisation of the containment. The SAMGs give necessary guidance to protect the containment and give additional strategies using operational systems.

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

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 (see SAG-4 in Annex 6.1).

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.

6.4 Accident management measures to restrict the radioactive releases

6.4.1 Radioactive releases after loss of containment integrity

6.4.1.1 Design provisions

To mitigate a radioactive release after loss of containment integrity the pressure inside the containment shall be reduced. To do so the following provisions are available:

- containment spray (TJ);
- air coolers inside the containment (TL);
- Containment recirculation filter system (TL037);
- Containment venting (TL003).

6.4.1.2 Operational provisions

The following accident management measures are applicable (see Annex 6.1):

- 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

6.4.2.1 Hydrogen management

The spent fuel pool is located in the containment. This means that there is no possible hydrogen production by oxidation of fuel cladding or molten core-concrete interaction (MCCI) anywhere other than inside the containment. With respect to possible hydrogen generation in the containment, the Borssele NPP has a hydrogen control system consisting of passive automatic catalytic recombiners located in the containment. The system is sized and designed for operation during a severe accident. Hydrogen produced by a zirconium water reaction in the spent fuel pool will also be recombined by the installed PARs.

In addition, with respect to the prevention of H_2 deflagration or H_2 detonation in the containment, the following accident management measures are applicable:

- 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.

For more information see section Management of hydrogen risk inside the containment.

6.4.2.2 Providing adequate shielding against radiation

The following accident management measures to refill the spent fuel pool are applicable:

1. Filling of the spent fuel pool with water from the TJ tanks according to instruction B-TG-01. Note that the TJ tanks are located outside the containment, in the auxiliary building 03;

2. A connection can be made between the demineralised water system TN to the suction side of the pool cooling pump 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;

3. 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 TR system (radioactive waste water system). This accident management measure is currently not mentioned in a separate procedure;

4. 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 spent fuel pool the fuel assemblies will be covered with water for at least 84 hours, see Chapter 5. There are several alternatives to restore the spent fuel cooling for instance:

1. Cooling of the spent fuel pool by the nuclear fuel storage pool cooling system TG / conventional emergency cooling water system VF with water supply by the Low pressure fire extinguishing system UJ. In this way, the regular cooling chain of the fuel storage pool could be reestablished. This accident management measure is mentioned in instructions S-VF-02 and B-VF-04;

2. Cooling of the spent fuel pool by the spent fuel pool cooling system TG with the backup heat exchanger TG080 / backup cooling water system VE, supplemented by the low-pressure fire extinguishing system UJ. If the VE system loses its heat sink (the groundwater wells), the UJ system can take over. This accident management measure is currently not mentioned in a separate procedure;

3. Direct injection of the ultimate heat sink system VE by a UJ pump from the fire-brigade. In the Reserve Supply Building, a connection with a fire-hose is available. In this way the cooling chain spent fuel pool cooling system with reserve heat exchanger TG080 / ultimate heat sink system VE can stay intact. This accident management measure is currently not mentioned in a separate procedure.

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

The spent fuel pool is located in 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 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 atmosphere. This will also give a reduction of the amount of radioactive material that is released in case of containment leakage.

6.4.2.4 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.

6.4.2.5 Availability and habitability of the control room

See section 6.1.3.4.

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

The SAMGs in combination with the containment spray and the installed accident management provisions: automatic catalytic recombiners and the filtered venting system TL003, are adequate measures to restrict the radioactive releases from the containment.

Alternative ways to cool the spent fuel pool and potential measures to enhance the robustness of the spent fuel pool cooling can be found in Chapter 5.

6.4.4 Measures which can be envisaged to enhance capability to restrict radioactive releases

Develop a setoff 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;
- develop check-lists for plant walk-downs and needed actions after various levels of the foreseeable hazards;
- connecting CCB/NS1;
- uncoupling lower rails in time in case of flooding;
- alternative supplies for UJ.

Note that attention should be given with regard to accessibility and conditions under which these measures can still be executed.

Other potential additional measures are:

- 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;
- a possibility for refilling the spent fuel pool without entering the containment would increase the margin to fuel damage in certain adverse containment conditions.

Hydrogen produced by a zirconium water reaction in the spent fuel pool will also be recombined by the installed PARs. Currently no detailed analyses were performed for the hydrogen production in the containment after uncovering the top of fuel in the fuel pool. 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.

Annex 6.1. Severe Accident Guidelines (SAGs), Severe Challenge Guidelines (SCGs) and Severe Accident Exit Guidelines

The Severe Accident Guidelines (SAGs), Severe Challenge Guidelines (SCGs) and Severe Accident Exit Guidelines for KCB are:

SAG-1 Inject into the steam generators

SAG-2 Depressurise the reactor coolant system

SAG-3 Inject into the reactor coolant system

SAG-4 Inject into the containment (outside the Cavity)

SAG-5 Reduce fission product releases

SAG-6 Control containment conditions

SCG-1 Mitigate fission product releases

SCG-2 Reduce containment pressure

SCG-3 Control hydrogen flammability

SCG-4 Control containment vacuum

SAEG-1 Long-term monitoring

SAEG-2 SAMG termination

Strategies for SAG-1: Inject water into the steam generators

The strategies currently in place in this guideline are:

1. Injecting water by the main feedwater pumps.
2. Injecting water by the emergency feedwater pumps.
3. Injecting water by the backup feed water system (the RS-system). Note that in the case of one redundancy of the RS system failing, a connection can be made between the RS pools according to checklist C-RS-109.
4. Injecting water from the RZ pools (pools of the demineralised water supply system) by the emergency feedwater pumps.
5. Secondary side bleed and feed by use of the main steam system RA and the feedwater system RL.
6. Secondary side bleed and feed from one steam generator to the other steam generator through the steam generator letdown system RY.

7. Injecting water from the RZ pools (pools of the demineralised water supply system) by the RZ pumps via the emergency feedwater system.

8. Injecting water from the RZ pools (pools of the demineralised water supply system) by the RZ pumps via the steam generator letdown system RY.

9. Injection of water from the UJ system (low-pressure fire-water supply system) by the fire-water supply system pumps UJ011 or UJ012 via the steam generator letdown system RY.

10. Injection of water by the fire brigade by a high-pressure pump of the fire brigade via the RS system into a steam generator.

11. Injection of water by the fire brigade's low-pressure pump via the RS system into a steam generator.

12. Injection water from the UK system (operational water system) by the high-pressure fire-water supply system pumps UF001/002 via the RS system into a steam generator.

13. Injection of water from the UJ system by the fire-water supply system pumps UJ011/012 via the RS system into a steam generator.

Explanation

The first three strategies mentioned above have priority over the other strategies because they have a higher injection pressure and/or a higher capacity and are automatically activated when an accident occurs.

If strategies 1-3 are not possible, then injection of water from the RZ pools (pools of the demineralised water supply system) by the emergency feedwater pumps or secondary side bleed and feed will be tried (strategies 5 and 6).

If these strategies are not successful, then the low pressure injection strategies are tried (strategies 7-13). Note that strategies 9-13 have less capacity than the other strategies, resulting in a slower water injection. For these strategies it might be necessary to decrease the secondary side pressure in order to inject water in the steam generators. The possible strategies to decrease the secondary pressure are:

1. Steam release by opening of the secondary side relief valves.

2. Depressurisation by using the turbine bypass and the condensers.

3. Secondary side steam removal direct to the feedwater tank.

4. Secondary side steam removal via the turbine-driven emergency feedwater pump RL023.

5. Secondary side steam removal to the steam generator letdown tank.

Strategies for SAG-2: Depressurise the reactor coolant system

The strategies currently in place in this guideline are:

1. Opening of the pressuriser tandem relief valves.
2. Steam release by opening of the secondary side relief valves.
3. Depressurisation by using of the turbine bypass and the condensers.
4. Secondary side steam removal direct to the feedwater tank.
5. Depressurisation by using of the pressuriser spray from the volume control system TA.
6. Depressurisation by using of the pressuriser spray from the backup coolant makeup system TW.
7. Depressurisation by using of the pressuriser spray from primary water YP (if available).
8. Secondary side steam removal via the turbine driven emergency feedwater pump RL023.
9. Depressurisation by using of the reactor vessel head vent line.
10. Depressurisation by using of the pressuriser vent to the relief tank.
11. Depressurisation by using of the vent line of the volume control tank to the pressuriser relief tank.
12. Depressurisation by using of the vent line of the volume control tank to the ventilation stack.

Explanation

The first seven strategies mentioned above (opening of the pressuriser tandem relief valves, opening the secondary side relief valves, using of the turbine bypass and the condensers, steam release to the feedwater tank, using the pressuriser spray) have priority over the other strategies, because these strategies have a higher capacity and/or are automatically activated in case of an accident. Note that the pressuriser spray (strategies 5-7) has less capacity than the other strategies, resulting in a slower depressurisation.

If these strategies are not available then alternative strategies are tried (strategies 8-12).

A decrease in the primary system pressure is, among others important, because lower pressures will allow more injection sources to inject into the RCS and decrease the potential for high-pressure melt ejection (HPME) and creep rupture of steam generator tubes. In the very unlikely case that all strategies for depressurisation are not successfully, a high primary pressure will be maintained during the core degradation phase. In this case the reactor coolant system is jeopardised by a hot steam/hydrogen mixture, which exits the reactor vessel at approximately 1000°C. In this case, the reduced tensile strength at elevated temperatures is likely to result in piping failure in the primary system.

Strategies for SAG-3: Inject into the reactor coolant system

The strategies currently in place in this guideline are:

1. Injection by the high-pressure safety injection system HD-TJ.
2. Injection by the low-pressure safety injection system LD-TJ.
3. Injection by recirculation from the containment sump and the LD-TJ pumps.
4. Injection of water from the accumulators. Note that for injection no pumps are required.
5. Injection of water from the backup coolant makeup system system TW.
6. Injection by water from the volume control system TA (normal path).
7. Injection of water from the volume control system tanks by the LD-TJ pumps.
8. Injection of water from the spent fuel pool by the LD-TJ pumps.
9. Injection by recirculation of water from the containment sump by the TE pumps.
10. Injection of water from the TJ storage tanks by the TE-pumps.
11. Injection of water from the volume control system tanks by the TE pumps.
12. Injection of water from the TJ storage tanks by the volume-control system pumps TA.
13. Injection of water from the TB-tanks (boric acid storage tanks) by the volume control system pumps TA.
14. Injection of water from the TD-tanks (main coolant storage tanks) by the volume control system pumps TA.
15. Injection of water from the TJ storage tanks by gravity drain. In addition it is possible to refill the TJ storage tanks with water from the TD tanks (main coolant storage tanks).
16. Injection of water from the volume-control system tanks TA via the seals of the main coolant pumps.
17. Starting the main coolant pumps YD.

Explanation:

The injection of water to an overheated core will result in the water flashing and heat being removed from the core.

The first six strategies mentioned above have priority over the other strategies because they have a higher capacity and/or a higher injection pressure and are automatically activated when an accident occurs. Note that the injection rate of the TW pumps is approximately 5.5 kg/s for each pump and of the TA pumps is approximately 4.4 kg/s for each pump.

If these strategies are not available then strategies 7-11 are tried. However these strategies have less capacity and/or injection pressure. If these strategies are not available then alternative strategies 12-17 are tried.

The last strategy in SAG-3 for restoring core cooling is starting the main coolant pumps. If water has remained in the cross-over leg, bumping the RCP will force this water to the core region where it can cool the core. Note that bumping the RCPs is only a short-term solution for direct core cooling if the primary system is highly voided. However, other longer-term benefits that are related to core cooling may be realised by bumping the main coolant pumps. First, non-condensable gases that have accumulated in the SG would be driven out by the main coolant pumps, which would benefit natural circulation once the primary system is filled. Second, if the primary system is mostly filled, bumping the main coolant pumps could kick-start natural circulation, which may be hampered by a loss of core geometry. Third, heat removal from a core debris bed or a core with significant blockage would be enhanced by bumping the main coolant pumps due to the increased flow through the core and the potential of the flow to blowholes in the debris bed to cut down on the pressure difference through the core region.

By injecting water at an (over)heated core steam will be produced. Note that the possibility for an extensive in-vessel steam explosion is very unlikely at KCB.

Strategies for SAG-4: Inject into the containment

The strategies currently in place in this guideline are:

1. Injection of water from the TJ storage tanks by the containment spray pumps.
2. Injection of water from the TJ storage tanks by the gravity drain.

Explanation

The Borssele plant has a dry cavity design, with a small free volume, and with very few connections, even for gas flow, to the containment regions. With this design it is expected that, following vessel failure, most of the core debris leaving the reactor system will be retained in the cavity. It is also expected that water accumulating in the containment outside the cavity area will not be able to enter the cavity. Thus, without any intervention, it is expected that molten core-concrete interaction (MCCI) will occur in the cavity in the long-term.

An extensive evaluation of the containment design and geometry was performed to try and identify ways to intervene and inject or spill water into the reactor cavity. The overall result of this evaluation is that no reliable means has been identified by which to ensure water is injected or available in the cavity during a severe accident. There is also considerable doubt about the effectiveness of water to cool ex-vessel debris in such a small reactor cavity. Filling the containment to the 'spill' level is not considered feasible since this would take the water level very close to the filtered vent system discharge location. Together with unreliable level indication at this height, an unacceptable risk of losing filtered vent capacity is associated with this strategy and it is therefore not adopted.

The detailed investigations performed have concluded that the only reliable way to get water into the reactor cavity in the KCB design is via the reactor system after vessel failure, and this is already addressed elsewhere in the SAMGs (see SAG-3). In view of this, SAG-4 is used to control sump water level (i.e. the level outside the cavity) to ensure an adequate level for recirculation capability and cooling any debris which may escape the cavity.

The major benefits that can be realised by injecting water into the containment during a severe accident are:

- cooling ex-vessel core debris outside the cavity;
- scrubbing fission product releases from ex-vessel core debris outside the cavity;
- providing adequate containment water level to allow recirculation.

The strategy at KCB is to inject the water from the TJ storage tanks, either by the containment spray pumps or by the gravity drain.

Strategies for SAG-5: Reduce fission product releases

SAG-5 contains four sections, related to different release paths:

A. Releases from the containment (building 01).

B. Releases from the steam generators (via the secondary side to the environment).

C. Releases from the annulus (building 02).

D. Releases from the nuclear auxiliary building (03).

Section A: Reducing the fission product releases from the containment (building 01)

The strategies currently in place in this section of the guideline are:

1. Use of containment spray.
2. Use of containment fan coolers (TL030/TL031/TL032) and/or

   the containment recirculation filter system TL037.
3. Use of containment biological barrier coolers (TM001/TM002).
4. Use of the TL040-045 and TL111-114 coolers in the containment.
5. Use of the annulus fan coolers (TL050-053, TL055-058, TL060/062/064, TL066/068).

Explanation

In case of containment fission product releases, the spray is used to decrease the containment pressure, to scrub the fission product releases and to allow a good fission product deposition in the containment. The containment fan cooler system TL030/031/032 is used to decrease the containment pressure and to allow the maximum time for the fission product deposition processes in the containment to be effective. The containment recirculation filter system TL037 contains active coal and aerosol filters, which can reduce the activity. If these strategies are not successful then the biological barrier coolers TM, the TL040-45 and TL111-114 containment air coolers or the annulus fan coolers will be used.

Section B: Reducing the fission product releases from the steam generators

The strategies currently in place, to reduce leakage of fission products from the secondary side to the environment, in this section of the guideline are:

1. Injection of water into the steam generators (see strategies for SAG-1).
2. Depressurise the Reactor Coolant System (see strategies for SAG-2).
3. Isolation of the defect steam generator(s).
4. Steam release from the defect steam generator(s) to the condensers.
5. Steam release from the defect steam generator(s) to the feedwater tank.

Explanation

For injection into the steam generators, please refer to the strategies in guideline SAG-1; for depressurisation of the Reactor Coolant System please refer to the strategies in guideline SAG-2.

Alternative strategies to terminate/mitigate the fission product releases from the steam generators are:

- isolating the defect steam generator;
- transferring the steam dump to the condensers or the feedwater tank so as to scrub the fission products from the steam while the SG is depressurised in case of SG fission product releases.

Section C: Reducing the fission product releases from the annulus (building 02)

The strategies currently in place in this section of the guideline are:

1. Isolation or reduction of the leakage from the containment to the annulus by isolation of the leakage path or reduction of the flow in the leakage path.

2. Usage of the annulus fan coolers TL050-TL053.
3. Usage of the annulus fan coolers TL055-TL058.
4. Usage of the annulus fan coolers TL060/TL062/TL064.
5. Usage of the annulus fan coolers TL066/TL068.

Explanation

The first strategy is the isolation or reduction of the leak path from the containment to the annulus, for example by isolating the defect ECCS recirculation path or the defect containment spray path. Alternative strategies are the use of the annulus fan coolers TL050-TL053, TL055-TL058, TL060/TL062/TL064 or TL066/TL068 to decrease the pressure in the annulus.

Section D: Reducing the fission product releases from the auxiliary building

The strategy currently in place in this section of the guideline is:

1. Isolation of the leak path from the containment to the auxiliary building.

Explanation

The strategy is the isolation of the leak path from the containment to the auxiliary building in order to mitigate the fission product releases.

Strategies for SAG-6: Control containment conditions

The strategies currently in place in this guideline are:

1. Use of containment spray.
2. Use of containment fan coolers (TL030/TL031/TL032).
3. Use of containment biological barrier coolers (TM001/TM002).
4. Use of the TL040-045 and TL111-114 coolers in the containment.
5. Use of the annulus fan coolers (TL050-053, TL055-058, TL060/062/064, TL066/068).
6. If after two days the containment pressure has not been decreased to 0.3 bar gauge, use the TL003 filtered venting system to the environment.

Explanation

In case of fission product releases into the containment, the spray is used to decrease the containment pressure, to scrub the fission products and to allow a good fission product deposition in the containment. The containment fan cooler system TL030/031/032 is activated to decrease the containment pressure and to allow a maximum time for the fission product deposition processes in the containment to be effective. If these strategies are not successful then the biological barrier coolers TM, the TL040-45 and TL111-114 containment air coolers or the annulus fan coolers will be used.

In the unlikely event of concrete melt-through (basemat penetration) it is important to have a reduced pressure in the containment in order to avoid fission product releases. Therefore the filtered vent system line TL003 is opened if two days after the start of the accident the containment pressure has not decreased to the control stable state value of 0.3 bar gauge.

A possible disadvantage of reducing the containment pressure is that a hydrogen burn might occur (above a certain hydrogen concentration). To aid in the diagnosis of the severe accident conditions and the selection of appropriate strategies for implementation, graphical computational aids (CAs) have been developed. One of the CAs is CA-6 (hydrogen flammability in the containment), which presents a containment depressurisation limit to avoid any possible hydrogen severe challenge or hydrogen burn when depressurising the containment. This limit will be taken into account.

Note that several major benefits can be realised by reducing the containment pressure:

- reducing fission product leakage from the containment;
- providing a containment heat sink;
- equipment survivability.

Strategies for SCG-1: Mitigate fission product releases

SCG-1 contains four sections, which are related to different release paths:

A. Releases from the containment (building 01).

B. Releases from the steam generators.

C. Releases from the annulus (building 02).

D. Releases from the auxiliary building.

Section A: Terminate/mitigate fission product releases from the containment (building 01)

The strategies currently in place in this section of the guideline are:

1. Use of containment spray.
2. Use of containment fan coolers (TL030/TL031/TL032) and/or

   the containment recirculation filter system TL037.
3. Use of the TL003 filtered venting system to the environment.
4. Use of containment biological barrier coolers (TM001/TM002).
5. Use of the TL040-045 and TL111-114 air coolers in the containment.
6. Use of the annulus fan coolers (TL050-053, TL055-058, TL060/062/064, TL066/068).

Explanation

Non-venting strategies have priority over the use of the filtered vent system. In case of containment fission product releases, the spray is used first to decrease the containment pressure, to scrub the fission product releases and to allow a good fission product deposition in the containment. The containment fan cooler system TL030/031/032 decreases the containment pressure and allows the maximum time for the fission product deposition processes in the containment to be effective. The containment recirculation filter system TL037 contains active coal and aerosol filters, which can reduce the activity. Use of the TL003 filtered vent system is the next strategy. The TL003 filter system is qualified for severe accidents and is highly efficient for fission product releases, except noble gases. If these strategies are not successful then the biological barrier coolers TM, the TL040-45 and TL111-114 containment air coolers or the annulus fan coolers will be used.

Section B: Terminate/mitigate the fission product releases from the steam generators

The strategies currently in place in this section of the guideline are:

1. Injection of water into the steam generators (see strategies for SAG-1).
2. Depressurisation of the Reactor Coolant System (see strategies for SAG-2).
3. Isolation of the defect steam generator(s).
4. Steam release from the defect steam generator to the condensers.
5. Steam release from the defect steam generator to the feedwater tank.

Explanation

For injection into the steam generators please refer to the strategies in guideline SAG-1; for depressurisation of the Reactor Coolant System please refer to the strategies in guideline SAG-2.

Alternative strategies to terminate/mitigate the fission product releases from the steam generators are:

- The isolation of the defect steam generator(s).
- The transfer of the steam dump to the condenser or the feedwater tank to scrub the fission products from the steam while the SG is depressurised in case of SG fission product releases.

Section C: Terminate/mitigate the fission product releases from the annulus (building 02)

The strategies currently in place in this section of the guideline are:

1. Isolation or reduction of the leak path from the containment to the annulus.
2. Usage of the TL012 filtered fan system from the annulus to the environment.
3. Usage of the TL070 filtered fan system from the annulus to the environment.

Explanation

The first strategy is the isolation or reduction of the leak path from the containment to the annulus, for example the defect ECCS recirculation path or the defect containment spray path. Alternative strategies are the use of the TL012 filtered fan system and the TL070 filtered fan system from the annulus to the environment. Note that these filter systems are not qualified as specific for severe accidents, but they might be useful during severe accidents.

Section D: Terminate/mitigate the fission product releases from the auxiliary building

The strategies currently in place in this section of the guideline are:

1. Isolation of the leak path from the containment to the auxiliary building.
2. Usage of the TL013 filtered fan system from the auxiliary building to the environment.

Explanation

The first strategy is the isolation of the leak path from the containment to the auxiliary building. Alternative strategies are the use of the TL013 filtered fan system from the auxiliary building to the environment. Note that this filter system is not qualified as specific for severe accidents, but it might be useful during severe accidents.

Strategies for SCG-2: Reduce containment pressure

This guideline starts when the containment pressure exceeds 6.3 bar gauge. The strategies currently in place in this guideline are:

1. Use of the TL003 containment filtered venting system to the environment.
2. Use of the TL075 filtered fan system from the containment to the environment.
3. Use of the TL010 filtered fan system from the containment to the environment.
4. Use of the TL004 system (vacuum breakers) from the containment to the annulus (building 02).

Explanation

The filtered vent systems are the preferred strategies to depressurise the containment. The unfiltered vent systems are the ultimate alternate methods to avoid containment failure.

Between the filtered vent systems an order of priority can also be established. The TL003 filtered vent system is used first. The TL003 filter system is qualified for severe accidents and is highly efficient for fission product releases, except noble gases. If this strategy is not successful then the TL010 and TL075 filter systems are tried. These filter systems are not qualified for severe accidents. If these strategies are not successful then the TL004 vacuum breaker system will be used.

Strategies for SCG-3: Control hydrogen flammability

Note that the plant has Passive Autocatalytic Recombiners so recombination is already performed through passive recombiner use. Measures to be taken after failure or malfunction of the recombiners are addressed in this guideline.

The strategies currently in place in this guideline are:

1. Opening of the relief hatches between the installation area and the operations area of the containment.
2. Termination of the heat removal from the containment using the following:
    - - Containment spray;
    - - Air coolers TL030-032;
    - - Biological barrier coolers TM001/002;
    - - Coolers TL040-045 and TL111-114;
    - - Annulus coolers TL050-053, TL055-058, TL060-TL064, TL066/TL068.
3. Increase of steam concentration in the containment by opening the pressuriser tandem relief valves.
4. Injection of nitrogen from the accumulators into the containment.
5. Use of the TL003 containment filtered venting system to the environment.
6. Use of the TL010 filtered fan system from the containment to the environment.
7. Use of the TL075 filtered fan system from the containment to the environment.
8. Use of the TL004 system (vacuum breakers) from the containment to the annulus (building 02).

Explanation

Any action which decreases the flammability of the hydrogen mixture in order to decrease the likelihood that combustion will occur is used in this guideline.

The first strategy is the remote opening of relief hatches between the installation area and the operations area of the containment (mixing).

The second strategy is to stop the heat removal from the containment.

The addition of steam or non-flammable gases is an alternative strategy to decrease the hydrogen concentration. The easiest way to do this is by adding steam to the containment by opening the pressuriser tandem relief valves. An alternative method is injecting nitrogen from the accumulators into the containment.

If the above-mentioned strategies are not successful, the filtered venting strategies will then be considered. The filtered vent systems are the preferred strategies and the unfiltered vent systems are the ultimate alternate methods.

Between the filtered vent systems an order of priority can also be established. The TL003 filtered vent system is used first. The TL003 filter system is qualified for severe accidents and is highly efficient for fission product releases, except noble gases. If this strategy is not successful then the TL010 and TL075 filter systems are tried. These filter systems are not qualified for severe accidents. If these strategies are not successful then the TL004 vacuum breaker system will be used.

A long-term concern to take into consideration in case of pressure reduction in the containment is a possible return to the hydrogen severe challenge area if hydrogen concentrations become very high (above 12%, dry measurement). To aid in the diagnosis of the severe accident conditions and selection of appropriate strategies for implementation, graphical computational aids (CAs) have been developed. One of the CAs is CA-6 (hydrogen flammability in the containment), which presents a containment depressurisation limit to avoid any possible hydrogen severe challenge or hydrogen burn when depressurising the containment. This limit will be taken into account. In this case the actions to stay outside the hydrogen severe challenge area are inertising the containment by steam addition or by nitrogen injection from the accumulators.

Strategies for SCG-4: Control containment vacuum

Use of the TL004 vacuum break system from the ringroom (building 02) to the containment.

Termination of the heat removal from the containment using the following:

- containment spray;
- air coolers TL030-032;
- biological barrier coolers TM001/002;
- coolers TL040-045;
- ringroom coolers TL050-053, TL055-058, TL060-TL064, TL066/TL068.

Injection of steam in the containment by opening the pressuriser tandem relief valves.

Explanation

The first strategy to be included in this guideline is the use of the vacuum breaker system. In case of vacuum breaker system failure or malfunction, other strategies to increase containment pressure, like stopping heat removal from the containment and adding steam to the containment, will be considered.

Long-term post-accident monitoring activities

Long-term post-accident monitoring activities are presented in guideline SAEG-1: Long-term monitoring activities.

Furthermore, activities for SAMG termination and long-term post accident activities are described in SAEG-2: SAMG Termination

SAEG-1 and SAEG-2 are summarised below.

SAEG-1 Long term monitoring

The activities currently in place in this guideline are:

1. Identify the equipment/strategies in use to control severe accident conditions
2. Monitor long-term concerns associated with the equipment in use.
3. Evaluate possible recovery actions.
4. Evaluate the need for recovered equipment and the refilling of tanks.
5. Order the control room shift to implement selected recovery actions.

SAEG-2 SAMG termination

The activities currently in place in this guideline are:

1. Identify plant status.
2. Identify ongoing fission product releases.
3. Identify long-term concerns for strategies in use.

Annex 6.2. Severe Accident Management Guideline (SAMG) Training

General

Severe accident management training has been provided to personnel within the plant staff who have been designated for a decision-making and support role in severe accident space. This training had sufficient depth and provided the staff with the ability to make independent judgements on severe accident conditions and appropriate response actions.

The operator training for the SAMGs consists of five phases:

- general introduction courses into the plant specific severe accident phenomena and the use of SAMGs;
- full-scale emergency exercise focused on training the emergency response organisation for SAMG implementation and validation;
- table-top exercises in the usage of the SAMGs focused on training the TAG members in understanding and applying the guidelines;
- table-top exercises in the usage of the SAMGs focused on training the operator in application of the EOPs and SAMGs within the ERO;
- full-scale emergency exercises focused on training the ERO up to and including SAM.

The main purposes of these courses and table-top exercises are to give the operator insight into the structure of the SAMGs, give insight into the strategies as proposed in the SAMGs and give the operator experience in their usage. At the same time the exercises serve as a review of the SAMGs and training in the different responsibilities within the Alarm Response Organisation.

The main areas of operator training are based on the following resources:

- safety evaluation and design-basis accident analyses. From this the behaviour of the plant during design-basis accidents is determined;
- plant simulator for design-basis accident analyses and training;
- full-scope PSA level 3;
- MAAP severe accident calculations;
- RELAP/SCDAP calculations;
- RELAP-SCDAP and MAAP-GRAAPH visualisation of Borssele NPP.

An Excel tool was created to represent the information that the TAG members normally retrieve from the plant status computer. The information is based on safety evaluations, MAAP calculations and design-basis accident analyses. The presented information contains the following information:

- the plant parameters needed to perform the SAMG diagnostics and evaluations, which are the parameters included in the diagnostic flowchart and the severe challenge status tree;
- the parameters which are needed for the availability of major equipment to perform the SAMG strategies.

The information on the screens is updated automatically every 15 minutes.

Furthermore, a RELAP/SCDAP simulator is used for the exercises.

Some more information about the phases of the SAMG training is presented below.

Introductory training

As a first step to the implementation of the KCB SAMGs, an initial one-week training programme for EPZ ERO staff responsible for evaluation, recommendation and decision in SAM space is provided. After that the shift personnel (responsible for implementation) receive similar overview training on the SAMGs.

Full scale exercise for SAMG-implementation and -validation

A full-scale exercise focused on training the ERO and SAMG implementation and validation was organised. The participants of these exercises are:

- the full management team (SED, MB, MOD and MSB);
- the shift personnel;
- the TAG;
- the BOC;
- the (safety) authorities.

The exercise participants are operating in the ACC bunker (the normal work location for the ERO). During the use of the SAMGs, information is communicated both from the control room to the ERO and from the ERO to the control room. For example, the ERO communicates the mitigative action required to complete the implementation steps.

The SAMGs contain strategies which may, under extremely challenging situations of containment integrity, call for venting the containment via available paths – thereby causing a deliberate release. The person in the shelter who is responsible for predicting fission product releases is the Radiation Protection Manager. This person is calculating the consequences of deliberate venting and is taking part in the full-scale exercises.

Table-top exercises

Specific accident scenarios are defined for the table-top exercises which work through specific parts of the SAMGs. There are two types of table-top exercises:

- exercises focused on training the TAG members in understanding and applying the guidelines;
- exercises focused on TAG responsibilities and communication within the ERO.
-
- The participants in these exercises are:
- scenario leader;
- members of the TAG (usually three members);
- the MB (during the second type of table-top exercises);
- a control room shift leader or his second-in-command (optional);
- the national safety authority KFD (optional);
- observers (optional).

Full scale exercise for ERO

Full-scale exercises focused on training the ERO, which include SAM, are held as part of the normal alarm exercises for the ERO.

A full-scale national emergency exercise is focused on training the complete ERO, including the authorities (regional and national) and other organisations that would be involved in a nuclear accident (fire brigades, police, etc.). The exercise participants from the plant's ERO are operating in the ACC bunker. The duration of the exercise is generally from 10 to 15 hours.

The participants of these exercises include:

- the full management team (SED, MB, MOD and MSB);
- the shift personnel (on the full-scope simulator);
- the TAG;
- the BOC;
- the radiation protection group;
- all relevant local, regional and national authorities;
- other organisations that would be involved in a nuclear accident (fire brigades, police, etc.).

Prediction of the Source Term and use of SPRINT

The determination and prediction of the source term is the responsibility of the TAG group at the plants Technical Support Centre in the ACC bunker. The source term is the quantity and characteristics of the release of radioactivity to the environment through available release paths. For this among others the SPRINT (**S**ystem for the **PR**obabilistic **I**nference of **N**uclear power plant **T**ransients) software module is used.

The SPRINT software module estimates the source term in case of an accident at a nuclear power plant. The fast running software module has been developed within the Euratom Framework Programs FP4, FP5 and FP6.

SPRINT uses information on NPP plant status that is deduced from key plant observations using a probabilistic model, known as Bayesian Belief Networks (BBN) analysis. The software module is based on manual input of plant observations and judgments by an operator or analyst, from which the possible final plant states is deduced by probabilistic inference in the BBN. The probabilistic element of the method can also overcome unknown or missing information by resorting to prior probabilities determined by plant experts who set up the model (e.g. by the use of PSA level 1 and 2 information). One of the major benefits of using a probabilistic model (rather than a deterministic model) is that it alerts the user to the existence of alternative possible plant states based on the known and unknown plant observations. Thus the outcome is typically a number of possible alternative plant states each with an associated environmental source term and probability ranking.

A specific model for the KCB NPP in the Netherlands has been developed. For preparation of the SPRINT model of the plant amongst others PSA level 1 and level 2 information has been used. Severe Accident Management measures are also implemented in the model.

SPRINT is used by the TAG group during SAMG exercises (including the use of Accident Management measures) and a full scale national emergency exercise. The source term prediction and training of the different responsibilities within the ERO are part of these exercises.

The use of SPRINT has shown that the group responsible for prediction of the source term is alerted at an early stage of the accident on the existence of a final source term with a low probability, but severe consequences. For this purpose SPRINT is well suited and a supplement on the earlier used method, a decision tree on paper.

SPRINT predictions are also useful in the communication of the possible source term from the ERO at the plant to the authorities and the planning for the authorities of possible emergency measures, especially at the early stage of the accident.

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

FHP-C-1 is entered when core exit temperatures are exceeding 650°C. The restoration strategy described in this procedure consists of four main sets of actions, which are to be executed in the following order:

1. Coolant injection into the primary system.
2. Secondary side feed and bleed.
3. Starting of the main coolant pumps.
4. Primary system bleeding.

a. Coolant injection into the primary system

Several systems are available at KCB to inject water into the primary system. If a system does not function or does not deliver the required flow rate, the next alternative system should be used.

Note that water will also be injected from the accumulators, even when no power is available.

The systems to be used are, in order of use:

1. Low-pressure safety injection system (LDTJ).
2. High-pressure safety injection system (HDTJ).
3. Backup coolant makeup system (TW).
4. Volume control system (TA).

If the core exit temperatures fall below 370°C and the core level is high enough, the procedure exits to the previously active procedure.

If the core exit temperatures stay above 650°C or stabilise above 370°C, the procedure must continue.

Before starting the next step of secondary side feed and bleed, the relief hatches in the containment building between the installation area and the operations area must be opened to improve atmospheric mixing and prevent local build-ups of hydrogen gas.

b. Secondary side feed and bleed

If injection into the primary system is impossible or does not provide sufficient cooling, secondary side feed and bleed is attempted to lower the primary side temperature and pressure to the level at which the low pressure injection system (LDTJ) can be started.

Feedwater can be supplied using a combination of:

- main feedwater system (RL);
- auxiliary feedwater system (RL);
- backup feed water system (RS).

If the feedwater tank level falls to a low level, the demineralised water supply system (RZ) should be started to inject water into the feedwater tank.

Secondary side bleeding can be accomplished by two measures, to be attempted in the following order:

1. opening of secondary motor operated relief valves;

2. opening turbine bypass valves (SF011/012/013) to condenser.

If the core exit temperatures fall below 650°C and primary temperature and pressure is low enough for the low pressure injection system to work, the procedure is exit.

Steam can also be released by using the steam-driven emergency feedwater pump. This method will not provide sufficient cooling and is therefore only to be started late in the procedure, after the main coolant pumps have been started.

If the core exit temperature does not fall or the feedwater supply is not able to keep the steam generator water at a sufficiently high level, the procedure moves to the next step.

c. Starting the main coolant pumps

Starting the main coolant pumps, one after the other, will provide temporary core cooling via the two-phase flow. This can assist in bringing the primary system to a state where primary water injection is effective enough to keep the core temperature low.

As long as the core exit temperature is below 650°C, the procedure will not move to primary bleeding.

d. Primary side bleeding

By opening the pressuriser tandem safety valves (YP011 to YP013), steam is released from the primary system to the relief tank. The safety valves are able to carry a two-phase mixture.

At this stage in the procedure the alarm staff has to evaluate the state of the plant and possibly implement extra measures. If these are not successful, the procedure exits to **SACRG-1** (Severe Accident Control Room Guideline 1).

If the core exit temperature is below 650°C and primary pressure is low enough for the low-pressure safety injection system (LDTJ) to inject water into the primary system, the procedure is exited.

Annex 6.4. Function Restoration Procedure H-1: Actions on loss of secondary heat removal

Procedure FHP-H-1 is entered when the feedwater flow is too low and the water level in both steam generators is not enough to cover the U-tubes.

The procedure consists of the following three sets of actions:

- primary cooling with no need for secondary cooling;
- recovery of secondary cooling;
- primary side bleed and feed.

a. Primary cooling with no need for secondary cooling

If the primary pressure falls quickly below 6 bar, this indicates a large leak from the primary system. In this case heat removal to the secondary system is not effective. The procedure then exits immediately and other relevant actions are carried out.

If primary pressure and temperature are low enough, heat removal from the primary system is possible without secondary side heat removal. Possible measures for this are:

- primary residual heat removal system (TJ);
- backup residual heat removal system(TE).

If primary cooling is achieved, the procedure is exit and the process reverts to the previously active procedure.

b. Recovery of secondary cooling

At KCB, several measures are available to provide feedwater to the steam generator:

- main feedwater system (RL);
- auxiliary feedwater system (RL);
- backup feed water system (RS);
- demineralised water supply system (RZ). For this measure to work, pressure must be decreased in one steam generator to below 20 bar using the main steam relief valves (RA);
- water supply from the feedwater tank to one steam generator while the other isolated steam generator is used to pressurise the feedwater tank;
- use of a mobile pump (fire truck) to inject water into a steam generator, by using the injection point in room 33.201 in the bunkered building 33.

If the feedwater flow rate is sufficient, or if the water level is sufficiently high in at least one steam generator, the procedure exits to the previously active procedure.

If at any point during the procedure the primary side pressure becomes too high or the water level in both steam generators becomes too low, primary side bleed and feed has to be started.

c. Primary side bleed and feed

Water is injected into the primary system through a combination of one or both high-pressure injection pumps (HDTJ) and the backup coolant makeup system (TW).

In order of preference the bleed path is formed by:

- pressuriser tandem relief valves;
- reactor vessel vent valve plus pressuriser vent valve to the relief tank. This should be accompanied by full secondary side depressurisation of at least one steam generator.

Efforts to recover secondary side cooling continue during primary side bleed and feed.

If primary side pressure and temperature are low enough to start the primary residual heat removal system (TJ) or backup residual heat removal system (TE) and these systems are capable of cooling the primary side without further bleeding from the system, the procedure exits to the previously active procedure.

Primary bleed and feed is to be gradually stopped if secondary side cooling is recovered (one steam generator water level is sufficient) and primary side temperatures are falling. If this is successful while maintaining enough cooling capacity, the procedure is exit and other relevant procedures are carried out.

Annex 6.5. Function Restoration Procedure S-1: Actions to restore subcriticality

Procedure FHP-S-1 is entered when the reactor is still critical, despite the fact that reactor scram (RESA) has taken place or should have taken place.

The actions to be taken can be divided into four sets:

1. Checks to be made immediately.
2. Check ATWS.
3. Actions to increase boronisation of the primary system.
4. Actions to be taken if borating does not work or is ineffective.

The actions will be repeated until several instrument readings show that subcriticality has been reached with a sufficient margin.

a. Checks to be made immediately

- reactor scram (RESA). If this has not happened automatically, the following measures have to be performed:
    - give manual reactor shutdown command;
    - physically switch off power to control rods;
- turbine stop (TUSA). If this has not happened automatically, the following measures have to be performed:
    - manual turbine shutdown command;
    - stop turbine using local emergency stop button;
    - stop turbine by closing fire valve (SC011);
    - stop turbine by pressing local button of fire valve.
- check that no cooldown through secondary relief valves (RA) is taking place. If this cooldown has automatically started, it must be stopped to prevent positive reactivity effects through the cooldown of water in the primary system;
- main feedwater pump operating. If this is not successful, the following systems to provide feedwater are to be activated:
    - auxiliary feedwater system (RL021-023);
    - backup feed water system (RS).

b. Check ATWS

The ATWS signal is automatically initiated when the reactor flux is above a certain limit and the control rods are not fully moved into the core. In that case, a series of measures should have started automatically. These are to be checked and, when needed, initiated manually.

If the ATWS signal is not given, the procedure moves directly to actions to borate the primary system.

ATWS measures to be checked:

- main coolant pumps YD are shut down;
- demineralised water supply system is stopped (TB001 & 002);
- volume control system TA is in operation;
- boric acid supply started (TB011 & 012).

If all these measures are active, the primary system is being borated. If this is not successful, the procedure moves to the relevant actions in step 3.

c. Actions to borate primary system

The first measures to be performed attempt to borate the primary system by using the volume control system TA. The TA system can be supplied with borated water via the following means:

- boric acid supply pumps (TB011 & 012);
- from TJ tanks TJ003/004 using the auxiliary suction side of the volume control system TA;
- from TJ tanks using one low-pressure injection system pump LDTJ via the low-pressure letdown of the volume control system LD-reducer.

If the previous measures are ineffective, borated water can be injected by using the backup coolant makeup system TW with a maximum injection pressure of 185 bar.

When the primary pressure is below 117 bar, borated water can also be injected by using the high-pressure injection system pumps HDTJ.

Measures to reduce the primary pressure are:

- main coolant pump sprays (if main coolant pumps are running);
- hot spray of volume control system TA;
- cold spray of volume control system TA;
- spray of backup coolant makeup system TW;
- opening one pressuriser tandem safety valve YP011.

If after these actions the measured reactor flux is low enough and both mid-range and impulse-range reactor periods are negative, the procedure exits to the previously active procedure.

If the measured impulse-range reactor period is not yet negative, actions to borate the system have to be performed again.

If reactor flux is still high or the mid-range reactor period is still positive, the previous measures have not been sufficient to achieve subcriticality and the procedure continues to the next set of actions.

d. Actions to be taken if borating does not work or is ineffective

- ensure that the water level in at least one steam generator is sufficient, or that the total feedwater rate is above 48 kg/s;
- close the demineralised water supply (TB001 & 002) if this has not happened through the ATWS signal;
- identify whether one or both steam generators are defective. If one steam generator is defective, isolate the defective steam generator. If both are defective, the feedwater supply should be minimised, but at least 4kg/s is required to prevent thermal stresses in the steam generators and to remove the decay heat.

If at this point, after previous actions and checks, the core exit temperatures are above 650°C and rising, the procedure exits to the Severe Accident Control Room Guideline **SACRG-1**.

Actions to borate the system and to identify sources of positive criticality should continue until reactor flux is low and reactor periods are negative. At this point the procedure exits to the previously active procedure.

Annex 6.6. Emergency Operating Procedure ECA-0-0, Actions in case of loss of auxiliary power

This procedure is entered into when no voltage is available on both 6 kV AC auxiliary power busses (BU & BV). This implies the loss of external power in combination with the loss of Emergency Grid 1's diesel generators.

The procedure consists of the following three stages:

1. Checks to be performed immediately.
2. Actions to restore emergency power.
3. Actions to be performed when emergency power is not restored.

The accident management measures mentioned in this procedure apply for a LOOP or SBO situation. For a more extensive evaluation of the LOOP-SBO scenario for KCB, together with a description of the measures to be performed in stages 1 and 2, please refer to Chapter 5. In this chapter, only the accident management measures to be performed when emergency power is not restored (stage 3) are described. In this stage, the procedures ECA-0-0 transits into the Severe Accident Management Guidelines (SAMGs) when the core exit temperature reaches 650°C and is still increasing.

a. Checks to be performed immediately

These checks are described in Chapter 5.

b. Actions to restore emergency power

These actions are described in Chapter 5.

c. Actions to be performed when emergency power is not restored

- check 400 V power system (CW & CX). The 400 V system powers the systems used in the next steps. The options to restore power (from instruction **S-EY-02)**, to be attempted for both 400 V busses in turn, are:
  - start diesel generator (EY040 & 050);
  - connect to 10kV system of neighbouring coal-fired power plant (CT23 or CT24);
  - connect to emergency power bus of KCB (Emergency Grid 1);
  - connect to external mobile power generator EY080;
- ensure secondary side cooling using the backup feed water system RS. The bunkered water pools of RS need eventually to be cooled, either through:
  - fire water supply system UJ, or
  - emergency cooling water from deep well pumps VE;
- maintain sufficient water level in at least one steam generator. Measures available are:
  - turbine-driven emergency feedwater pump RL023;
  - backup feed water system RS. Bunkered injection water tanks of the RS system will be kept filled by the fire-water supply system UJ or by tanker trucks;
- ensure primary volume control through the backup coolant makeup system TW. Bunkered injection water needs to be resupplied;
- isolate the primary system. All isolation valves in the primary system need to be checked and closed;
- check cooling of the spent fuel pool. If the pool temperature exceeds 70°C, start cooling the spent fuel pool by using spent fuel pool cooler TG080 and the VE pumps.

If at this point the core exit temperature reaches 650°C and is still increasing, the pressuriser tandem safety valves are opened and the procedure exits to Severe Accident Control Room Guideline 1 (**SACRG-1**).

Otherwise attempts to restore emergency power have to continue after consultation of the ERO.

Annex 6.7. Function Restoration Procedure Z-1:

Actions in case of containment overpressureThis procedure is entered into when containment pressure exceeds 3.9 bar gauge. The goal of the procedure is to reduce containment pressure below 3.4 bar gauge.

The procedure contains three main sets of actions:

1. Containment spray.
2. Use of the containment filtered vent.
3. Actions once pressure is reduced.

a. Containment spray

- start containment spray TJ061;
- resupply TJ tanks when tank level is too low.

If the spray is not effective in reducing pressure, the procedure continues by opening the filtered containment venting system.

b. Use of the filtered containment venting system

Before a release to the environment is started, the following actions have to be performed:

- switch the air supply of the main control room to filtered operation;
- consult with the alarm staff and safety authority;
- check the water level in venturi filter TL003 and, if needed, adjust the level.

Pressure is decreased using the TL003 filtered containment venting system to the environment, until pressure falls below 3.4 bar. Note that the TL003 filter is designed for severe accidents and is highly efficient in filtering fission product releases, with the exception of noble gases.

c. Actions once pressure is reduced

- attempt to localise and isolate the leak in the primary or secondary systems. If not successful, postpone to a later time;
- check all containment isolation valves and put them in the right position if necessary;
- open the relief hatches in the containment building between the installation area and the operations area to improve the atmospheric mixing. This will prevent local build-ups of hydrogen gas pockets.

Chapter 7 Other extreme hazards

7.1 Introduction

By special request of the Dutch Ministry of Economic Affairs, Agriculture and Innovation, EPZ has identified a number of other extreme hazards to be addressed for the nuclear power plant KCB in connection with the ENSREG stress test. These hazards include unintentional hazards only, and not intentional hazards like sabotage.

First, a group of explosions and fire-related hazards have been identified, which are discussed in detail. Another possibility is an airplane crash and this is discussed in a separate section. There is also the possibility of toxic gases being released on the KCB premises; these are discussed separately as a special consequence of an explosion or fire.

Two electrically related issues are also discussed: a large grid disturbance and a failure of systems by introducing computer malware.

Finally, two water-related issues are discussed: internal flooding and a blockage of the cooling water inlet. External flooding is addressed in Chapter 3.

7.2 Internal explosion

7.2.1 General description of the event

7.2.1.1 Design basis

Internal explosions are defined as being explosions that originate from plant systems and storage areas.

To protect the safety-relevant systems against the impact of internal explosions, the following measures were included in the design:

- apply, as far as possible, fire-resistant or inflammable gases and liquids instead of more obvious but combustible ones;
- reduce the number and volume of explosive materials;
- limit the release of explosive materials in case of disturbance;
- subject the storage of explosive materials to special precautions;
- monitor risk areas combined with automatic safety measures;
- ventilate risk areas.

Special attention was paid to the following parts of the installation.

A Gas and compressed air supply system (TP)

The gases for the TP system are stored in a dedicated separate building (26); supply lines are partly provided with leak detection systems.

The gas and compressed air supply system provides hydrogen gas to:

- the Volume control system (TA), to control the pressure in the volume control tank and to provide a very low oxygen concentration in the primary water;
- the Radioactive gas treatment system (TS), to maintain the correct balance between hydrogen and oxygen in this system;
- the Generator cooling supply system (ST), for supply to the generator cooling system.

B Cooling of the generator (ST)

The hydrogen cooling system of the generator includes the required precautions to protect the system against leakages and explosions (ATEX-zone). These are the storage of H2 in a special building and using automatic isolation valves in case of leakage, which reduce the amount of H2 that can be released. In case of leakage, the hydrogen is not contained in the storage building due to open ventilation on the roof. Hydrogen leakage detection is included in this system.

C Radioactive gas treatment system (TS)

The radioactive gas treatment system is designed to:

- maintain under pressure the tanks in which hydrogen (by radiolysis) and radioactive gases (released from primary water) are set free and collected;
- remove all released and collected radioactive gases after a certain period;
- maintain low H_2 and O_2 concentrations to prevent an explosive mixture and corrosion occurring by applying a recombiner.

D Volume control system (TA)

The volume control tank is placed in a dedicated room that contains no other equipment or ignition sources. The supply is manually controlled.

E Transformers

The main transformers are protected against unacceptable temperature increases, overloads, etc. by means of temperature control, differential guarding, overcurrent protection and buchholz relais in such a way that disturbances, which may lead to explosions, will be prevented. The transformers under consideration are:

- step-up transformer AT000;
- house-load transformer BT000;
- start-up transformers BS001 and BS002;
- low-voltage transformers.

With the exception of the low-voltage transformers, all the above transformers are protected by a sprinkler installation.

If a fire or explosion occurs in one of the high-voltage units, there will be no damage to the reactor system because these units are placed in closed areas outside the reactor building, which are designed to withstand these events.

F Others

A special explosion source could be a mobile gas cylinder, like one on a cart with welding equipment.

7.2.1.2 Beyond design conditions

For the assessment of the beyond design conditions, it is assumed that the amount of explosive material present will be released either nearby or in a containing area and deflagrate.

The buildings that would be affected by an internal explosion event are:

- nuclear auxiliary building (03)
- turbine building (04)
- electrical building (05)
- diesel generator building (10).

7.2.2 Potential consequences for the plant safety systems

In a detailed analysisthe buildings relevant to this event were identified, together with the safety systems they are housing. The extent that the event can impact the building itself and redundancies that are placed in the building were considered first. Next, the loss of combinations of buildings was examined, especially those combinations which would lead to the unavailability of at least one of the three cooling functions: cooling down, decay heat removal and spent fuel pool cooling. In general, it can be stated that failure of those building combinations is hardly possible due to their spatial distribution.

The following 'explosion' sources were identified as having a possible severe impact on safety (related) SSC:

- **Cooling of the generator (ST) in building 04: If, in spite all the measures taken (ATEX** zone, leak detection, non-enclosed area, free ventilation through the roof of the building), an explosion occurs, the worst-case scenario would be a loss of building 04 in combination with buildings 05 and 10. However, the loss of these buildings will not lead to the loss of any of the three cooling functions.
- **Volume control system (TA) in building 03: The explosion hazard from the Volume** control system originates from the volume control tank in room 03.227. This is an enclosed room with heavy concrete walls, floor and ceiling. The wall facing building 02 is double-thickness and the room has two exterior walls. No safety-related equipment is kept in the surrounding rooms. The amount of hydrogen is limited to approximately 7 m^3 at 2 bar. There is no continuous flow from the supplementation system TS, so in case of leakage the pressure will drop and the flow will stop. A drop in hydrogen pressure is indicated in the control room and the pressure is restored manually at the location by an operator. The room contains no ignition sources. Given the lay-out of the building an explosion will affect only a small part of building 03, resulting in a loss of TA. TW is still available as backup. Given the combination of buildings that have to be lost before loss of cooling occurs, an explosion in room 03.227 would not cause a problem.
- **Radioactive gas treatment system (TS):** The parts of the radioactive gas treatment system in the relevant buildings contain a very limited amount of hydrogen, thus a leakage would have a minor effect. However, there is no leak detection in the system so hydrogen could collect unnoticed in certain areas.

The following can be concluded with regard to the three cooling functions:

- cooling down: both TW and RS will remain available;
- decay heat removal: the reserve cooling chain TE/VE will remain available;
- spent fuel cooling: the reserve cooling chain TG/TG080/VE will remain available.

Regarding the mobile explosive source like the welding equipment, it can be stated that this is covered by the previous analyses, as only one room will be affected.

This shows that the design features, e.g. spatial separation of redundancies of safety-related systems, methods to prevent fires and explosions and reduce their impact to a minimum, provide sufficient protection if an internal explosion occurs. It is also concluded that the extent and range of these events cannot be so large that the plant would lose its cooling functions for the reactor and spent fuel pool. Therefore the final conclusion that can be drawn is that the plant is well equipped to deal with this event, both for design basis and beyond design conditions.

However, the findings of this assessment indicate that hydrogen leak detection is not in place for some systems and some areas of the plant. Although these are small amounts, the leaking hydrogen might collect unnoticed in some areas and cause hazardous situations.

7.3 External explosion

7.3.1 General description of the event

Pressure waves from an explosion (EPW) may generally result from an accident in the industrial environment, from pipelines or from an accident on a nearby road or railway or the river. The resulting risks for these events have been evaluated in the PSA for external events and were found to be very low. Four causes of pressure waves can be identified: industrial facilities, road accidents, railway accidents and accidents with shipping.

- **Explosions at nearby industrial facilities**

In the framework of the PSA, the area within a 10 km radius of the plant site has been selected to evaluate industrial and military facilities The only military facility in this range is the ammunition depot Ritthem at a distance of 5.5 km. Accidents in this depot cannot cause a challenging pressure wave at the plant site.

Regarding industrial activities, the Province of Zeeland has published a Risicokaart[60] (risk map) of the province. The map shows that the 1.10-6/y risk contours of all industrial activities are well away (at least 900 m) from the KCB site. Based on this information it is concluded that the risk from pressure waves from industrial activities is negligible, and accidents in nearby industrial facilities cannot cause a challenging pressure wave at the plant site.

- **Accidents on nearby roads**

The NPP Borssele is located approximately 7.2 km from the major A58 road. At this distance, there are no chemicals that could cause a problem to the plant site. Transportation on the local road (Europaweg) of an explosive chemical will present a potential explosion risk to NPP Borssele but only when the transport vehicle passes within a range of 340 m from NPP Borssele (a shock wave of 0.1 barcan be generated within a distance of 340 m). As the nearest point of the local road to the plant is 500 m, the potential risk to NPP Borssele from explosions of materials transported on this road is negligible

Materials transported to NPP Borssele do come within closer range of the plant, however the trucks used for transport have a much smaller capacity (45 m³) than the storage tanks on site. Therefore, the risk from transported chemicals does not create an additional risk.

- **Railway accidents**

Overpressure waves of 0.1 bar can only result from rail car explosions within 490 m. However, LPG and butadiene are transported at a distance of more than 1,500 m. No other explosive chemicals are transported within this range, thus there is no risk of damage to the Borssele NPP from pressure waves caused by railway accidents.

[60] http://nederland.risicokaart.nl/risicokaart.html?prv=zeeland

- **Shipping accidents**

The nuclear power plant KCB is located on the shore of the Westerschelde estuary, which contains the main shipping lanes into and out of Antwerp, Ghent, Terneuzen and Vlissingen, with intensive traffic of (mainly sea-going) ships. The nearest shipping channel is located approximately 1,500 m from the shore where the plant is located. Between 40,000 and 50,000 ships pass through this estuary each year. Ship accidents with the potential to cause damaging effects to life or property on land are mainly expected from gas tankers carrying cargos of flammable gases (e.g. LPG, LNG, butylene, etc.) or toxic gases (e.g. ammonia). The cargo tanks of these ships usually contain the substances in liquefied form, either by compression or by refrigeration

Collisions of ships on the Westerschelde estuary carrying large quantities of toxic or flammable gases may cause two effects that are relevant for KCB plant safety: spreading toxic gases on to the KCB premises, and an explosion causing fire and high waves to affect the KCB premises. Another possibility is an oil spill on the Westerschelde with subsequent fouling to the cooling water intake systems. The effects of toxic gases are described in 7.6. The explosion effects are described below.

Of all the flammable materials transported along the shipping channel, the quantity of LPG is the greatest. This description will therefore focus on LPG shipments.

LPG tankers are generally well protected from tank ruptures. However, a collision with another vessel in the shipping lanes could result in the rupture of one or more cargo tanks. If the vapour does not ignite due to the energy and sparks from the collision, a vapour cloud could form and drift toward the Borssele nuclear plant. Therefore two general types of scenarios are identified

- immediate ignition: a pool of LPG (liquid) is released at the accident location and is ignited immediately, leading to a pool fire, boiling liquid expanding vapour explosion (BLEVE), flash fire or explosion;
- delayed ignition: a quantity of LPG is released at the accident location and is not immediately ignited; a vapour cloud forms and may be ignited after a delay.

Both types of scenarios have two physical effects: thermal radiation, which may disable personnel and set fire to buildings, and pressure waves, which potentially damage the plant and injure personnel.

In order to gain insight in the external explosion event regardless the probability of causes, four relevant scenarios could be envisaged. The external explosion could be assumed to destroy:

1. the cooling water inlet building (21);
2. the emergency diesel generators (building 72) by an EPW coming over the dike from the eastern sea direction;
3. the backup systems bunker (33) and remote shutwdown building (35) by an EPW coming over the dike from the western sea direction;

4. the transformers and emergency diesel generator (building 10) by an EPW coming from inland.

These scenarios will be elaborated in the next section.

An oil screen is available on the EPZ premises in case of an oil spill, and preparations have been made to attach this in the cooling water inlet channel. Instructions are available In case the screen appears not to be sufficient, an additional larger screen can be obtained from the regional waste processing company Delta Milieu.

7.3.2 Potential consequences for the plant safety systems

Defence against explosion pressure waves is provided by a deterministic protection concept in which a loss of all SSCs not protected by qualified structures is conservatively assumed. As mentioned above, a shockwave of 0.1 bar (maximum overpressure due to reflection 0.15 bar) has been chosen as the design-basis load case. The deterministic protection concept ensures that all essential safety functions necessary with respect to the fundamental safety functions are possible due to safeguards in EPW-protected buildings.

7.3.2.1 Safety systems and building arrangements

These essential safety functions have been defined in a functional analysis considering both power and shutdown states. They encompass the following:

- Reactor Coolant Pressure Boundary (RCPB) isolation, Reactor Coolant System (RCS) inventory make-up
  - integrity of the RCPB;
  - closure of the RCPB isolation valves;
  - automatic injection of borated water to the RCS with the bunkered injection system TW;
- Shutdown of the reactor, insurance of long-term subcriticality
  - Fast negative reactivity insertion due to SCRAM function;
  - Long-term subcriticality injecting borated water to the RCS with the bunkered injection system TW;
- Decay heat removal
  - feedwater supply to the steam generators with the reserve decay heat removal system RS;
  - manual secondary cooldown to residual heat removal conditions;
  - long-term decay heat removal from both reactor pressure vessel and spent fuel pool (SFP) using the backup cooling chain (TE/VE, TG 080/VE);
- Containment isolation
  - containment isolation is only required in case of additional failures, especially a break in the piping that connects it to the RCS. The governing scenario would be a rupture of the chemical volume and control system letdown line between the second RCPB isolation valve and the high pressure cooler. The relevant lines penetrating the containment wall and potentially leading to significant radioactive releases are the supply and exhaust lines of the containment ventilation system (TL 004, TL 010, TL 075).

TW and RS are engineered safeguards that are actuated automatically, whereas the reserve cooling chain TE/TG080/VE is actuated manually. For the latter case, emergency procedures are available in both the main control room and the emergency control room. The proper use of these emergency procedures is covered in regular simulator training with the shift team.

The SSCs needed to support these functions are installed in the following structures protected against EPW:

- reactor building with annulus (01, 02);
- backup systems bunker (33);
- remote shutdown building (35);
- wells of the VE system.

7.3.2.2 Consequences of external explosion

IAEA regulations state that any nuclear power plant should be designed against the impact of pressure waves from an explosion.

Analysis of LPG shipping accidents by TNO shows that immediate ignition has no impact on the KCB site, neither from the pressure wave nor from the thermal radiation. The pressure wave is below the 0.1 bar limit for damage to plant buildings and equipment. Heat intensities are below the 10 kW/m2 threshold for damage to buildings. Personnel in the control room are not affected; however serious heat burns can occur on unprotected personnel present outside. Also, any eventual large wave will be too limited for serious impact, as the explosion will occur above the water surface.

In case of delayed ignition, a vapour cloud could float towards the KCB premises and may be ignited after a delay. LPG is heavier than air so therefore the dyke between the Westerschelde and the KCB forms a barrier. However, the cloud (or part of it) might get over the barrier if it moved faster and was perpendicular to the dyke. To further reduce the probability of an explosion on site, an automatic detection and ignition system is placed at a safe distance from the KCB site The igniters are placed at two locations on the seaward side of the dyke in front of the plant: at the cooling water inlet and the cooling water outlet channels. The result of this measure is that the shock wave impact after an ignition on the buildings will be limited and will cause no structural damage. Although the pressure wave will stay below the 0.1 bar threshold (no damage), the heat wave will cause damage and disable personnel. In that case the consequence of this event will be the same as for the event described in section 7.6 (toxic gases), except that this event will be very local and short term. The plant will always take care of itself during the autarky period of ten hours, in which the transition to a hot shutdown state will be made After this, personnel will again be available to bring the plant to a cold shutdown state, if necessary from the emergency control room.

The design of the EPW-protected KCB buildings is based upon incident and reflected pressures of 0.1 bar and 0.15 bar respectively. Building 33 is designed to withstand an even more severe load case, i.e. 0.3 bar / 0.45 bar. The analysis of available margins shows that load cases of 0.3 bar / 0.45 bar (BMI guideline) can be mitigated with confidence for the other EPW-protected buildings (01, 02, 35) as well. Even higher margins are demonstrated (load case of 0.36 bar / 0.54 bar) for the reactor building and annulus (01, 02).

Significant margins are further established by conservative modelling techniques, assumptions and boundary conditions, as well as a conservative choice of material data. A detailed description of the different margins has been given in Chapter 2 on earthquakes. These margins are also applicable for EPW. There are no cliff-edges, meaning that a small increase in the EPW load of more than 0.3 bar / 0.45 bar would induce a 'severely abnormal plant behaviour' or disproportional release of activity

With regard to the four scenarios introduced in the previous section, the following can be said regarding the availability of safety systems.

- Scenario 1: In case of destruction of the cooling water inlet building 21, cooling down of the plant would be ensured by the backup cooling systems TE/VE for the reactor and TG/TG080/VE for the spent fuel pool.
- Scenarios 2 and 3: In case of destruction of the emergency diesel generator building (72) or buildings 33 and 35, the plant will be shut down normally as both cooling systems and grid power are still available.
- Scenario 4: In case of an EPW coming from the inland, destroying both the transformers on site and those connecting the site with the 150 kV grid, a LOOP situation develops. The impact on the safety systems is elaborated in Chapter 5 (LOOP and LUHS) of this report.

With regard to induced vibrations, the design against earthquake and airplane crash provides sufficient reserves which are also applicable for induced vibrations from EPW. The KCB earthquake response spectra may not necessarily cover EPW in the high-frequency domain. However, mechanical equipment is generally not vulnerable to high-frequency vibrations characterised by small displacements. For I&C and electrical equipment, a vulnerability against high-frequency accelerations cannot be precluded by default. However, equipment similar to that at KCB has been qualified for demanding floor response spectra in German NPP projects.

Therefore the final conclusion that can be drawn is that the plant is well equipped to deal with this event for both design basis and beyond design conditions.

7.4 Internal fire

7.4.1 General description of the event

7.4.1.1 Design basis

In dealing with fire safety in the design, there is a prerequisite to maintain the basic safety functions, which are:

- shutdown;
- residual heat removal;
- confinement of radioactive materials.

Based on the inclusion of redundancies and physical and spatial separation, as well as fire mitigating measures, these functions will be assured

To cope with this prerequisite, a 'defence in depth' approach is applied. This includes the following steps:

1. Prevention of fires;
2. Detection of fires and fire-fighting;
3. Containment of fires.

This approach is completed by including the fire-fighting organisation.

For these steps the following applies:

1. Prevention of fires

- The number and amount of combustible and caustic materials are reduced to a minimum.
- Special precautions are introduced for storage of these materials.
- Ignition sources are isolated or avoided.
- For locations with large fire loads (e.g. H_2 cooling of the generator) ATEX zones are established.

2. Detection of fires and fire-fighting

Detection systems are installed in relation to the fire risks per room. This means that numbers, types and distribution of detectors are related to the concentration and kind of fire-ignition sources and (type of) combustible materials.

In relation to this, fire-fighting provisions (automatic/manual equipment, water supply or alternative fire-extinguishing methods like water-spray, inergen and CO2 systems) are installed.

3. Containment of fires

Depending on the areas that the system redundancies are placed in, a distinction is made between the 'containment approach' and the 'impact approach'

For the containment approach, usually one train of a redundant system is installed in an area where a fire can be isolated from the rest of the plant (the fire-compartment). This area is surrounded by fire-resistant and fire-retarding materials so that fire will be confined to that area for at least one hour (F60 demand). This isolation also includes closing fireproof doors, closing shutters or shutting down the ventilation system and applying fire-proof cables and cable penetrations. Sometimes, large parts of a building are included in one fire compartment. This compartment can be divided into separate fire cells that meet the same F60 condition – not by physical means but by distance.

With regard to the impact approach, redundancies cannot or are not being installed in separate compartments. This area is then divided into more or less open cells that contain the separate redundancies. The fire is prevented from spreading from one cell to another by spatial separation (distance) and by additional (automatic) means for fire detection, retardation, suppression and fire-fighting. This again ensures, in case of a fire, the availability of at least one of the redundancies being present in one area for at least F30 demands (30 minutes availability).

Using this overall approach the risk of fire is minimised, and fire-fighting and confinement is optimal for the design-base event ‘internal fire’. The design meets the requirement that, due to its redundancies, the basic safety functions are assured in case of fire.

7.4.1.2 Beyond design conditions

In contrast to the design basis in which fires will be limited to sole areas, beyond design conditions are defined to be those conditions that occur when fires spread to other compartments or buildings.

The following buildings are affected by the internal fire event:

- reactor building (01);
- annulus (02);
- nuclear auxiliary building (03);
- turbine building (04);
- electrical building (05);
- diesel generator building (10);
- cooling water inlet building (21);
- backup systems building (TW/RS) (33);
- remote shutdown building (35).

7.4.2 Potential consequences for the plant safety systems

In a detailed analysis the buildings relevant to this event were identified, together with the safety systems they are housing. The extent that the event can impact the building itself and redundancies that are placed in the building were considered first. Next, the loss of combinations of buildings was examined, especially those combinations which would lead to the unavailability of at least one of the three cooling functions: cooling down, decay heat removal and spent fuel pool cooling. In general, it can be stated that failure of those building combinations is hardly possible due to their spatial distribution. The simultaneous destruction of such combinations could be regarded as cliff edges.

The impact of the internal fire event on the three cooling functions (for combinations of buildings) will be considered. First, the impact of the unavailability of the remote shutdown building (35) is discussed.

Two situations can be identified for all three cooling functions when the remote shutdown building (35) is not available:

1. The emergency control room is not available.
   This is not a problem since the main control room will be available.
2. The control systems are affected.
   The redundancies are installed in separate compartments, designed to contain fires in the dedicated room. Ignition sources are limited, as is the amount of combustibles, and all fire compartments have a F90 rating. This means that in case of fire it will be very unlikely to lose both redundancies. Furthermore, the majority of equipment can be controlled locally or from the main control room.

7.4.2.1 Cooling down

Scenarios which lead to a complete loss of cooling-down capacity necessitate independent fires in at least two buildings. When taking into account that building 33 in fact contains two separate buildings, one for each redundancy, all but one of these scenarios require at least four fires.

Now turning to the fires themselves, it should be noticed that none of the buildings in each scenario are next to each other. Furthermore, a fire should spread inside the buildings to at least two fire compartments that are designed to contain the fire. This is true for buildings 01, 02, 04 and 33. It is only in buildings 05 and 10 that several specific combinations of one fire cell per building exist, thus disabling the electricity supply for CU, CV, BU and BV. However, achieving this would require a fire spread over at least two fire compartments in another building.

The protection against fire beyond design conditions is therefore adequate.

7.4.2.2 Decay heat removal

Decay heat removal fails when either building 02 or building 33 is destroyed. Decay heat removal is lost when TF or TJ fails in combination with TE. As all systems in building 02 are at least twofold redundant and all redundancies (including cabling) are placed in separate fire compartments with at least an F60 rating (see design base) in order to contain the fire, a minimum of four independent fires is needed or two fires have to spread over more than one compartment.

Building 33 consists of two separate sub-buildings, both protected against fire. Each of these sub-buildings contains one of the redundancies of the systems under consideration, including their cabling. Loss of both sub-buildings by internal fire requires two separate fires.

All the other scenarios require fires in more than two buildings that are not next to each other.

The protection against fire beyond design conditions is therefore adequate.

7.4.2.3 Spent fuel pool cooling

For a loss of spent fuel pool cooling, the possible scenarios match similar scenarios of a loss of decay heat removal. For these scenarios the same rationale is valid, so the conclusion is that the protection against fire beyond design conditions is adequate.

In addition, loss of spent fuel pool cooling will occur when the combination of buildings 04 and 33, which encloses two separate sub-buildings, suffers fire. This needs three internal fires spread over these three buildings.

The conclusion is that the design, with its included features of spatial separation of redundancies of safety-related systems and the means provided to prevent fires and explosions and to reduce their impact to a minimum, copes very well with these events for design conditions. It is concluded that the extent and range of these events cannot be so large that the plant will lose its cooling function for reactor and spent fuel pool completely. Therefore the final conclusion that can be drawn is that the plant is well equipped to deal with this event for design basis and beyond design conditions.

7.5 External fire

7.5.1 General description of the event

There is the possibility of external fires originating from the neighbouring coal-fired power station CCB, from other industries located on the nearby Sloe industrial estate, and from ships or trains carrying flammable material on the nearby transport routes. A special cause of external fire is an airplane crash in the area around the nuclear power plant. Chapter 7.5 discusses the effects, including fire, in case of an airplane crash at the NPP.

The transport of flammable material by ship, rail and road is covered in Chapter 7.2 on external explosions.

Of the industries on the Sloe industrial estate, only the Total refinery may be of relevance. However, this is located 1.5 km away from the plant, a distance large enough to exclude fire hazard from this source at KCB. However, a special case may be the relatively new natural gas pipeline from Woensdrecht to the Zeeland Seaports industrial area, which passes the nuclear plant within a few hundred meters. An external safety study has been executed which mentions the nuclear plant but focuses on population density-related local risk and group risk only.

Fires on the adjacent coal-fired power station CCB may originate on the coal storage site, the biomass storage site or the coal bunker. With regard to the nuclear power station KCB, the coal storage site is more than 700 m away, behind a dyke and the CCB building itself. The biomass storage is closer, at about 100 m, but is in four closed silos. The coal bunkers are also in a closed facility, at about 50 m distance. A fire starting in one of these two areas may be caused by either dust explosions or mowburn. As dust explosions can cause severe fires, safety precautions have been taken in the form of detection and repression. Also explosion hatches are installed, precluding pressure build-up with deflagration.

Mowburn is a slow process so a sudden large fire can be excluded. In case of mowburn in the coal bunkers, the supply is interrupted and the coal is quickly burnt in the boiler, or it is ejected through a hatch, after which it can be extinguished on the ground. If mowburn occurs in the biomass storage, the stock is removed.

Both KCB and CCB have on-site fire brigades, each in a separate location, and in case of additional demand (large fires), support can be obtained by fire brigades in the vicinity: Borssele and Nieuwdorp. Besides the usual fire-fighting equipment, a crash tender is also available. All these fire brigades are trained for hazardous situations on the industrial sites, and they also attend an annual training programme within the controlled area on the KCB site.

7.5.2 Potential consequences for the plant safety systems

The effect of elevated temperatures or coal cinders hurled up into the air is too small to be of any significance.

If there is a leak of natural gas from the nearby pipeline, ignition sources exist on the sites of Total, COVRA and the coal-fired power plant CCB. Additionally, the orientation of KCB is such that if there is a major fire following a natural gas leak, the plant would be shielded from the heat by the CCB buildings to a large extent.

Smoke formation may affect the breathability of the air on site. The control room, however, is protected against toxic gases and smoke: detectors for these substances are installed, and will initiate occlusion of the control room from the surroundings and recirculation of the air Severe smoke formation may also cause problems for the air intakes of the emergency diesel engines. However, the diesel engines are in three different locations on the premises. In the backup systems bunker (33), air intakes occur on two different sides of the building. Therefore a total unavailability of the emergency diesels may be excluded.

In a very extreme fire event, smoke formation may result in a precipitation of soot on the isolators of the transformers and electric equipment located outside. This may lead to damage to these components, eventually resulting in a loss of off-site power (LOOP) event. Chapter 5 provides a description on how the plant can be shutdown safely, exiting from a LOOP situation.

In conclusion, the plant is well equipped to handle this event safely.

7.6 Airplane crash

7.6.1 General description of the event

In the initial plant design, an airplane crash (APC) was not considered as a relevant load case since the site is located outside the direct influence range of commercial and military airports. This is confirmed by PSA studies like that showed that the probability of this event is well below those of the various design-basis accidents, and therefore is regarded as a residual risk.

Accidents resulting from an aircraft impact can be broken down into classes based upon the aircraft size, mass, speed and usage. On the one hand, there are small and medium-sized commercial and private aircraft versus large commercial airliners, and on the other hand, there are the military aircraft, which are also classified as lighter and heavier.

7.6.1.1 Design basis

As the Midden-Zeeland Airport is located only about 10 km from the plant, the crash from a private aircraft was introduced as an external hazard to be considered in the second 10 yearly safety evaluation in the mid-Nineties. Taking into account that this airport serves only light aircraft (less than 5.7 ton) the crash of a Cessna 210 has been considered as a corresponding load case. Considering the characteristics of air traffic at the site, this choice is still judged to be reasonable.

7.6.1.2 Beyond design conditions

In the wake of the 9/11 events in the USA in 2001, more demanding load cases have been introduced in analyses for foreign nuclear power plants, which are similar to the design basis of nuclear power plant Borssele (KCB) with regard to various external hazards, especially APC of aircraft beyond the design basis of the NPP. These investigations provide confidence that significant structural reserves are available:

- The Swiss authority HSK has investigated the ability of the Swiss plants to withstand commercial APC. This investigation also includes plants Beznau (PWR) and Mühleberg (BWR), which are of the same construction period as KCB and also not designed for APC. HSK stated that these reactors could withstand the loads from a crash of a commercial APC (Boeing 707) with a velocity of 340 km/h. Further margins for more demanding load curves would be available.
- The German PWR Neckarwestheim 1, also of the same construction period as KCB, has a 60 cm-thick concrete shield, as has KCB. This shield has been shown to withstand military APC (Starfighter). The calculations demonstrate significant margins for more demanding load curves. Analyses also show that a crash of a commercial APC with a mass of 65 tons and a velocity of up to 360 km/h could be mitigated.
- A study conducted at the request on the Nuclear Energy Institute following the events from 11 September 2001 emonstrates that all US power plants and their fuel storage facilities are expected to mitigate the crash of a Boeing 767-400 (take-off weight 225 tons) at a speed of 350 miles per hour (ca. 563 km/h). However, APC is not a design-load case in the USA.

The structural capacity of KCB's reactor building regarding APC of small and medium-sized aircraft has been analysed . The study aimed at identifying the ultimate resistance of the shell against bending and normal forces on the one hand, and punching on the other. Punching of the shell by a motor has also been considered. The results indicate that the cylindrical part of the shell can certainly resist the impact of a small sized aeroplane. It was further shown that a short-range, medium-sized airliner weighing about 20 tons, with an impact speed of 360 km/h, are also not likely to induce significant cracks or even punching. The shell resistance of the upper spherical part of the reactor building has been found to be even higher by a factor of approximately 1.6. This fact induces significant additional margin since the lower cylindrical part of the building is well shielded by the Westerschelde dyke and on-site structures as explained above. However, the study considers it is not possible to demonstrate the integrity of the shell regarding the impact of long-range airliners.

In connection with the 9/11 events, four different realistic scenarios of a large aircraft APC have been analysed for KCB], concerning a large passenger aircraft with almost full kerosene tanks (175,000 kg):

1. coming from the west, the aircraft crashes into buildings 03, 04 and 05 of KCB, with one wing and its engines land in the containment (building 01);
2. the aircraft crashes into the main control room and the turbine building (buildings 04 and 05);
3. the aircraft hits the primary/secondary penetrations, main steam valves and conduits, the main steam relief station and ventilation stack (buildings 03, 04, 05 and 10);
4. an engine or engine parts penetrate the reactor building (buildings 01 and 02).

Buildings 01 and 02 are expected to be penetrated, causing loss of containment. However, this type of loss of containment does not automatically mean a core damage or a large release of radioactivity. The other buildings mentioned are expected to be destroyed or heavily damaged, either by the crash itself or by the kerosene fire. Core damage can be prevented if a single redundancy train of core cooling systems can be kept intact. This will be discussed in the next section.

7.6.2 Potential consequences for the plant safety systems

7.6.2.1 Safety systems and building arrangements

The safety functions for external events are mainly ensured by the Backup feed water system RS and Backup residual heat removal system TE, the bunkered Backup coolant makeup system TW, the Spent fuel cooling system TG 080, the Backup cooling water system VE and their supporting systems. The engineered safeguards RS and TW are actuated automatically by the Reactor protection system YZ. In case the automatic actuation is not functioning, sufficient time for manual actuation is available. For long-term residual heat removal, the reserve cooling chains can be actuated manually (TE, TG 080, VE).

The SSCs needed to support these functions are installed in the following buildings, which are protected against APC:

- reactor building with annulus (01/02);
- backup systems bunker (33);
- remote shutdown building (35);
- wells of the VE system: underground.

The KCB site arrangement provides the following sources of margin against military or commercial APC:

- partial protection due to upstream or adjacent buildings within the plant area limiting potential routes to the buildings 01, 02, 33 and 35. These buildings include in particular the coal-fired plant on the site, the radioactive waste-storage building 34, the switchgear building 05, the nuclear auxiliary building 03 and the turbine building 04. The reactor building itself serves as a shield for buildings 33 and 35. Other less massive buildings at least provide some protection;
- essential safety functions can be taken by different safeguards installed in spatially separated buildings. Examples include feedwater supply, which is possible from turbine building 04 and backup systems bunker 33, and emergency power supply distributions D1 and D2, which are installed in divisionally separated buildings. Similarly the main control room and the emergency control room are installed in the spatially separated buildings 05 and 35 and are not likely to fail simultaneously.

7.6.2.2 Consequences of an airplane crash

The consequences of an APC can be broken down into the direct destruction of structures and the subsequent kerosene fire. For the buildings with protective functions, the effects resulting from fuel fires and deflagration are mitigated by the structural design and spatial separation. Due to the thickness of the concrete structures, high temperatures resulting from fuel fires on the plant site are also controlled. Additionally, a modified military crash tender can be used to fight a kerosene fire with foam, not only effectively but also from a safe distance. A possible deflagration of fuel is covered by the design against explosion pressure wave.

Buildings 01, 02, 33 and 35 are shown in dynamic calculations to withstand the bending and shear stresses from direct impact of small and medium-sized airplanes, and to preclude an entrance of their fuel oil . The induced vibrations are covered by earthquake-induced vibrations. A specific qualification from SSCs against induced vibrations from APC is therefore not necessary. Also, the wells of the VE system are protected against APC by adequate soil coverage and spatial separation.

With regard to a large airplane crash, the buildings and systems directly hit and threatened by leaking kerosene have been identified for each of the considered scenarios The following consequences have been identified for the relevant safety systems for each of the four scenarios:

1. The Safety injection & residual heat removal system TJ, and the main and auxiliary feedwater system RL are threatened if burning kerosene enters the lower levels of buildings 04, 05, 02 or 03. At least one redundancy train of the bunkered Backup coolant makeup system TW and one of the Backup feed water system RS are not threatened because of the external hazard-resistant structure of building 33.
2. TJ will remain available for at least one hour, but can fail due to a lack of power supply. RL will fail immediately due to the destruction of building 04 with the (emergency) feedwater supply. TW and RS will remain available.
3. TJ will not be hit directly. It can fail, however, by fires in buildings 04 and 05 or in the lower levels of 02 and 03. If the feedwater lines are hit, RL will fail immediately. TW and RS will remain available.
4. TJ will not be hit directly, except one buffer tank might be. Failure of this system by a kerosene fire in building 01 is unlikely as the necessary valves, pumps, heat exchangers and water supplies are in other buildings or already in the right position for emergency operation. Leakage of the buffer tanks will not cause a primary leak as the relevant set of valves will be closed for both high and low-pressure operations. TW could fail if its ducts inside building 01 are hit. Engines or engine parts could drop into the spent fuel pool (SFP). This is comparable with the accidental dropping of a spent fuel container in the pool. The relatively small mass of the engine or engine parts, compared to the spent fuel container, leads to the conclusion that the SFP will not fail; at most it will develop a minor leak. Due to the large water volume in the SFP this effect is considered inferior to the prevention of a reactor core melt.

For each of these scenarios, priorities for accident mitigation have been identified. Also, the highly improbable worst cases have been analysed, including the occurrence of radioactive releases. To cover these cases, the National Plan for the Control of Nuclear Accidents (Nationaal Plan Kernongevallenbestrijding) has been drawn up and involves relief organisations like the fire brigade, community health service (GGD) and police, with whom preparatory consultations have taken place.

The final conclusion that can be drawn is that the plant is well equipped to deal with airplane crash according to the design basis. The same conclusion is likely to be valid for the impact of medium-sized airplanes (beyond design). For large airplanes (beyond design), the availability of at least one redundancy of the backup feed water system RS and the backup coolant makeup system TW is required to handle this event safely.

Recommendation: 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.

7.7 Toxic gases

7.7.1 General description of the event

An accident with toxic gases may pose a threat to the control room personnel but will not threaten the plant systems. These gases may originate from:

- chemicals present on the premises of CCB and KCB;
- chemicals on trains on the railroad passing the KCB site;
- chemicals on ships passing the KCB site on the Westerschelde estuary;
- chemicals present on the premises of companies on the nearby industrial estate.

The considered chemicals present at CCB and KCB are hydrochloric acid, hydrazine, a hydrazine/levoxin mixture and ammonia ($NH4OH$). The industrial chemicals transported on the nearby railroad are phosphor, phosphoric acid, sodium-tri-polyphosphate, butadiene, butyl-acrylate and methyl acrylate, Ammonia ($NH3$) and LPG have been identified as chemicals being transported in large quantities by ships on the Westerschelde

In the vicinity of KCB, seven companies dealing with a large variety of potentially toxic chemicals were identified , . The toxic hazards from these companies within a 10 km radius of the plant site have been evaluated based on the Risicokaart (risk map) of the Province of Zeeland[61]. The map shows that the 1E-6/y risk contours of all industrial activities are well away (at least 900 m) from the KCB site. Based on this information, it is reaffirmed that the toxic risk from industrial activities is negligible.

7.7.2 Potential consequences for the plant safety systems

The relevant locations for toxic gases at the KCB nuclear plant are the main control room, the Backup systems bunker (33) and the emergency control room (35).

Various release scenarios (mostly maximum releases) have been modelled in a TNO study for toxic chemicals on site. It is concluded that, at the most relevant location (the air inlet of the KCB main control room), the alert threshold value (AGW) is not exceeded in any of the cases.

It was also concluded that collisions involving a ship carrying ammonia could lead to significant ammonia concentrations at the KCB site and in the inlet ventilation air of the KCB control room It is for this reason that the probabilities of railway and ship accidents with toxic release were analysed taking into account amounts of chemicals, passing frequencies of the trains and ships, and implemented control room habitability measures. Small, although finite, probability figures have been derived for these events. In these situations, control room ventilation will be switched to recirculation. Following the TNO study, additional sensors have been installed in the ventilation inlet of the KCB control room. Also, self-contained breathing apparatus with compressed air is available for all control room personnel and other operators, with a capacity of at least one hour. After this period, it can be reasonably assumed that the toxic

[61] http://nederland.risicokaart.nl/risicokaart.html?prv=zeeland

cloud will have dispersed in the air. Refilling equipment is available for the breathing apparatus.

In case the measure above would not work sufficiently and control room personnel were incapacitated by the toxic gases still entering the control room, the plant will continue to run, as long as no malfunctions or resource shortages occur. If this were to happen, no human intervention is required during the autarky period of ten hours, in which the transition to a hot shutdown state will be made . After this, human intervention is required to bring the plant to the cold shutdown state, if necessary from the emergency control room (35).

Ventilation of the Backup systems bunker (33) is essential to keep system temperatures within prescribed limits. Although entering this building is only necessary after the ten-hour autarky period, no protection against toxic gases is available at the moment. The same situation applies to the emergency control room in building 35.

In conclusion, the plant is well equipped to handle this event safely.

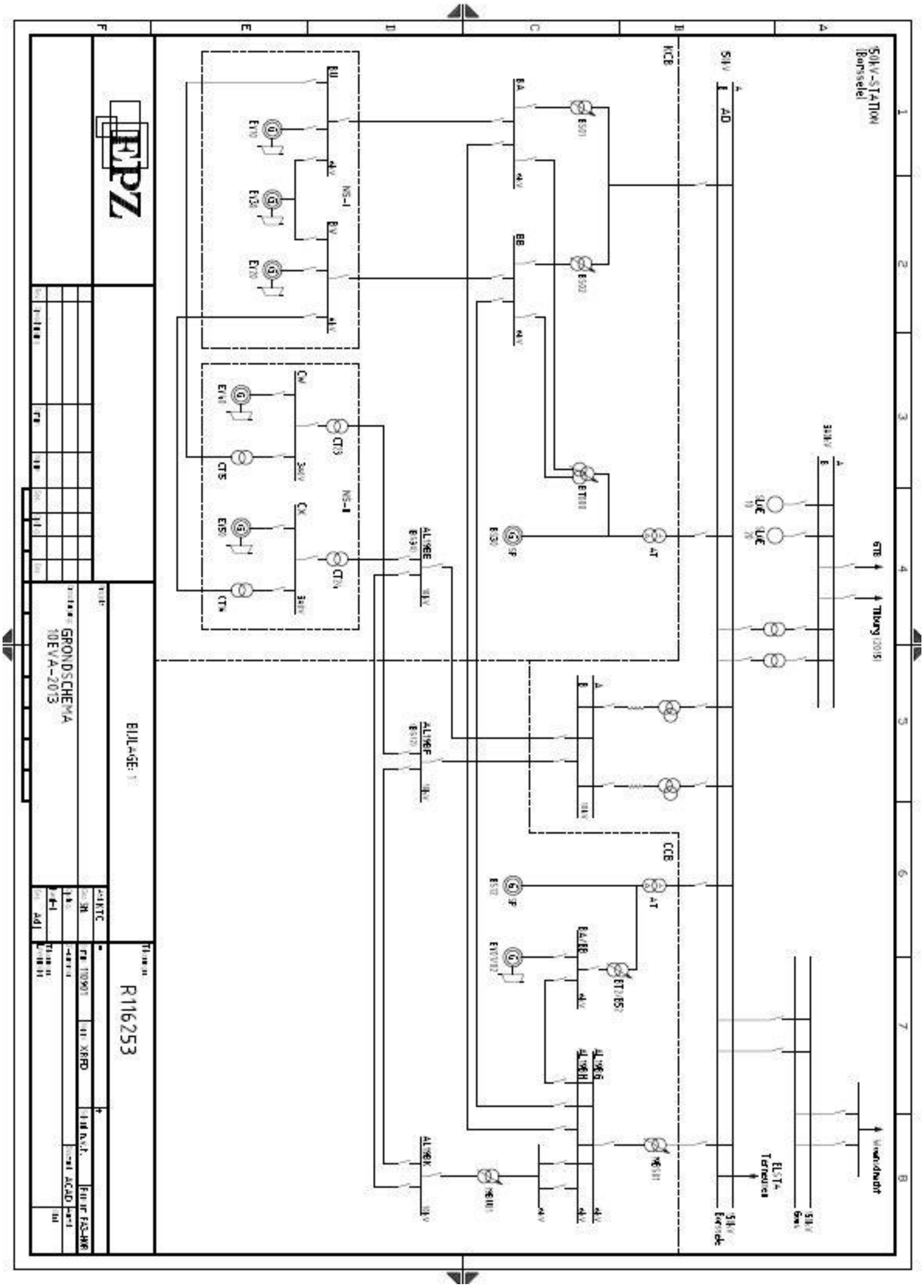


Figure 7.1 Schematic overview of the 150/380 kV grid in the south-western part of The Netherlands, with its coupling to the house grid of the nuclear power plant KCB and the coal-fired power plant CCB

7.8 Large grid disturbance

7.8.1 General description of the event

The EPZ nuclear plant KCB is connected to the external grid by a generator/step-up transformer (21/150 kV) / house-load transformer (150/6 kV), and by two start-up transformers (150/6 kV). Indirectly, the 10 kV supply system of NS 2 is connected to the grid, as can be seen in Figure 7.1. The first two are used during normal power operation; and the latter, NS 2, during shutdown and start-up of the plant. Large grid disturbances, including overvoltage, voltage interruption/short circuit and frequency transients, could cause damage to the house-load transformers and the turbine train (turbine plus generator); in the worst-case scenario causing their permanent unavailability or causing damage to safety systems such as the emergency feedwater system. Therefore the plant is protected against large grid disturbances by unit, bus bar and feeder protection systems that can isolate the generator, transformers and bus bars from the grid.

In the context of this study the spectrum of grid disturbances includes:

- short circuit (near field) with a voltage interruption within or exceeding the critical short circuit time of the unit;
- overvoltage transients resulting in internal overvoltages;
- transient instability of the national /regional grid;
- over- or underfrequency;
- undervoltage.

The design basis of the operational limits (voltage/frequency) was defined by the requirements of the Electric Utilities Cooperation SEP, which were succeeded by the grid and system codes of the Dutch energy regulator DTe (2004). These codes provide limits for operating at frequencies and/or voltages divergent from the nominal levels.

7.8.2 Failure mode effects of the grid disturbances

7.8.2.1 Short circuits

A short circuit (SC) in the grid will result in a voltage dip with a loss of power load resulting in acceleration of the turbine train. Classically, short circuits occur due to flashovers in overhead lines or switching failures. High currents may occur, connected with resistive heat generation and elevated temperatures.

The unit will retain stability if the critical short circuit time is not exceeded. Eventually the LOOP scenario can be initiated when the turbine has tripped and the transfer to the auxiliary connection fails. The most extreme scenario is a short circuit combined with a simultaneous failure of turbine and unit protection systems: in this case the shaft acceleration and torque transients could lead to overspeed and possible damage to the turbine train. Emerging projectiles like turbine blades could then cause damage to the feedwater tank or the emergency feedwater system. Due to the spatial orientation of the turbine, the potential damage remains limited to this, and the control room or the reactor building cannot be hit.

7.8.2.2 Overvoltage transients

The unit is protected against overvoltage (turbine trip) during normal operation; however, the start-up transformers and the grid supply of the emergency grid NS 2 do not have an overvoltage protection. Overvoltage could result in tripping the operating safety components of the emergency grid NS 1 if this protection fails. Overvoltage surges will normally result in actuation of the component protection systems (overcurrent). If these protection devices fail, the surge could affect NS 1, which provides the uninterrupted power supply (UPS). An overvoltage surge in the 10 kV connection of the second emergency power system NS 2 will be damped by the cables, but in severe cases of overvoltage the NS 2 could be affected as well.

7.8.2.3 Transient instability

Transient instability will result in large excursions of grid voltage and frequency. These excursions will initiate load rejection of the turbine, or even a turbine trip, which will usually lead to a stable situation. In the worst-case scenario, when exceeding the frequency thresholds, NS 1 and NS 2 will be isolated by the reactor protection system.

7.8.2.4 Over-/Underfrequency

An unbalanced load and production will result in a deviation of frequency. Deviating frequencies will result in a control reaction of the turbine load controller. In the worst cases, the load controller is overruled by a frequency boundary controller with priority control. This controller prevents a further escalation of frequency during isolated regional operation (no main connection to the national grid). Underfrequency will actuate a disconnection of the unit at 48 Hz and initiate an island operation of the unit.

7.8.2.5 Undervoltage

The unit is designed to operate at 80% of the nominal voltage. This threshold will also initiate the emergency power supply system. Operation near this limit during a design-basis event could lead to the unavailability of safety components located at large distances from the main feeding points. An early warning for this is provided by alarms in the control room.

7.8.3 Protection limits

The protection consists of a turbine protection and unit protection. The turbine protection provides a redundant component protection for overspeed. The unit protection provides electrical protection for overload, grounding failure, short circuit and overfluxing .

7.8.4 Potential consequences for the safety systems

Large grid disturbances could result in equipment being damaged (transformers, turbine shaft), resulting in long downtime of the plant, but without impacting safety. Besides this, the reliability of safety systems may be affected in the following ways:

- collapse of the power system itself (e.g. by triggering protective system actions) resulting in overloading transmission lines/transformers and requiring time to restore offsite power;
- unavailability of safety-related electrical components, with the most serious consequence being the loss of the UPS, supplied by the emergency grid NS 1;
- in the case of a severe grid instability causing the turbine to break apart and lose parts, there would be physical damage to the auxiliary feedwater system.

The most serious consequence of the overloading of transmission lines and transformers will be the disconnection from the off-site power grid for a longer time. This time period is determined by the replacement or repair of the damaged transformers and effective communication for restoring power to the nuclear power plant KCB. Until then, the plant will be in a LOOP situation.

The most serious consequence of the loss of safety-related electrical components would be the unavailability of the emergency grid NS 1, resulting in a situation of station blackout (SBO), but with the emergency grid NS 2 still available. Chapter 5 provides a description on how the plant can be shut down safely, departing from both a LOOP situation and a SBO situation.

In conclusion, the plant is well equipped to handle this event safely.

7.9 Failure of systems by introducing computer malware

7.9.1 General description of the event

In this report, a ’failure of systems by introducing computer malware’ means the accidental contamination of KCB by computer malware. This could be caused either by users on the EPZ premises or by an internet connection.

The plant systems are mostly controlled by non-computerised electronic hardware. A few non-safety-related auxiliary systems at the cooling water inlet and the main generator are equipped with Programmable Logic Controllers (PLCs), which do have connections that are vulnerable for accidental malware contamination.

The control room is equipped with a process presentation system that can be accessed from outside by plant personnel, the Dutch nuclear regulator and the Siemens plant crisis support staff in Erlangen. If there was an accidental malware contamination, false information could hypothetically enter through this access, and therefore cause confusion to the control room personnel.

7.9.2 Potential consequences for the plant safety systems

The plant systems are in general controlled by non-computerised electronic hardware. Also the reactor protection system is not processor or PLC controlled. The process computer is only supportive for plant control: it only shows information and does not control plant systems. The bunkered systems (emergency diesel generators of NS 2, RS, TW) don’t contain PLC controlled safety systems. The emergency diesel generators only have PLCs in non-safety relevant auxiliary systems. Failures will never prevent manual starting. All this offers sufficient protection for the reactor and its safety systems.

False presentations by the Process Presentation System (PPS) in the control room could cause false actions by control room personnel. The (non-programmable) reactor protection system will always interfere, both in normal and accident conditions. Control room personnel are not allowed to use the PPS during accident conditions, and have been trained to execute emergency operation procedures without the use of the PPS. Under these conditions, decisions must always be based on data from diverse sources (e.g. both steam generator water level and feedwater flow). During accident management, it is assumed that some of the instrumentation could become unreliable, and therefore verification against other process information is required. Overruling of the reactor protection system is impossible from the control room.

Components equipped with potentially vulnerable PLCs are not safety-related . As a result, malfunctioning of these components by false programming of their PLCs will not jeopardize the plant safety functions.

In conclusion, the plant is well equipped to handle this event safely.

7.10 Internal flooding

7.10.1 General description of the event

7.10.1.1 Design basis

By definition, internal floods are floods that originate from systems that are part of the primary, secondary and/or auxiliary systems that constitute the nuclear plant.

The prerequisite for mitigating internal floods is to prevent the loss of redundancies of safety-relevant systems and components through flooding The means that are implemented are:

- guidance of leak flows and collection tanks;
- local leakage storage tanks;
- water shielding;
- exclusion of leakage;
- physical separation of redundancies.

In case these means fail, the RS mitigation concept will deal with the consequences of flooding for all flooding events and non-isolatable secondary leakages inside the containment. With this concept, the following generic safety functions are assured:

- shutdown of the reactor and maintaining sub-criticality;
- automatic residual heat removal via the steam generators;
- decay heat removal from the spent fuel pool and shutdown by the ultimate residual heat removal chain TE/TG080/VE.

Assurance is gained by systems that are placed inside the Reactor building (01), the Annulus (02), the Nuclear auxiliary building (03), the Backup systems bunker (33) and the atmospheric steam dump. Flooding these systems and, as a consequence, their failure is prevented by design.

For each system that contains water, the amounts that can be drained into the building or specified rooms have been determined. Maximum water levels have been assessed and systems that will be flooded identified. The following is concluded

A Reactor building (01)

Several valves of the Safety injection & residual heat removal system TJ will be flooded, as well as safety-relevant signal transducers (pressuriser level, pressure and temperature). Measures to protect these components from water penetration are implemented to ensure the operation of these components; all safety-related components within building 01 are designed to cope with conditions during LOCAs (steam and water flooding).

B Annulus and the Nuclear auxiliary building (02, 03)

It is concluded that the elevation of the safety-relevant systems is above the maximum water level that can occur from leakage of the systems present in this area. It is noted that the pumps of the backup residual heat removal system TE can even operate when submerged.

C Backup systems bunker (33)

The backup feed water system RS is a complete, spatially separated, redundant system installed on the upper floors of the Backup systems bunker (33). Internal floods inside this building will affect only one of the redundancies. The event will be mitigated because the other redundancy will remain in operation. The RS function can only be lost if two flooding events happen at the same time in both redundancies.

D Remote shut-down building (35)

No flooding can occur as there is no internal water source.

E Athmospheric steam dump

The elevation of the safety-relevant components of this system (on the roof of building 03) is such that flooding by water drained from the adjacent system will not reach these components. The doors, which are at lower levels, can be opened in due time.

7.10.1.2 Beyond design conditions

For this event, the beyond design conditions are defined as those conditions where an excess of water can be drained to the rooms or buildings under consideration by external means, e.g. supply by the low pressure fire extinguishing system UJ or conventional emergency cooling water system VF.

The following buildings are affected by an internal flooding event:

- annulus (02);
- nuclear auxiliary building (03);
- turbine building (04);
- electrical building (05);
- diesel generator building (10);
- cooling water inlet building (21);
- backup systems bunker (TW/RS) (33).

7.10.2 Potential consequences for the plant safety systems

In a detailed analysis, the buildings relevant to this event were identified, together with the safety systems they are housing. The extent that the event can impact the building itself and redundancies that are placed in the building were considered first. Next, the loss of combinations of buildings was examined, especially those combinations which would lead to the unavailability of at least one of the three cooling functions: cooling down, decay heat removal and spent fuel pool cooling. In general, it can be stated that failure of those building combinations is hardly possible due to their spatial distribution. The simultaneous destruction of such combinations could be regarded as cliff edges.

From the point of view of internal flooding, the safety-relevant equipment in building 04 can never be damaged by internal flooding as it is all located at a minimum level of 6.6 m + NAP and the building is completely open at ground level (3.2 m + NAP). As the same is true for buildings 05 and 10, and the adjoining building 03 is watertight up to at least 7.7 m + NAP, internal flooding from building 04 will have no impact on other relevant buildings. Moreover, the plant walkdown for the probabilistic internal flooding showed that steam as well as water leakage/flooding will only affect a maximum of one redundancy of the (safety) systems located in building 04. This means that all scenarios that result in a loss of cooling with building 04 in them cannot be caused by flooding.

The same rationale is valid for building 10. In building 05, the amounts of water present that could affect systems are in the fire hoses located in the open stairwells. The water will flow outside without major impact. Scenarios with building 05 or 10 in them can therefore be discarded.

As result of the analyses, no scenarios exist in which internal fllooding will make cooling down impossible.

Other scenarios for internal flooding for decay heat removal and spent fuel cooling are identical: namely loss of building 02 and loss of buildings 21 and 33.

The last scenario is very improbable, because buildings 21 and 33 are wide apart (in fact there is a dyke with a crest height of 10 m + NAP between them). Furthermore one has to keep in mind that building 33 consists of two separate buildings, one for each redundancy. This means that three independent internal flooding events, which have to exceed design conditions, are necessary to result in this situation.

Internal flooding, exceeding design conditions, which causes problems for the safety (related) systems of building 02 would be caused by a specific event in building 03 a rupture /leakage of the VF piping. Buildings 02 and 03 are, from a flooding point of view, one building, so the water will spread to building 02 where the TJ pumps are housed. The VF piping in building 03 is in an area enclosed by a small wall of limited height (approximately 50 cm). Within this area a level alarm is present. Furthermore, VF leakage will trip the VF pumps and stop the flow. TJ will only flood when the pumps don't trip automatically and the level alarm fails. Given the design of building 03 (watertight to at least 8 m + NAP), the TE pumps can become submerged, but they are designed to operate under flooding conditions. The TG pumps are located on the 13.2 m +

NAP floor and will continue to function. This means that decay heat removal and spent fuel cooling will also continue to function.

It has been shown that the design with the included features, like spatial separation of redundancies of safety-related systems and the elevation of safety-related equipment, copes very well with these events for design conditions. As regards the beyond design conditions, it is concluded that the extent and range of these events cannot be so large that the plant will lose its cooling functions for the reactor and spent fuel pool completely. Therefore the final conclusion that can be drawn is that the plant is well equipped to deal with this event, both in design basis and beyond design conditions.

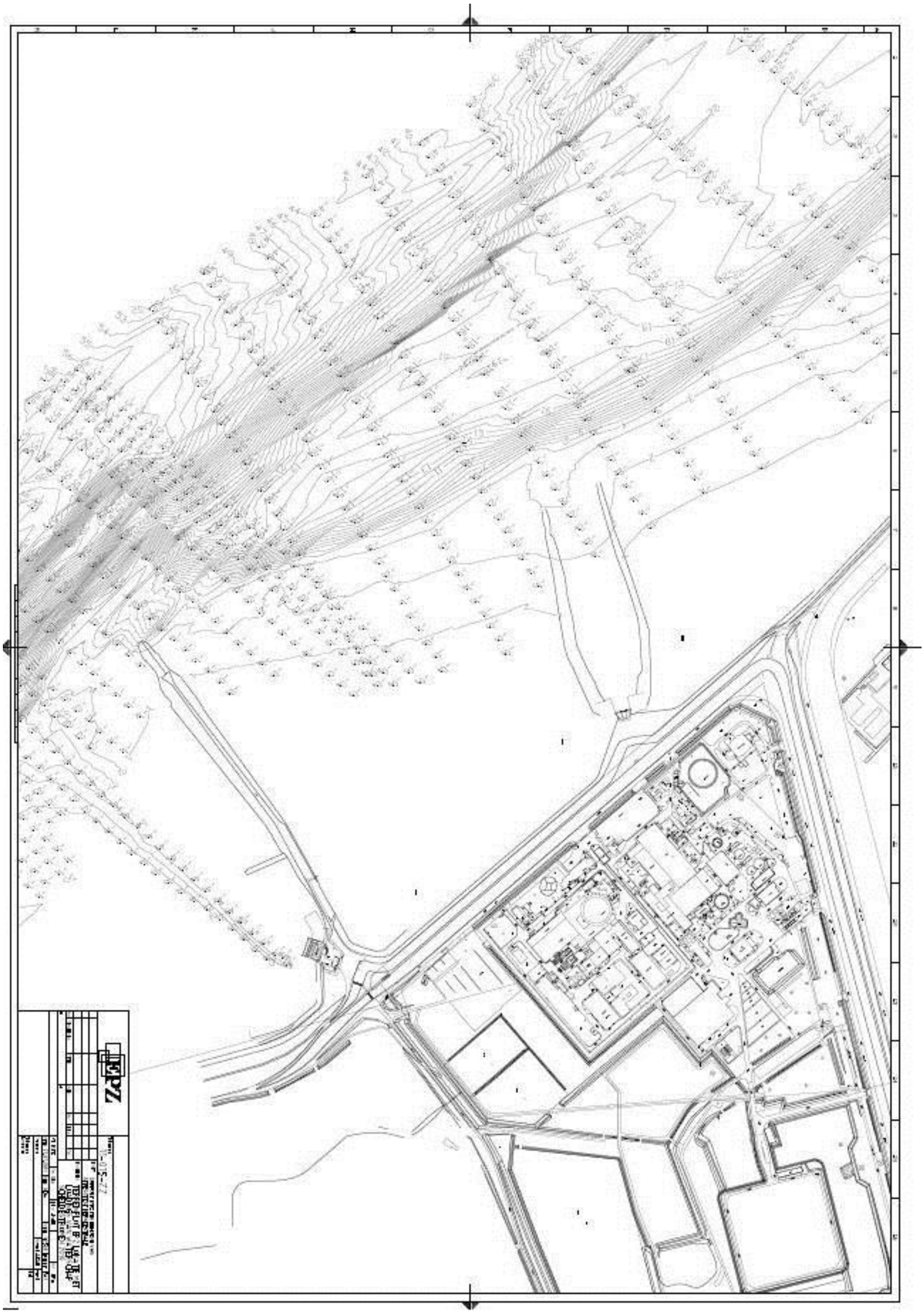


Figure 7.2 Map of the KCB Plant showing the depths of the Westerschelde

7.11 Blockage of cooling water inlet

7.11.1 General description of the event

A blockage of the cooling water inlet could be caused by two initiating events:

1. Ship grounding;
2. Biological phenomena.

In the sections below, these phenomena are described, together with their impact on the plant's safety systems. Blocking by ice during cold weather is addressed in Chapter 4.

7.11.1.1 Ship grounding

In this event, a ship takes a wrong direction and runs aground at the location of the nuclear power plant. This can be at the location of the channel of the cooling water inlet, or at a second location on the beach in front of the power plant.

The dyke between the power plant and the Westerschelde has been designed as a protection against flooding, but may also serve as a protection against grounding ships. Between the dyke and the shipping lane, the Westerschelde is at first very shallow and inappropriate for large ships. After a few hundred meters, the water depth increases quickly to a depth suitable for large ships. The beach and the shallow part of the Westerschelde together are about 1 km wide. Therefore it can safely be assumed that a grounding ship will remain on the beach and not damage the dyke or the power plant buildings behind it. A map of the plant showing the water depths of the relevant part of the Westerschelde can be seen in Figure 7.2.

A ship that enters the cooling water inlet, which is dredged periodically, may (partially) block the cooling water inlet or damage the Cooling water inlet building (21). A partial blockage can lead to a shutdown; however the Conventional emergency cooling water system VF will still be available as this system only requires a minimum of 2% of the regular water intake. If there is a complete blockage or damage to building 21, VF is lost and a loss of UHS scenario develops.

7.11.1.2 Biological phenomena

The only biological phenomenon of relevance is the clogging of the cooling water inlet by large amounts of jellyfish. However, the jellyfish problem is not really an extreme hazard like the other events discussed in this report as it occurs regularly.

In the Cooling water filtering system VA, located in the cooling water inlet building (21), the water from the Westerschelde estuary flows through a seal grid, a coarse filter and a fine filter in succession. The fine filters have an automatic cleaning system; however they could become clogged when a large number of jellyfish are present It should be mentioned that mussels and algae also grow here but this is a slow process.

7.11.2 Potential consequences for the safety systems

7.11.2.1 Ship grounding

It can be shown that the event of a ship grounding outside the cooling water inlet is sufficiently covered by the measures against explosion pressure waves and toxic chemicals.

When a ship grounds on the beach, which separates the nuclear power plant site from the Westerschelde, it will not cause any damage to the plant.

If a ship grounds in the cooling water inlet channel, the cooling water inlet building and its systems may be damaged. In that case the reactor will be scrammed, and the plant will be cooled down with the Emergency feedwater system RL and if necessary the Backup feed water system RS. Provisions are also available to remove the residual heat with the fire brigade's equipment. If such a situation were to occur for a long period of time (more than 13 hours), the residual heat can be removed via the Backup cooling water system VE (connected to eight deep water wells) . In conclusion, the plant is well equipped to handle this event safely.

7.11.2.2 Biological phenomena

A too great pressure drop over the fine filters of the Cooling water filtering system VA will be indicated in the control room, and cause the Main cooling water system VC to be shut down in such a way (step-wise) that the Conventional emergency cooling water system VF remains functional . An instruction is available to handle clogged filters with enhanced cleaning and a jellyfish net .

A partial blockage can lead to a shutdown; however, the Conventional emergency cooling water system VF will still be available, as this system only requires a minimum of 2% of the regular water intake.

In case all these measures are failing and the jellyfish clog all the systems in the cooling water inlet building (both VC and VF), a loss of UHS situation develops: the reactor will be scrammed, and the plant will be cooled down with the Main and auxiliary feed water system RL and if necessary the Backup feed water system RS. Provisions are also available to remove the residual heat with mobile equipment. However, a long-term loss of UHS scenario can be excluded as it is always possible to remove the jellyfish with limited additional resources within a sufficiently short period of time. In conclusion, the plant is well equipped to handle this event safely.References

List of Acronyms & Abbreviations

AC	Alternating Current
ACC	Alarm Coördinatie Centrum (Alarm Coordination Centre)
A_{DBE}	Amplitude (design basis earthquake)
AM	Accident Management
APC	Air Plane Crash
ASME	American Society of Mechanical Engineers
A_{SSE}	Amplitude (safe shut-down earthquake)
ATEX	ATmosphères EXplosives
ATWS	Anticipated Transient Without Scram
BBN	Bayesian Belief Networks
b&f	bleed and feed
BOC	Bedrijfs Ondersteunings Coördinator (Operations Support Coordinator)
BWR	Boiling Water Reactor
CA	Computational Aids
CALWEB	CALamiteiten WEB (Calamity WEB)
CCB	Conventionele Centrale Borssele (Borssele Coal-fired Power Plant)
CCDF	Complementary Cumulative Distribution Function
CDF	Core Damage Frequency
CFD	Computational Fluid Dynamics
CSA	Complementary Safety margin Assessment
CVCS	Chemical and Volume Control System
DBE	Design Basis Earthquake
DBF	Design Basis Flood
DC	Direct Current
DFC	Diagnostic Flow Chart

DG	Diesel Generator
DHR	Decay Heat Removal
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EDMG	Extensive Damage Mitigation Guidelines
EMS	European Macroseismic Scale
ENSREG	European Nuclear Safety Regulator Group
EOP	Emergency Operating Procedure
EPR	European Pressurised Reactor
EPRI	Electric Power Research Institute
EPW	Explosion Pressure Wave
EPZ	N.V. Elektriciteits-Produktiemaatschappij Zuid-Nederland EPZ
ERD	Earthquake Resistant Design
ERO	Emergency Response Organisation
EU	Europian Union
10EVA13	Current 10 yearly safety evaluation
EY	Diesel Generators
IAEA	International Atomic Energy Agency
IPCC	Intergovernmental Panel on Climate Change
FHP	Functie Herstel Procedure (Function Restoration Procedure)
FRG	Functional Restoration Guidelines
FSAR	Final Safety Analysis Report
HCLPF	High Confidence Low Probability of Failure
HKCB	Head of Nuclear Power Station Borssele
HP	High Pressure
HPME	High Pressure Melt Ejection
HPSI	High Pressure Safety Injection

HVAC	Heating, Ventilation and Air Conditioning
HSK	Hauptabteilung für die Sicherheit der Kernanlagen (Switzerland)
I & C	Instrumentation and Control
ICAWEB	Integrale Crisis Advies Website
ISLOCA	Interfacing Systems LOCA
KCB	Kerncentrale Borssele (NPP Borssele)
KFD	Kernfysische Dienst (Nuclear Safety Department)
KNMI	Koninklijk Nederlands Meteorologisch Instituut
KTA	Kerntechnische Ausschuss
KWU	Kraftwerk Union
LCMS	Landelijk Crisis Management Systeem
LHSI	Low Head Safety Injection
LNG	Liquefied Natural Gas
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
MAAP	Modular Accident Analysis Program
MB	Manager Bedrijfsvoering (Manager Operations)
MCCI	Molten Core-Concrete Interaction
MCR	Main Control Room
MCT	MCT Brattberg, Industrial fire proof and pressure sealed cable transits
MMI	Modified Mercally Intensity
MOD	Manager Ondersteunende Diensten (Manager Support Services)

MOV	Motor-Operated Valve
MOX	Mixed Oxides
MSB	Manager Stralingsbescherming (Manager Radiation Protection)
MSIV	Main Steam Isolation Valve
MSK	Medvedev, Sponheuer en Karnik
MSRT	Main Steam Relief Trains
mSv	milliSievert
MW_e	Megawatts Electrical
MW_{th}	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
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NS 1	Nood Stroom net 1 (Emergency Grid 1)
NS 2	Nood Stroom net 2 (Emergency Grid 2)
NUREG	Nuclear Regulatory – A Series of United States Nuclear Regulatory Commission Reports
PAR	Passive Autocatalytic Recombiner
PGA	Peak Ground Acceleration
PLC	Programmable Logic Controller
PORV	Power-Operated Relief Valve
POS	Plant Operational State
ppm	parts per million

PRA	Probabilistic Risk Analysis
PRZ	Pressurizer
PS	Protection System
PSA	Probabilistic safety Analysis
PSM	Plant Security Manager
PWR	Pressurised Water Reactor
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RELAP	**R**eactor **E**xcursion and **L**eak **A**nalysis **P**rogram
RESA	Reaktor Schnell Abschaltung (Reactor Scram)
RHR	Residual Heat Removal
ROT	Regional Operational Team
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRS	Required Response Spectra
RWS	Rijkswaterstaat
SACRG	Severe Accident Control Room Guidelines
SAEG	Severe Accident Exit Guidelines
SAG	Severe Accident Guidelines
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout
SCG	Severe Challenge Guideline
SCRAM	Security Control Rod Axe Man
SCST	Severe Challenge Status Tree
SED	Site Emergency Director
SITRAP	SITuation REPort (in Dutch SITRAP)

SFP	Spent Fuel Pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SMA	Seismic Margin Assessment
SOB	Splijtstof Opslag Bassin (Spent Fuel Pool)
SOER	Significant Operating Experience Report
SQUG	Seismic Qualification Utility Group
SSCs	Structures, Systems and Components
SSIE	Special System Initiators
STC	Source Term Category
TAG	Technische Analyse Groep (Technical Support Group)
TCDF	Total Core Damage Frequency
TIP	Technisch Informatie Pakket (Technical Information Package)
TNO	Nederlandse Organisatie voor toegepast-natuurwetenschappelijk onderzoek
TRANS	Transients
TRS	Test Response Spectra
TS	Technische Specificaties (Technical Specifications)
TUSA	Turbine Schnell Abschaltung (Turbine Stop)
UK	United Kingdom
UPS	Uninterrupted Power Supply
US	United States
USAEC	United States Atomic Energy Commission
VDC	Volt Direct Current
VROM	Volkshuisvesting Ruimtelijke Ordening en Milieu
WANO	World Association of Nuclear Operators
WINREM	WINdows application for REM (radiation emergency management) calculations
WOG	Westinghouse Owner's Group

EPZ

constantly improving

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