Complementary Safety Margin Assessment “Onderzoekslocatie Petten"

Under the contract of NRG

rev. no.	date	description
C	29/2/12	Final
B		After review ERC
A		1st draft

name:	Final Report Complementary Safety Margin Assessment 29 feb 2012 definitief (ELI).docx :
referencenr.:	25192/12.113089
468 pages	29/02/2012

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Prologue

On March 11, 2011, a large part of the Japanese eastern coastal area was struck by an earthquake of enormous magnitude, followed by a devastating tsunami. This natural disaster killed thousands of people and caused extensive damage to Japanese cities and infrastructure. After the earthquake, the nuclear reactors from the Japanese Nuclear Power Plant station Fukushima Dai-ichi shut down automatically. However the station failed to adequately maintain all of its safety functions after been hit by the tsunami initiated by the earthquake. As a result the local environment suffered from radioactive releases, requiring large zones to be evacuated, and generating serious concerns internationally about nuclear safety.

In the wake of the disaster the European Union decided to assess safety of all operating nuclear power reactors in its member states. This safety assessment 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 and addresses 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.

On June 1, 2011, The Dutch Regulatory Body asked NRG to perform a safety assessment for the High Flux Reactor (HFR) and the other associated nuclear facilities located on the “Onderzoekslocatie Petten” (OLP).

Consequences of these extreme hazards for the HFR and the other associated nuclear facilities located on the “Onderzoekslocatie Petten” have been evaluated using available safety analyses, supplemented by engineering judgement. In this way, the robustness of the existing facilities was assessed and possible measures to further increase the safety margins were identified. The present report documents the results of the Complementary Safety margin Assessment (CSA) performed for all the nuclear facilities located on the “Onderzoekslocatie Petten”.

The distinct difference between this report and prior risk analysis reports in general and the existing Safety Reports of the nuclear facilities 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 nuclear facilities on the OLP. The NRG project team has been supported by several external experts. Apart from NRG internal reviews, review has also been performed by a dedicated External Review committee consisting of independent outside experts.

The main purpose of this technical report is to address the questions posed on NRG by the Ministry of Economic Affairs, Agriculture and Innovation and the outcome of the assessment.

Executive Summary

Complementary Safety Margin Assessment (CSA)

As a consequence of the accident at the Fukushima Dai-ichi Nuclear Power Plant (NPP) in Japan, the European Council meeting on March 24^{th} and 25^{th} 2011 declared that 'the safety of all EU nuclear power plants should be reviewed, on the basis of a comprehensive and transparent risk assessment ('stress tests')". Based on this decision, the Dutch Ministry of Economic Affairs, Agriculture and Innovation (EL&I) requested NRG, the licensee of the nuclear facilities on the Petten site, on June 1^{st} 2011 to perform a stress test of their nuclear facilities based on the ENSREG specifications. In addition, the Ministry indicated that in the assessment 'deliberate disturbances' should be taken into account as well. This request was implemented by NRG as the Complementary Safety margin Assessment (CSA) of the "Onderzoekslocatie Petten (OLP)" of which the results are documented in the present report.

Methodology

The nuclear installations included in the CSA are the High Flux Reactor (HFR), the Low Flux Reactor (LFR), the Hot Cell Laboratories (HCL) (including the Molybdenum Production Facility), the Decontamination & Waste Treatment Facility (DWT), STEK hall, Jaap Goedkoop Laboratory (JGL) and Waste Storage Facility (WSF).

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

The assessment approach is essentially deterministic: when analysing an extreme scenario, sequential loss of protective systems, functions and potential other measures is assumed.

The assessment considers three elements:

- Provisions incorporated in the design basis and the facility'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 always 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 nuclear facilities on the OLP react, even if the emergency measures provided for that situation would fail. This means that for the determination of the safety margins, a deterministic approach is applied. The intention is that an increasing threat (for example, an increasingly higher tidal wave or more serious earthquake) is assumed, so as to determine how the nuclear facilities on the OLP and their safety management system respond and to what level of threat the safety systems work may be expected to work adequately. As a result, the assessment delivers the following insights:

- The performance of the nuclear installations located on the OLP and safety management systems under escalating accident conditions where protective measures and systems are considered to sequentially fail;
- The robustness of installations and their safety management systems;
- The potential for modifications to improve the robustness of the installations and safety management systems.

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 subsequent failures. The focus is on the following issues:

- Earthquake;
- Flooding;
- Extreme weather conditions;
- Other extreme events caused by various means such as external or internal events;
- Loss of electrical power supply (LOOP), including station blackout;
- Loss of the ultimate heat sink (LUHS);
- Combination of both (I.e. LOOP and LUHS).

The following combinations of hazards for which a causal relation exists, have been evaluated:

- Earthquake and consequent flooding;
- High air temperature + high water temperature;
- Low air temperature + low water temperature + snow;
- Snow + extreme wind;

- Extreme wind + extreme rainfall + lightning.

CSA in relation to continuous improvement at the nuclear facilities on the OLP

Within NRG, safety, including nuclear safety, has overriding priority both in organisation and operations. This is implemented through a pro-active approach to prevent and control possible hazards with respect to safety of individuals and the environment and a pursuit of excellence through continuous improvement. Using self-assessments and periodic safety reviews by external organisations, recommendations are obtained that lead to further improvement.

Every ten years, an extensive safety evaluation (10-EVA) is performed concerning nuclear safety and radiation protection. The objective of such a ten yearly safety evaluation is compare the existing situation to the state-of-the-art requirements and best practices using a comprehensive assessment as to whether the design basis and the safety documentation are still valid, arrangements in place to ensure the plant's safety are still correct and effective, and the research reactor conforms to current national and international safety standards and practices.

In addition to this periodic safety review, the HFR is subjected to a periodic integrated safety assessment of research reactors (INSARR) carried out by expert teams compiled by the International Atomic Energy Agency (IAEA). Following the INSARR 2005 and Safety Review Missions concerning the bottom plug liner repair in 2010, an INSARR was carried out in 2011. The recommendations of that mission have been integrated in the second extensive safety evaluation (10-EVA2012) currently in progress. The 10-EVA 2012 is due by December 31st, 2012.

The underlying CSA expands these safety assessments and reviews the robustness of the existing facilities. As an outcome, possible measures to further increase the safety margins have been identified that will lead to the implementation of improvements.

Main conclusion of the assessment

The main conclusions on the robustness of the HFR and associated nuclear facilities on the "onderzoekslocatie Petten" (OLP) against the assessed extreme hazards are given below. For each extreme hazard the validity of the design basis, the conformity with the design basis and the margins are briefly discussed.

For the HFR and associated nuclear facilities, no deviations with respect to the licensing base have been established.

Earthquake

The earthquake response of the nuclear facilities on the Petten research location (OLP) was assessed and evaluated using the Design Basis Earthquake (DBE) defined in 1998. There is no evidence of any tectonically induced (natural) earthquakes in the north of The Netherlands in the past. The DBE for the OLP site is a combination of a natural earthquake and a gas exploration induced earthquake with a horizontal peak ground (PGA) acceleration of 4 m/s^2. Based on all the current available data, the theoretical considerations for a gas exploration induced earthquake at the OLP, and the distance of the OLP to the nearest gas exploration location it is likely that the maximum PGA will be much smaller than the assumed DBE PGA of 4 m/s^2. Hence, the currently used DBE with a PGA of 4 m/s^2 is certainly adequate and holds in itself already a considerable safety margin.

Based on the defence in depth approach the consequences of the DBE, in relation to the three fundamental safety functions (reactor shutdown and sub-criticality, decay heat removal, and confinement of radioactivity), were assessed and evaluated for the High Flux Reactor (HFR). The evaluation revealed that the HFR can withstand a Design Basis Earthquake (DBE). The minimum calculated safety factor of the safety related structures, systems and components (SSC) of the HFR to avoid loss of heat removal capacity compared to the DBE is 1.1. The safety factor of the steel dome to avoid loss of containment with respect to a DBE is 3.2.

For the other nuclear facilities on the OLP site the prevention of release of radioactivity is ensured by different barriers. Based on design, analysis and engineering judgement, the barriers against radioactive release are qualified as seismic category 3. This qualification ensures that the fundamental safety functions for the other nuclear facilities on the OLP site are satisfied. The minimum safety margin for the other nuclear facilities compared to the DBE is 1.1.

Flooding

The impact and consequences of external flooding on the HFR and associated nuclear facilities at the Onderzoekslocatie Petten (OLP), were evaluated with respect to the three basic safety functions: control of sub criticality, decay heat removal, and confinement of radioactivity. In case of the HFR all three safety functions apply, whereas for all other facilities mainly confinement of radioactivity is at stake.

For the facilities on the OLP no design basis flood was defined in the past. Therefore the adequacy of resistance against flooding was analysed on the basis of the impact on safety functions of an off-site and/or on-site flood using various postulated flooding sources.

The OLP is located in a dune area. The North Sea is to the West and the Zijper Polder is towards the East. The distance between the North Sea and the relevant buildings ranges from 250 to 500 m. A minimum of two rows of dunes separates the buildings from the sea. At the Zijper Polder side a third row of dunes marks a sharp transition towards the polder. The dunes themselves reach an elevation between approximately 8 and 16 m + NAP. The Zijper Polder is at 0 m + NAP. Terrain levels vary between 2 and 15 m + NAP. The minimum elevation of the buildings containing relevant safety functions is 4.1 m + NAP.

In case of an extreme storm surge resulting in a breach of dikes or dunes, it is likely that the Zijper Polder would be flooded, while the OLP itself will not be affected and effectively becomes an island. In such a case, it is almost certain that the OLP loses all off-site power. In that situation, the safety related structures, systems and components of the HFR are certainly sufficient to provide a window of at least 25 days (600 hrs.) for remedial actions. For the other nuclear facilities emergency diesel generators of the OLP can supply power for 42 hours before refuelling is necessary and which provide ample time to safely shutdown the operations in these facilities.

When the OLP itself becomes flooded the safety margin is determined by loss of electrical power which starts to occur at flooding heights of 4.1 m +NAP.

Extreme Weather Conditions

The resistance of the HFR and associated nuclear facilities against postulated severe weather conditions was evaluated. The design basis was established using the original construction strength calculations. In case of missing building data, engineering judgement or data of a building comparable in construction with known properties were used. In many cases severe weather conditions were not included in the original design base.

The assessment shows that no cliff-edges exist with regard to 1) water temperature of the Noordhollands Kanaal, 2) formation of ice on the Noordhollands Kanaal, or 3) air temperature. With regard to snowfall, for the HFR and the majority of OLP buildings no cliff edge exists. For effects of wind missiles and hail and salt deposition, as well as for lightning and combination of weather conditions, no quantifiable margin can be given. Based on construction strength calculations, most OLP buildings appear not to be designed to resist the postulated extreme wind gust (57 m/s) and rain conditions (133 mm within 48 hrs.), except the Reactor building. Due to the nature of assumptions in the assessment, these results indicate that for these types of weather conditions, exceeding potential cliff-edges could lead to consequences to all other OLP facilities except for LFR and STEK hall. However, based on the observed factual meteorological data, the performance of the safety systems under severe weather conditions in the past 30 years and the expected interference with safety functions, the consequences are expected to be very limited.

None of the analysed combinations of extreme weather conditions is considered to lead to loss of safety functions.

Loss of electrical power and loss of ultimate heat sink

The HFR and associated nuclear facilities on the OLP are protected against a loss of electrical power supply. Apart from the emergency power supply system with 3 redundant emergency diesel generators that act as double back-up with load shedding possibilities up to 89 hrs. For the HFR, back-up battery systems are available that can supply power up to 3 hrs. These systems are protected against earthquake and flooding (4.1 m +NAP). The installation and equipment currently present is adequate to provide electric power for the relevant safety systems to ensure safe shutdown of the HFR and associated nuclear facilities.

In case of loss of ultimate heat sink, the HFR is able to cool the core (after reactor shutdown) and spent fuel by systems that are available on site supported by the external Provincial water grid (PWN) for at least 25 days (600 hrs.). Additional possibilities have been identified to extend the margins of the emergency power supply in case of extreme events.

Severe Accident Management

The Fukushima Dai-ichi accident caused a paradigm shift concerning management of severe accidents at nuclear facilities. In view of the management of severe accident at the nuclear facilities on the OLP, both the on-site emergence response organisation as well as its means have been reassessed. For each of the nuclear facilities emergency procedures and measures exist to manage numerous accident and emergency conditions. The HFR, the most prominent nuclear facility on the OLP, comprises many inherent safety features: it has a tank in-pool and is operated at low pressure and temperatures, providing days to weeks leeway to perform restoration activities in all feasible severe accident conditions. To even further strengthen accident management possibilities, currently a remote monitoring system (RMS) is being installed, providing the same monitoring information as the HFR control room (available mid-2012). All other nuclear facilities on the OLP are protected by sturdy structures against all but the most severe external influences, and provided with fail safe isolation systems and adequate fire extinguishing detection and means at relevant locations. Two widely separated fully equipped emergency coordination centres (ECC) assure the redundancy of the on-site emergency response organisation. A fully equipped and well-trained corporate fire brigade is permanently stand-by on-site to act as fast response force to limit and mitigate any incidents on-site.

The on-site emergency response organisation is tuned in with the regional coordination centre for disaster response. In case the consequences of an accident threaten to exceed the site level, the response coordination is escalated to the regional or ultimately national level, if deemed necessary.

As result of the reassessment of the severe accident capabilities management of the on-site emergency response organisation possibilities have been identified to improve the margins. An obvious challenge is associated with maintaining emergency response capabilities in case of long-lasting absence of external supplies (electricity, tools and equipment, drinking water, food), communication and assistance (fire brigade, medical support, manpower, etc.). With respect to the possible uncertainty in the margins concerning criticality safety under extreme conditions, the currently available studies will be extended to encompass the impact of possible deformation due to extreme events.

Other Extreme Hazards

As an additional requirement of the Dutch safety authorities on top of the ENSREG requirements, a list of ten so-called other extreme hazards has been analysed for the safety margin assessment of the OLP. These hazards comprise: internal explosion, external explosion, internal fire, external fire, airplane crash, toxic gases, large grid disturbance, failure of systems by introducing computer malware, internal flooding and blockage of the secondary cooling system inlet. In general it can be concluded the HFR and associated nuclear facilities on the OLP are well equipped to handle these events due to redundancy in safety systems. In addition, safety systems are in generally designed as “hard-wired” without digital equipment connected to the internet. Two potential external explosion events were identified for further investigation. With respect to an airplane crash, the currently existing analyses will be extended.

Measures which can be envisaged to increase robustness of the nuclear facilities on the OLP

Combinations of extreme situations indicated that there are areas at the OLP where possibilities exist to enlarge the safety margins. These are elaborated in the report and summarized in the following table.

Implementation of these measures will most probably require hardware modifications	
M1	Installation of seismic instrumentation to notify the HFR control room and if necessary initiate a scram. This would increase the HFR's robustness in case of an earthquake.
M2	Securing the power supply of the Emergency Communication Centre ("INO room") under extreme conditions would strengthen the Emergency Response Organisation (ERO).
M3	The use of autonomous wireless battery-based techniques on site and satellite-based communication systems for off-site and emergency voice and data communication would strengthen the ERO.
M4	Bolting down of waste tanks in the HCL-MPF, JGL and HFR would improve the robustness of the waste tanks in case of flooding and or seismic events.
M5	Installation of external connections for auxiliary diesel generators for power supply to vital components would increase the margin in case of Station Black-out.
M6	(Additional) leakage detection systems should be installed in the HFR Reactor Outbuilding, HFR Primary Pump Building (PPG) and WSF.
M7	A possibility to remotely control/add PWN water flow to the pool of the HFR and storage pool at HCL will increase the margin to control fuel damage or shielding/confinement.
M8	A possibility to remotely control and operate the convection valves at the HFR would overcome a potential cliff-edge with respect to emergency cooling abilities.
M9	Introduction of emergency-proof on-line monitoring system at the STEK facility would increase the margin to prevent/control releases in case of events that damage stored waste containers.
M10	The installation of a Remote Monitoring System for the HFR will be completed by mid-2012 and will provide the capability to monitor the HFR under extreme adverse conditions.
M11	The protection of the control room against toxics, smoke etc. would improve the margin in case of other events.
M12	Provisions will be made to add neutron absorbers to improve HFR reactivity control in emergency situations.
M13	The storage facility of fissile material in the WSF trenches should be strengthened in such a way that deformation is excluded under all accident conditions.
M14	Provision of additional locations of cabinets with alarm procedures ('alarm roles') and other contact information will created redundancy in Severe accident Managen.t
M15	Installation of a C2000 emergency network repeater inside the HFR would increase emergency network capability
M16	The possibility to provide long-term provisions for employees (e.g. food supply, sheltering) would increase the margin for the ERO.
M23	The possibility to improve the level of autonomy of the Emergency Response Organisation (ERO) would enhance the capability to handle threatening external events and severe accidents in case of lack of external emergency support due to extreme conditions in the hinterland.
M25	The possibility to assign a location for sheltering and treat (un)contaminated and possibly injured people on site would improve the capability of the ERO to handle severe accidents in case of unavailability of external emergency support due to extreme conditions in the hinterland.

In the framework of the CSA, the maximum resistance of the OLP against external events has been investigated, whereas traditionally the OLP design is investigated against certain pre-defined external events. These different approaches require different analysis and studies. Within the timeframe of the CSA, it was not possible to perform extensive studies and in some cases engineering judgment has been applied for establishing the margins. In general a conservative approach is chosen when applying engineering judgment. In some areas, additional studies could therefore reveal that the actual margins are larger than those margins presented in the CSA report. Furthermore, in some cases, additional studies could reveal measures for further increasing the margins.	
S1	The currently used Design Based earthquake will be evaluated against the current IAEA requirements and recommendations established in the IAEA.
S2	The seismic site characterisation OLP will be finalised in reference to the current requirements and recommendations established in IAEA safety related standards.
S3	A systematic seismic qualification of the SSCs of the HFR consistent with international accepted requirements and guidelines will be made. Included under this point are the recommendations from the recent INSARR mission to the HFR.
S4	The existing seismic analysis of the HCL building structure and related components will be extended.
S5	The possibilities to replenish diesel fuel and potable water supply during flooding conditions will be investigated.
S6	The possibility to seal the penetrations of the cabling duct between the Primary Pump Building and the Reactor Outbuilding to reduce leakage to the Reactor outbuilding will be evaluated.
S7	The uncertainty of the extreme weather margins can reduced by including certain missing weather conditions in the original design basis. Building specifications with respect to extreme weather conditions that not yet have been retrieved should be identified and subsequently margins and potential cliff edges should be re-evaluated.
S8	The possibility of injection of (sea)water in the secondary cooling system will be evaluated
S9	The possibility to inject water to the pool water system via the Stortz couplings to extend the margin with respect to available pool water volume will be evaluated
S10	The extent to which gravity driven supply of PWN water can be applied to replenish the pools or to re-establish the water level of the pools to the level of the water stocks of PWN ("niveau reinwater kelders") will be evaluated, meeting all constraints for direct refill of the pools by PWN.
S11	The necessity to increase battery capacity to enlarge monitoring and recording time will be investigated for relevant safety systems.
S12	The potential radiological impact on emergency workers due to lowered water levels in the HCL storage pool will be analysed.
S13	The currently available criticality safety analyses for the various storage locations for fissile material will be extended in the light of possible deformation of the storage facilities due to extreme events.
S14	An analysis will be performed to evaluate the need and feasibility of a secondary communication room in case of failure of the main CAS-room (centrale meldpost)
S15	The possibilities to improve availability of replacement staff and internal resources that can be used in case of severe accidents will be made.

S16	Evaluation of the possibilities to prepare and conduct countermeasures like use of radiation shielding, covering of debris, collect, store, and process contaminated water, decontamination of equipment and persons in case of severe accidents.
S17	The effect of internal explosions on the safety functions of the HFR besides loss of containment will be investigated.
S18	The on-going investigations on the explosives present in the naval artillery test range during testing will be monitored, and it should be made sure that the safety functions of the OLP nuclear facilities remain unaffected during accidents with these explosives.
S19	The effects of an explosion/fire originating from the natural gas pressure reducing station located on the OLP will be analysed.
S20	The fire analysis of the HFR will be updated.
S21	The analysis of the consequences of radioactive release caused by air plane crash on the HFR will be extended under the assumption that all pool water is lost.
S22	All safety systems will be checked for the presence of PLCs and their vulnerability for malware. A general policy of checking vulnerability for malware with facility hardware adaptations and upgrades should be established.

The CSA showed that the robustness of the OLP against external hazards can be increased by implementation of a number of procedures.	
P1	All existing procedures with respect to the impact of flooding will be reviewed. Procedures with respect to the identified bottlenecks will be adapted.
P2	A procedure to safely shut down nuclear facilities in case of defined extreme weather conditions will be established.
P3	To meet the 72 hours operation recommendation (ENSREG), the minimum required volume of the main fuel stock will be kept at a minimum of 8.1 m^3 for operation of diesel generator A and B for that time frame, or at a minimum of 8.6 m^3 for 72 hours of operation for all three engines.
P4	A set of procedures will be developed (or the existing set enhanced) to address the following issues in case of LUHS:
	· Heating up of the primary system
	· Related switch over to core cooling by pool water
	· Replenishment of pool water
	A training program will be implemented.
P4	A set of procedures will be developed (or the existing set enhanced) to address the following issues in case of LUHS-SBO:
	· Supply of pool water for core cooling
	· Replenishment of pool water by the fire brigade
	A training program will be implemented.
P6	A set of Accident Management Procedures as supplement to the existing procedures ("bedrijfsvoorschriften") will be developed and a training program should be implemented. Examples of recommended issues to be addressed are:
	. accident management measures which are possible at the HFR, but currently not mentioned in a separate procedure
	. possible leak repair methods for larger pool leakage
	. use of autonomous mobile pumps
P7	A Function Restoration Procedure for maintaining the containment function of the HFR will be developed (as supplement to the existing procedures).
P8	Develop function restoration procedures for maintaining the confinement function of the HCL-RL, the HCL-MPF, JGL, WSF, DWT and STEK buildings.
P9	The placement of the Cd plate next to the HFR vessel will be trained in the current situation.
P10	Provide clear instructions in case of temporary failure of CAS.
P11	Provide guidance on GSM-based on-site communication, in case of emergency
P12	Provide procedures for triggering of ERO by external events.
P13	Provide procedures for the management of accidents in case of inaccessibility of OLP for external emergency organizations.
P14	Define procedures that facilitate mutual support between facilities in case of severe accidents
P15	Establish a communication protocol from the regional fire brigades to the OLP in case of fire or other external hazards.

Introduction

As a consequence of the accident at the Fukushima Dai-ichi Nuclear Power Plant (NPP) in Japan, the European Council meeting on March 24^{th} and 25^{th} 2011 declared that ‘the safety of all EU nuclear plants should be reviewed, on the basis of a comprehensive and transparent risk assessment (‘stress tests’)”.

On the basis of the proposals made by the Western European Nuclear Regulator 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”. The “EU ‘stress test’ specifications” were finally provided by ENSREG.

These EU wide tests will be an addition to the safety standards already in place at national level. The aim of these tests is a targeted re-assessment 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.

NRG has agreed with the regulator to participate in the stress test on a voluntary basis. The Regulatory Body has sent NRG, the licensee of the nuclear facilities on the Petten site, on June 1^{st} 2011 a letter in which they request to perform a stress test based on the ENSREG specifications (see Appendix: A.1 and A.2).

In addition the Ministry indicated that in the assessment ‘deliberate disturbances’ should be taken into account. This request was implemented by NRG as the Complementary Safety margin Assessment (CSA) “Onderzoekslocatie Petten (OLP)” of which the results are presented in this report.

The objective of the CSA is to perform a reassessment of the design basis, the safety margins and cliff-edge effects of the HFR research reactor and associated nuclear installations on the OLP and is mainly based on the guidance provided by ENSREG regarding the so-called ‘stress-test’. This assessment consists on the one hand of an evaluation of the response of the HFR research reactor and associated nuclear facilities facing a set of extreme initiating events like earthquake and flooding, and on the other hand of a verification of the preventive and mitigating measures that ensure the safety of the HFR research reactor and associated nuclear facilities.

The CSA focuses on extreme natural events like earthquake and flooding. It will also assess the consequences of loss of safety functions due to indirect initiating events like a large disturbance from the electrical power grid impacting AC power distribution systems, external fire or airplane crash. Furthermore, an assessment of man-made and of other extreme natural events has been carried out. For extreme natural events (i.e. extreme weather conditions) and man-made events without an intentional character, this assessment will initially consist of determining the relevancy of such events in comparison with the two initiating events (earthquake and flooding). Relevant aspects of these other natural and man-made events were evaluated in more detail. Aspects concerning security are not dealt with in this assessment.

The CSA includes a verification of the preventive and mitigating measures identified following a defence-in-depth logic: initiating events, consequential loss of safety functions, severe accident management. In addition, logistic problems caused by local, regional and possible national chaos due to extreme natural events are taken in consideration. In these extreme situations, sequential loss of the lines of defence is assumed, in a deterministic approach, irrespective of the probability of this loss.

The nuclear installations included in the CSA are the High Flux Reactor (HFR), the Low Flux Reactor (LFR), the Hot Cell Laboratories (HCL) (including the Molybdenum Production Facility), the Decontamination & Waste Treatment Facility (DWT) and Waste Storage Facility (WSF). The potential interference of the installations during the events was included on the basis of the PIE's, DBA's and BDBA's as defined in their current license documents. In this respect, the worst case source term per facility as typically referred to in the existing license has been used in the assessment.

The outline of the CSA was presented to the International Safety Expert Team (ISET) on 14th November 2011. The members of the ISET were satisfied with the scope and outline of the CSA. The ISET will be informed about the outcome of the CSA and how the findings are being dealt with.

The licensee has the prime responsibility for safety; hence it is up to the licensee to perform these assessments and up to the regulatory bodies to independently review them. The assessment leads to insights into severe accident conditions and how the OLP reacts, even if the emergency measures provided for that situation would fail. As a result, the assessment delivers the following insights:

- The performance of the nuclear installations on the OLP and safety management systems under escalating accident conditions where protective measures and systems are considered to sequentially fail;
- The robustness of installations and its safety management systems;
- The potential for modifications to improve the robustness of the installations and safety management systems.

This final report gives the results of the Complementary Safety margin Assessment (CSA) “Onderzoekslocatie Petten (OLP)” including conclusions and the potential to increase the robustness of the OLP.

General safety policy

Within NRG, safety, including nuclear safety, has an overriding priority. NRG has a nuclear safety policy which is formalized through various policy statements. This is implemented through a pro-active approach to prevent and control possible hazards with respect to safety of individuals and the environment. The management of NRG carries out a role model to strictly follow and maintain rules and regulations, licenses, procedures and instructions and promotes the continuous improvement of the (safety) management system.

NRG is continuing improving its safety culture. This implies that safety is not only based on rules and regulations, but becomes an organisational goal with emphasis on continuously improvement. In this context, there is a strong emphasis on communication, training, including emergency drills, management style and improving efficiency and effectiveness.

With regard radiological hazards, this implies that all actions are intended to minimize exposure to the risks of (radiological) hazards, both for individuals and the environment. All this is achieved by setting up and maintaining an effective defence mechanism against (radiological) hazards. This defence mechanism includes the 3 principles for dose limitation recommended by the International Commission on Radiological Protection (ICRP), namely justification, optimization, and dose limitation. Justification of a practice means that any proposed activity that may cause exposure to persons should yield a sufficient benefit to society to justify the risks incurred by the radiation exposure. Optimization of the practice means that the radiation exposures resulting from the practice must be reduced to the lowest level possible considering the cost of such a reduction in dose (“the As Low As Reasonable Achievable” principle; ALARA-principle) and the use of dose limits which are setting the upper limits on the dose that may be received by any member of the public from all man-made exposures other than medical exposures and are imposed by regulatory agencies.

Periodic safety reviews

Every ten years, an extensive safety evaluation (10-EVA) is performed on nuclear safety and radiation protection. The objective of such a ten yearly safety evaluation is compare the existing situation to the state-of-the-art requirements and best practices using a comprehensive assessment as to whether the design basis and the safety documentation are still valid, arrangements in place to ensure the plant’s safety are still correct and effective, and the research reactor conforms to current national and international safety standards and practices.

The goal of this periodic safety review is to improve the design of the research reactor and its operation, in such a way that the nuclear safety and radiation protection performance will increase. This means that the reactors design is, as far reasonable achievable, in accordance with the latest technical design levels for modern nuclear reactors and is operated in line with the latest safety guidelines and best practices.

In addition to this periodic safety review, The HFR is subjected to periodic integrated safety assessment of research reactors (INSARR) carried by expert teams from the International Atomic Energy Agency (IAEA). Following the INSARR 2005 and Safety Review Missions on the bottom plug liner repair in 2010, an INSARR was carried out in 2011. The recommendations of this mission are integrated in the second extensive safety evaluation (10-EVA 2012) currently in progress. The 10-EVA 2012 is due by

The underlying CSA is line with NRG's policy that nuclear safety has an overriding priority. This is expressed in the pursuit of excellence through continuous improvement. The outcome of the CSA will lead to the implementation of recommendations resulting in improvement.

Context of the CSA OLP (EU- ENSREG, appendix A.2)

Facilities and Installations

Although ENSREG prescribes only the assessment of (power) reactors, the CSA has been carried out for all nuclear facilities on the OLP. This extension of the scope, requested by the Dutch competent authority (see Appendix: A.1), is justified by the inherent risks of the installations on the OLP for significant radiological releases and the potential for mutual effects.
The following nuclear facilities located on the research location Petten (OLP), operated by NRG have been assessed:

- Decontamination and Waste Treatment Facility (DWT);
- Jaap Goedkoop Laboratory (JGL);
- High Flux Reactor (HFR);
- Hot Cell Laboratory (HCL) including the Molybdenum Production Facility (MPF);
- Low Flux Reactor (LFR);
- Waste Storage Facility (WSF);
- STEK hall (STEK).

The risks assessed as part of the CSA are only related to the nuclear nature of the design and operation of the facilities: radiation and radioactive release. In the assessment, the focus is on core and fuel damage, potential external exposure to radiation and the release of radioactive materials with subsequent dose consequences. As part of the assessment, the potential effects associated with the risks will be further quantified. The individual facilities differ in the risks that need to be assessed:

- Damage to the HFR core due to re-criticality that has the potential to lead to significant external exposure to radiation and the release of significant amounts of short and long-lived fission products and actinides;
- Damage, including criticality, to the Spent Fuel (SF) stored in HFR or HCL that can lead to external exposure to radiation and release of long-lived fission products and actinides;
- Damage to the containment(s) and/or confinement(s) of HFR, HCL, JGL, WSF, DWT, STEK hall or LFR that might result in a release of radioactive materials and/or an increase of radiation levels.

The scenarios to be considered will link events with the potential consequences broadly categorised as above. Logically, the larger the risk involved, the more complicated the scenarios will be, due to the number of safety functions and systems included in the design to prevent, control or mitigate that risk.

Besides the nuclear facilities also support facilities have been identified that were included in the assessment. These are a.o. the fire department, Forum and health physics building.
For the facilities on the OLP site, the assessments will be based on the situation as of August 31st 2011: the so-called 'as-is' situation. Modifications to installations after this date are not included. ENSREG prescribes June 30th 2011 as the 'as-is' date. However, the stress test for the OLP follows a different time line and consequently sets the 'as-is' date at August 31st 2011 which is more realistic given the completion of safety relevant modifications within this period. For the LFR it means that the fuel is removed from the reactor and stored in the fuel storage pipes in the Fermi building. However, the experimental conversion plate containing 18 LFR fuel plates is currently stored at a shielded position at the front side of the irradiation trolley of the LFR.

Technical scope

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;
- Other extreme events.

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 confinement/containment integrity.

b) and c) are not limited to earthquake and tsunami as in Fukushima since flooding is included, regardless of its origin. Furthermore, extreme weather conditions have been included (see chapter 4). Also, the assessment of consequences of loss of safety functions is relevant if the situation is provoked by indirect initiating events such as a large electrical power grid disturbance impacting AC power distribution systems, an external fire or airplane crash. In the Netherlands, other events have been included in this list.

The review of the severe accident management issues focuses on the licensee's provisions including relevant planned off-site support for maintaining the safety functions of the nuclear facilities. 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 will be only addressed in the context of logistic problems caused by local, regional and possible national chaos due to extreme natural events.

Assessment methodology

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

The assessment approach is essentially deterministic: when analysing an extreme scenario, sequential loss of protective systems and measures is assumed.

The assessment considers three elements:

- Provisions incorporated in the design basis and the facility'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 always 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 OLP reacts, even if the emergency measures provided for that situation would fail. This means that for the determination of the safety margins, a deterministic approach is chosen. The intention is that an increasing threat (for example, an increasingly higher tidal wave or more serious earthquake) is assumed, so as to determine how the OLP and its safety management system respond and to what level of threat the safety systems work adequately. Following a deterministic approach for assessing the safety margins for the different issues, it is important to know the likelihood of such an event as basis for possible further evaluation and improved measures. This information is also presented in this report.

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

Earthquake

Using a postulated characteristic earthquake, the so-called Design Basis Earthquake (DBE), the response of the HFR and other nuclear facilities was assessed and evaluated. The evaluation of the consequences of the DBE is based on a compilation of existing documents and reports that exist for the HFR and other nuclear facilities on the OLP site. 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 the OLP are derived and finally an estimate of the seismic margin is given.

Flooding

For the OLP no design basis flood is defined. Therefore the adequacy of protection against flooding conditions on safety functions is assessed using increasing flooding levels.

The flooding conditions which are considered are:

- high tides and storm surges;
- tsunamis;
- extreme rain fall;
- on site pipe rupture;
- credible combinations of the items mentioned above.

Extreme weather conditions

The following weather conditions are taken into account:

- High and low-water temperature of the Noordhollands kanaal;
- Extremely high and low air temperature;
- Extremely high wind (including storm and tornado);
- Formation of ice;
- Heavy rainfall;
- Heavy snowfall;
- Lightning;
- Credible combinations of the weather events listed above.

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

Loss of electrical power supply and loss of ultimate heat sink

Impact of loss of electrical power supply or of loss of the ultimate heat sink should be dealt with for all nuclear facilities that exist at the NRG premises in Petten, for an increasing level of severity of these events and/or their combinations.

For loss of electrical power supply this means that a distinction is made between:

- loss of offsite power (LOOP), i.e. that there is no supply from external grid(s);
- Loss of offsite power and emergency AC power supply (SBO1), i.e. that in addition to the loss of external grid(s) connection also the ordinary emergency power system providing AC power is not available
- Loss of all electrical power supply (SBO2), i.e. that in addition to SBO1 also the “permanent installed diverse back-up AC power sources” are not available

For loss of the ultimate heat sink this means loss of cooling by the water of the Noordhollands kanaal. An alternate ultimate heat sink, as it is indicated by ENSREG, does not exist at the OLP. Therefore this situation will not be considered.

Severe accident management

'Severe accident management' refers to the overall range of capabilities of the nuclear facilities on the OLP and its personnel to:

- prevent the escalation of an Beyond Design Basis Accident (BDBA) into a severe accident;
- mitigate the consequences of a severe accident (e.g. those that might lead to severe fuel damage in the reactor core);
- to achieve a long term safe stable state.

A subset of accident management measures deals with the management of severe accidents. It objectives are to:

- limit or mitigate core damage after its occurrence;
- maintain containment/confinement as long as possible;
- minimize on-site and off-site releases, and return the plant to a controlled safe state.

The accident management programme comprises plans and actions undertaken to ensure that the facilities and their personnel with responsibilities for accident management are adequately prepared to take effective on-site actions to prevent or to mitigate the consequences of a severe accident.

In line with the specifications of the stress test published by ENSREG (Appendix A.2), the following postulated situations are considered:

- loss of HFR shutdown ability;
- loss of HFR core cooling abilities;
- loss of cooling function of HFR spent fuel storage;
- loss of HFR, HCL or WSF containment integrity after fuel damage;
- loss of containment integrity for all facilities.

The description and assessment with respect to accident management that have been carried out, cover:

- Accident management measures currently in place at the various stages of a scenario of a loss of HFR shutdown ability;
- Accident management measures currently in place at the various stages of a scenario of a loss of HFR core cooling function;
- Accident management measures currently in place at the various stages of a scenario of a loss of cooling function in the spent fuel storage;
- Accident management measures and facility design features for protecting the containment function after an occurrence of fuel damage;
- Accident management measures currently in place to mitigate the consequences of a loss of containment integrity of all nuclear facilities.

The assessment also addresses:

- Availability and the possibility to deploy available resources during accident conditions;
- Availability of personnel such as fire fighters, installation experts and radiation protection specialists;
- The necessary means such as communication systems, compressed air and fire fighting equipment;
- Infrastructure/utilities such as drinking water supply and electrical power;
- Alternative resources that can be made available;
- Logistics of offsite resources in the context of regional or national chaos.

The impact of extensive destruction or hazardous work environments on the accident management and mitigating actions will be evaluated. A timely response to accident situations is important to avoid further aggravation of the situation. Therefore the assessment includes an evaluation of the time before the next significant failure or possible cliff-edge effects.

Other extreme events

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

- explosion and fire related hazards:
    - internal explosion;
    - external explosion, including a sinking ship with explosion and a resulting tidal wave;
    - internal fire;
    - external fire;

    - airplane crash;
    - toxic gases;
- electrical related events:
    - large grid disturbance;
    - millennium-bug kind of failure of systems, renamed as failure of systems by introducing computer malware;
- water related events:
    - internal flooding;
    - blockage of cooling channel inlet;

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

Security issues, an analysis, which is conducted by the government, will not be made public. Consequently, this will not be part of this report. Therefore, manmade violence against the nuclear facilities on the OLP of any form is not mentioned explicitly in this report.

Project organisation

By order of the CEO, NRG established an experienced project team, led by a project manager and supervised by a project board. In addition, an External Review Committee (ERC) was established that advised the project manager on the preparation and execution of the CSA. In addition the ERC evaluated and reviewed the project brief, project plan and scenario document related to the CSA. Secondly, the ERC evaluated and reviewed the final CSA report on consistency with the European Stress test specifications and on technical merit. The External Review Committee consisted of Prof. dr. ir. T.H.J.J. van der Hagen, Prof. dr. W.C. Turkenburg, Prof. dr. ir. A.H.M. Verkooijen and ir. J. Versteeg and represents members from outside the nuclear environment and from outside NRG.

The NRG Reactor Manager of the HFR, responsible for a safe operation of the HFR, was delegated sponsor of the CSA and responsible for results, and in general for the technical quality of the report. To ensure the necessary expertise and resources to generate the CSA report, experienced parties within NRG have taken part in the project from the beginning.

Independent review of the final report and selected underlying reports has been performed by the External Review Committee, the Reactor Safety Committee and, depending on the contents, by the HFR Safety Committee (in parallel).

1 General data about the site

1.1 Brief description of the site characteristics

The current HFR and NRG facilities are situated on an industrial site ('Onderzoekslocatatie Petten" (OLP)) in the North Holland dunes to the north of Petten (municipality of Zijpe). The site houses the Joint Research Centre Institute for Energy (JRC-IE) of the European Commission, the Energy Research Centre (ECN), the Nuclear Research and consultancy Group (NRG) and Covidien (former Mallinckrodt Medical)(Figure 1-3).

The part of the site on which these organisations are located covers about 80 hectares. A large part of the site was leased to ECN by the State by deed of 26 April 1958, most recently amended by deed of 23 December 1996. The lease has recently been extended to 2032. JRC-IE was granted a lease for both the HFR and the JRC-IE site for 99 years by deed of 31 October 1962.The use of the site is defined in the "Petten research and industrial site" land destination plan of 24 January 1997 as laid down by the Zijpe local council. This destination plan specifies which part of the Petten site can be used for industrial activities.

Several types of nuclear facilities are located on the OLP which are operated under the Nuclear License of NRG. These facilities are:

- High Flux Reactor (HFR), a 50 MW_{th} tank-in-pool-type multipurpose reactor. Construction started in 1957, first critical in 1961;
- Low Flux Reactor (LFR), an 30 kW_{th} Argonaut type reactor, Construction started in 1959, first critical in 1960, end of service 2010, fuel elements were removed mid-2011 and placed in the fuel storage pipes in the FERMI building. The experimental conversion plate of the LFR containing 18 LFR fuel plates is currently stored at a shielded position at the front side of the irradiation trolley of the LFR.
- Hot Cell Laboratories (HCL), consisting of a research laboratory (HCL-RL) and a Molybdenum Production Facility (HCL-MPF); operational since 1964 (HCL-RL) and 1995 (HCL-MPF).
- Decontamination and Waste Treatment Facility (DWT); operational since 1962 for decontamination and waste (water) treatment;
- Waste Storage Facility (WSF); operational since 1962, in use for intermediate waste storage;
- STEK Hall (STEK); operational since 1962, in use for short-term waste storage;

- The Jaap Goedkoop Laboratory (JGL); a type B radionuclide laboratory, operational since 2007.

The industrial site of these facilities lies in an area of dunes approximately 2 km north of Petten (Zijpe municipality) in the province of North Holland, some 50 km north-west of Amsterdam (Figure 1-1). The dune area, locally about 1 km wide, forms the dividing line between the North Sea and the Zijper Polder to the east.

On the eastern side runs the dividing line between the Zijperzee dyke and the Westerduinweg road. On the northern and southern side, the site adjoins the neighbouring dunes of the Dutch Forestry Commission. The site is separated from the North Sea on the western side by a small stretch of dunes about 250 metres wide (Figure 1-2). The whole industrial site is surrounded by a fence. Between the JRC-IE site and the ECN, NRG and Covidien site there is also a fence with two gates. In addition, the HFR is surrounded by a double fence.

Both the JRC-IE site and that of the ECN, NRG and Covidien are accessible from the public highway for people and goods via one gate per site. Both gates are guarded permanently.

The aerial photograph shows the way the premises are embedded in the dunes and the surrounding area (Figure 1-2). The map shows the location of the HFR complex and the NRG facilities on the OLP. The HFR and other nuclear facilities of NRG lie in an area of dunes whose height/position varies between about 2 m and 15 m above Normal Amsterdam Level (NAP). This area of dunes encompasses a stretch of about 1 km along the coast and adjoins a polder area behind it. The transition to the polder area (about 0 m NAP) is very abrupt.

There is limited industrial activity in the immediate surroundings of the site. About 5- 7 km south , there is gas exploration in Groet en Bergen and about 10 km north, the pipeline between Bacton Gas Terminal , United Kingdom and the Netherlands enters the Netherlands at the North Holland compressor station in Anna Paulowna. The nearest major industrial activities are situated along the Zaan River and the North Sea Canal, both at a distance of about 35 km from the site (Figure 1-1: Map of the Province North Holland with location (A) of the "Onderzoekslocatie Petten"

). The nearest medium-sized industrial activities are in Den Helder and Alkmaar, both at a distance of about 18 km. There is also a large naval base in Den Helder.

A secondary road (N502) runs along the eastern border of the OLP site. Parallel to this road, at a distance of approximately 2 km, runs a major local road from Alkmaar to Den Helder (N9). The nearest track and train station is at Schagen, at a distance of about 8 km.

Transport by water occurs on the Noordhollands Kanaal at a distance of approximately 2 km. There are no shipping lanes by sea close to the Petten coast west of the OLP. The lanes for various ships including container ships are about 27 km from the coast, and those for ships carrying hazardous chemicals about 90 km from the coast.

There is no regular civilian air transport route over the Petten site. There are therefore few aircraft movements over the site. The nearest airfield for military and small (private) aircraft is De Kooy, located

Figure 1-1: Map of the Province North Holland with location (A) of the "Onderzoekslocatie Petten"

near Den Helder, at a distance of about 18 km from the OLP site. Military aircraft in principle will not fly over the premises, but since they do not have fixed flight paths it cannot be excluded.

Directly adjacent to the OLP site, a naval artillery test range with a mechanically fixed cannon in the sea-directed position is in use by the navy. Ammunition for the shooting exercises is carried by road via the ECN side of the OLP and thus at some distance from the HFR.

The area east of the OLP is a mainly agricultural area, while the area south and north is mainly dunes with small forest areas.

Figure 1-2: Aerial photograph of the "Onderzoekslocatie Petten"

The village of Petten (number of inhabitants: 1,645) is located about 2 km south of the site. The cities of Schagen, Bergen, Alkmaar en Den Helder lie at distances of 11, 11, 18 and 18 km respectively. Their number of inhabitants is 18,770; 30,870; 93,940; and 57,210 respectively.

Figure 1-3: Map of the "Onderzoekslocatie Petten" (OLP) with colour indication of the various institutions on the OLP

1.2 Main Characteristics of the nuclear facilities

1.2.1 The High Flux Reactor (HFR)

Figure 1-4: Aerial photograph of the HFR site

The High flux Reactor (HFR) was designed, developed and built by the American Car and Foundry Industries Inc. on the initiative and at the expense of the former Reactor Centre Netherlands (RCN) as a facility for the Dutch nuclear research and development programme. On the basis of an agreement between the Dutch Government and the Commission of the European Communities of 25 July 1961, the Petten site was established as one of the four Joint Research Centres. On 31 October 1962 the HFR was transferred to the European Atomic Energy Community (Euratom). Since then, the operation and use of the reactor have formed part of the research programmes of the Commission of the European

Communities. From that time on, the reactor was operated by RCN, later renamed Energy Centre of the Netherlands (ECN) under the terms of a contract between RCN and the Commission of the European Communities. Operation of the reactor was transferred in 1998 from ECN to the Nuclear Research and consultancy Group (NRG), a full daughter institute of ECN. In February 2005, following the conversion of the HFR to LEU as part of the worldwide non-proliferation policy as well as major modifications, NRG became also the legal license holder of the HFR. An overview of the HFR site in given in Figure 1-4.

The reactor became critical for the first time on 9 November 1961, following a construction period of about five years, and the 20 MW power level was first reached on 25 May 1962. The major operational milestones in the history of the HFR are as follows:

- increase of reactor power from 20 to 30 MW on 8 May 1966;
- increase of reactor power from 30 to 45 MW on 20 February 1970;
- replacement of the reactor vessel after 21 years' operation in the period from 30 November 1983 to February 1985;
- Conversion from HEU to LEU in period October 2005 to May 2006.

The High Flux Reactor (HFR) is a 50 MW_{th} tank-in-pool-type multipurpose reactor. The reactor serves as a neutron source for civilian, technological and scientific research and for the production of radioisotopes for medical and industrial applications. This reactor uses plate-type fissile materials with low-enriched uranium and burnable neutron poison, light water as moderator/coolant and beryllium as reflector. Characteristic technical features of the reactor are that the reactor vessel is only slightly pressurized and is reactor vessel is placed under water in a pool.

1.2.1.1 Reactor core assembly

The reactor core is composed of 72 core positions within the reactor vessel that are often occupied as follows (see Figure 1-6 for a characteristic core configuration):

- 33 low enriched (LEU) fuel elements;
- 6 control rods;
- 10 or 12 beryllium reflector elements;
- 4 unique corner elements containing a beryllium plug and
- 17 or 19 standard aluminium filler elements, whether or not filled with experiments or production facilities. Filler elements not used usually contain a standard aluminium plug.

Furthermore, on the east side of the core an external reflector row is located.

There are three pools (see Figure 1-5). The reactor vessel is placed in the reactor pool (151 m^3) . The other two pools (106 and 84 m^3) serve for the storage and handling of spent fuel and radioactive material, including radioisotopes. The water in the pools serves as shielding and to cool the radioactive objects located in the pools. There are several positions in and close to the reactor vessel for the purpose of experiments, radioisotope production and other irradiations. Experiments can be placed in special filler elements or sample holders in the core and against the outer wall of the reactor vessel at the west side of the core.

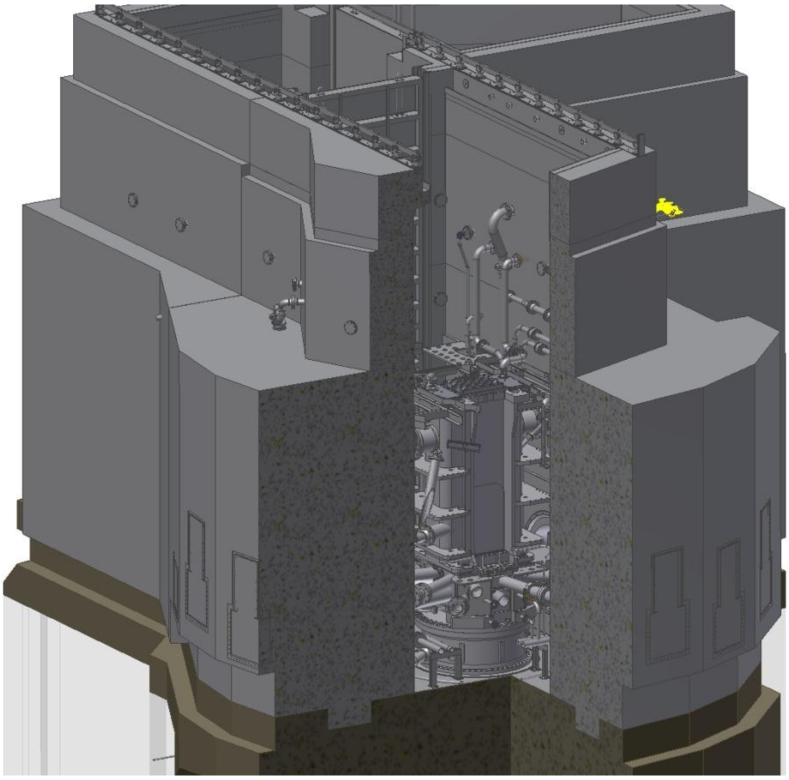

Figure 1-5: Artist impression of reactor vessel in the reactor pool and adjacent pools

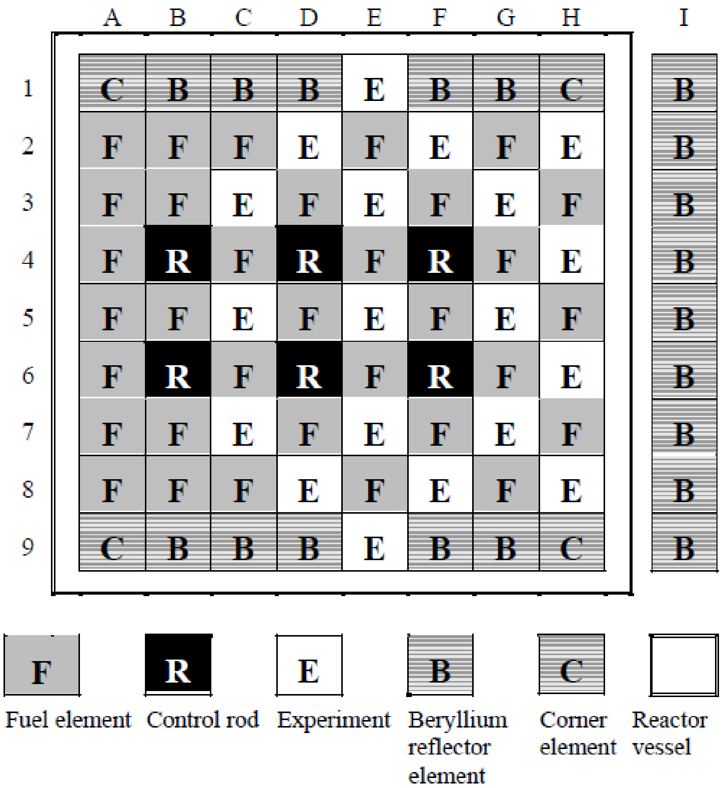

Figure 1-6: Characteristic HFR core configuration

The reactor's nominal operating power is currently 45 MW_{th} and its licensed power is 50 MW_{th}. The heat generated in the reactor vessel is removed through the primary cooling water system (PKWS). This system transfers its heat to the secondary cooling water system (SKWS) through heat exchangers. In the closed primary cooling water system, demineralised water is used as the heat transport medium and in the secondary cooling water system water from the Noordhollands Kanaal is used. This water is discharged to the North Sea. Pumps are used for the circulation of the primary cooling water and the transport of the secondary cooling water. The decay tank in the primary cooling water system serves to allow the activity of short lived radioactive products to decay. It also serves as a de-gasser of the primary cooling system. The reactor vessel is placed in a large water pool, shielding off the radiation from the reactor core and cooling the vessel outer wall. The pool water is cooled by the pool cooling water system (BKWS). This system transfers its heat to the secondary cooling water system.

The control room and its supporting systems are situated centrally within a security zone in the HFR buildings. Access is only possible via a guarded entrance. The control room area has an independent heating and ventilation system and is equipped with ergonomically designed lighting and adequate emergency lighting. The control room is also equipped with devices for operating the communication systems, such as the PA system, evacuation announcement, telecommunications, etc. The instrumentation and monitoring systems are housed in a centrally located operating console with the instrument panel immediately above it. In the event that the control room is inaccessible (for instance, as a result of a fire or an excessive radiation level), there will be an emergency monitoring panel operational mid-2012 in a bunkered building. The important process parameters are displayed on this panel to enable the plant to be brought manually into, and maintained in, a safe state. The panel is located in an adjacent building which is physically separate from the control room.

The concept of the HFR is based on the common conservative designs, the simplicity of the operating systems and the use, where possible, of natural physics to control processes and to achieve safe operating conditions in the event of a systems failure. On this basis, (active) systems which guarantee the safety functions (shutdown, cooling and containment) have been implemented. The adherence to broad safety margins in process safety is part of conservative design. In addition, the design contains a number of inherent safety aspects. One important inherent safety feature of the HFR is the negative temperature coefficient for both fuel and moderator. This is the self-inhibiting nature of the chain reaction by feedback to the fission process of the temperature in both the reactor fuel assembly and the cooling water. A constant increase in power can therefore be excluded on physical grounds. As a result, the consequences of incidents that can result in a significant increase in reactor power are limited. Furthermore after a reactor scram, power reduction or manual shutdown, the start of core poisoning due to the disturbance Xe/Sm-equilibrium will occur. A third aspect is the location of the reactor in the reactor pool and the use of an amply sized primary cooling water system. Both structures possess considerable heat capacity to absorb decay heat.

1.2.1.2 Containment

The containment building consists of a gastight vertical steel cylinder with a diameter of about 25 m and a semi-elliptic roof (Figure 1-7) .

The containment building extends to a height of about 24 m above ground level. The internal volume is approximately 12000 m^3. The containment building has the following functions:

- Housing of the reactor, pools and support systems;
- Containment of part of the primary system;
- Protection against external events;
- Housing of the experimental facilities and equipment;

- Gastight barrier to protect the environment.

The containment building is maintained at a slight under-pressure relative to the surrounding area by the ventilation system so that any radioactive contamination from the containment building present in the air is only released after sampling and monitoring. The so-called “waterslot” of the containment building is designed for an overpressure of 0.5 bar. The containment building is at the same time designed for a negative pressure of 0.007 bar. Leak tests, with a duration of at least 24 hrs., are performed annually at an overpressure of 0.2 bar and once every 4 years at an overpressure of 0.5 bar.

1.2.1.3 Emergency power supply system

The on-site electrical power supply system (EPPS) provides electrical power on a voltage level of 0.4 kV to its users in case of a loss of external electrical 10 kV power from the public grid. It is qualified for the earthquake response spectrum of the HFR. The system consists of three 0.4 kV-50 Hz diesel generator sets A, B and C, each with a nominal design power of 450 kVA with a continuous overload capability of 10%. However, during operation the real connected load is rather low; it varies from 45-68 %.

The diesel generators sets are housed in a separate building on the OLP some 50 m from the HFR. Inside the building the diesel generators A and B are located in one room along with their diesel day tanks and starter batteries. For fire protection reasons diesel generator C, including its day tank and starter battery is located in an adjacent room. The same physical separation is applied to the rooms for the matching switchboards and control panels. All three diesel generator sets are fuelled from a common underground main tank located just outside the EPS building.

If the public grid fails, the three diesels are started automatically and diesel generators A and B are connected to their associated loads; diesel generator C then runs without a connected load. Diesel generator A is the emergency power source of preference for the HFR. The HCL (Hot Cell Laboratories) and the DWT (Decontamination and Waste Treatment) of NRG are also supplied by diesel generator A.

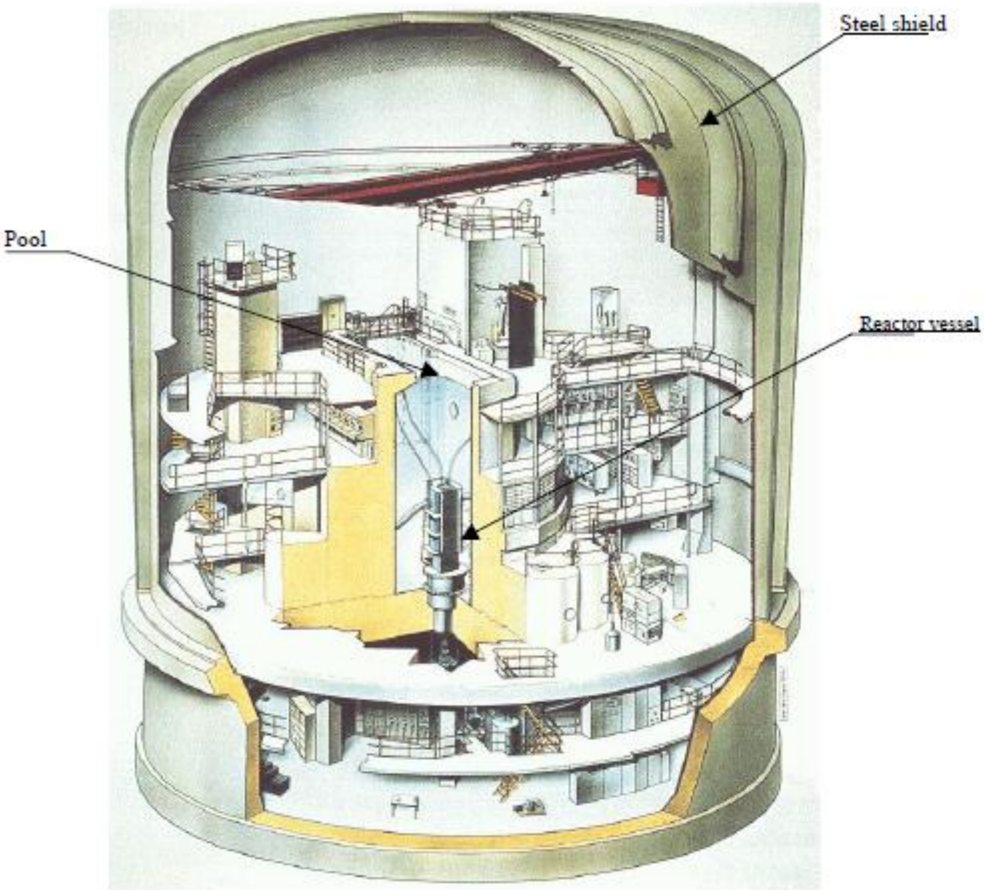

Figure 1-7: Cross section of the containment building.

Diesel generator B supplies the emergency power for the ECN and diesel generator C stands ready in reserve. If diesel generator A fails, its load is then switched automatically to diesel generator C. If diesel generator C also fails, the normal load (ECN) from diesel generator B is switched off after approximately one minute and the components of the HFR are supplied by diesel generator B. If the power from the public grid is restored, the switch back to the public grid is done manually.

1.2.1.4 Annual programme

The annual programme of the HFR comprises 11 to 12 cycles at 45 MW_{th}. A reactor cycle on average lasts 28.5 days. In between cycles, a number of days are available for reactor shutdown. These shutdown periods are used to change the fuel assemblies and experiments and to carry out preventive and corrective maintenance. In addition to the brief reactor shutdown periods, there are two longer shutdowns each year. The longer reactor shutdown periods are used for holidays, training, periodic in-service inspection, modifications and major servicing. Average annual reactor availability is about 80%.

1.2.1.5 Key reactor parameters and safety systems

Table 1-1 and Table 1-2 present the key reactor parameters and main safety system for the High Flux Reactor.

Main (global) data for the HFR reactor at nominal operation	
Description	**Value**
Reactor:	
Maximum power	50 MW_{th}
Pressure (absolute, above the core)	0.34 MPa
Pressure difference over the core	0.11 MPa
Primary Cooling water flow rate	4100 m^3/h
Secondary Cooling water flow rate	1000 - 3125 m^3/h
Pools:	
Depth	8.7 m
water height above reactor vessel	4.2 m
volume of reactor pool	151 m^3
volume of storage pool 1	106 m^3
volume of storage pool 2	84 m^3
Reactor vessel:	
height	5.4 m
core box wall thickness	40 - 50 mm
design pressure (abs.)	0.49 MPa
Reactor core:	
number of positions (within reactor vessel)	72 (8 x 9)
horizontal dimensions (within reactor vessel)	0.73 x 0.62 m
height of fuel	0.60 m
specific power	310 MW/m^3
nominal inlet temperature	40 - 50 °C
nominal outlet temperature	50 - 60 °C
Licensed outlet temperature	66 ^{o}C
heating of cooling water over the core	9 - 10 °C
Coolant flow speed through reactor core	6.8 m/s
LEU fuel element:	
length	0.92 m
number of fuel plates per element	20
fuel matrix material	U_3Si_2-Al
degree of enrichment	19.25 - 19.95 %
^{235}U mass per element	550 g
uranium density in fuel matrix	4.8 g/cm^3
test temperature integrity fuel/cladding binding	425 °C
LEU control rod:	
length of fuel section	0.74 m
length of absorption section	0.79 m
length of extension piece	0.91 m
total length (including coupling for drive mechanism and lifting eye)	3.04 m
fuel	U_3Si_2-Al
degree of enrichment	19.25 - 19.95 %
^{235}U mass per element	440 g
number of fuel plates per element	17
cadmium mass per control rod	1680 g

Table 1-1: Key reactor parameters of the HFR

Safety Systems	Features/ characteristics
Reactivity control systems	6 control rods Cadmium plate (Δ -300pcm)
(Emergency) cooling systems	Primary cooling system Decay heat removal pump system Pool water injection system Convection cooling system Supplementary pool water storage tanks Supplementary demin water system
Containment systems	Reactor building isolation system Off-gas system with absolute/carbon filters Hot cell recirculation system

Table 1-2: Main safety systems of the HFR.

1.2.2 The Low Flux Reactor (LFR) / LFR Hall

The LFR is located in a separate hall in the Enrico Fermi Laboratory which is situated in the western valley at the western side of the OLP.

Figure 1-8: Frontal view of the Fermi building with the LFR hall on the left and the STEK hall on the right.

Besides the LFR with its control panel, this hall contains a calibration installation which e.g. can be used for calibration of dose meters and radiation detectors.

The LFR (Figure 1-9) is a water-cooled 30 kW research reactor of the Argonaut type with a cylindrical reactor vessel of 0.9 m external diameter and 1.2 m height, which contains a cylindrical inner vessel of 0.6 m diameter and 1.2 m height with the internal graphite reflector. Both vessels are made of pure aluminium, the ring-shaped space in between forms the core space. The outer reactor vessel is surrounded by stacked graphite blocks (external reflector), in its turn surrounded by barite blocks for shielding of ionising radiation (Figure 1-10). The LFR was started using a ^{241}Am-Be neutron starting source. Its neutron emission rate is around 1.0 E7 neutrons per second. Based on the activity of the originally 1.5 gram ^{241}Am in the starting source, the activity presently amounts to 1.7 E11 Bq. The LFR has always been operated in a service mode at variable power levels.

In the last years its use declined ever more. The Low Flux Reactor (LFR) ceased operation at the end of 2010. The fuel elements have been removed and stored in the storage pits in the reactor hall. The fuel elements will remain in this storage facility until they can be transported to COVRA.

The pits for the LFR fuel storage (pluggennest) are located in the south-westerly corner of the LFR hall. This pit storage consists of a 6 x 5 rectangular array of vertical pits, 1.70 m long steel containers of 18 cm cross-section, 38 cm apart (heart distance), encased in dense concrete. These containers contain steel fuel storage tubes of 113/127 mm cross-section. The annulus between the container and the storage tube is filled with sand. Each storage tube is closed by an 85 cm long steel plug. The geometry of this pit storage is such that re-criticality of stored LFR fuel elements is excluded in all operational and accidental situations, including inundation, and has ample leeway for geometrical changes due to structural deformations.

An experimental conversion plate (BIBOP) has been used to enable irradiation with 1 MeV fast neutrons in the LFR. This plate contains 18 LFR fuel plates. Since this conversion plate has been irradiated on dedicated occasions only, the fission product inventory of this plate is negligible compared to those in fuel elements. The conversion plate is currently stored at a shielded position at the front side of the irradiation trolley of the LFR.

For specific research concerning the irradiation of tumours at the LFR a facility was built for irradiation of tissue with thermic neutrons. This facility consists amongst others of 1087 kg graphite and a 273 kg bismuth filter. Activation of bismuth (^{209}Bi) produces mainly ^{210}Po. The current ^{210}Po inventory is estimated at $3.5 \cdot 10^7$ Bq. This filter is stored at the back side of the reactor.

In the LFR hall, a number of sealed sources are present that are being used for training and other activities such as calibration of dose meters and radiation detectors (Table 1-3).

The LFR hall is a steel construction built of 4 steel frames supporting the 8.8 m high roof spanning the 14 m wide hall. The walls of the 20 m long hall are self-supporting masonry walls with glass windows and door openings. The floor of reinforced concrete is covered by an abrasion resistant, well decontaminable top layer.

Figure 1-9: Photograph of the Low Flux Reactor (LFR)

Three semi-gastight, fire retardant doors form the entrance to the LFR hall from the Fermi building. Next to these connecting doors the hall is directly accessible from outside by a fourth large, 4 m high door. This is a semi-gastight double steel door, provided with a personnel flight door. All doors are provided with special seals by which under-pressure in the LFR hall can be maintained.

The LFR hall is ventilated with filtered and pre-heated fresh air. The capacity of the separate supply and exhaust ventilators is chosen such that under-pressure in the hall is guaranteed if all doors are closed. Ventilation air is exhausted over a filter unit. The ventilation exhaust air is continuously monitored.

During operation of the reactor mainly ^{41}Ar was measured, a short-lived noble gas produced by neutron activation of ^{40}Ar, a normal constituent of air.

Radiological conditions in the LFR hall are monitored by the radiation monitoring system with 3 detectors in the vicinity of the LFR. According to procedure, this monitoring system is only active during operation of the LFR and is now switched-off, consequently.

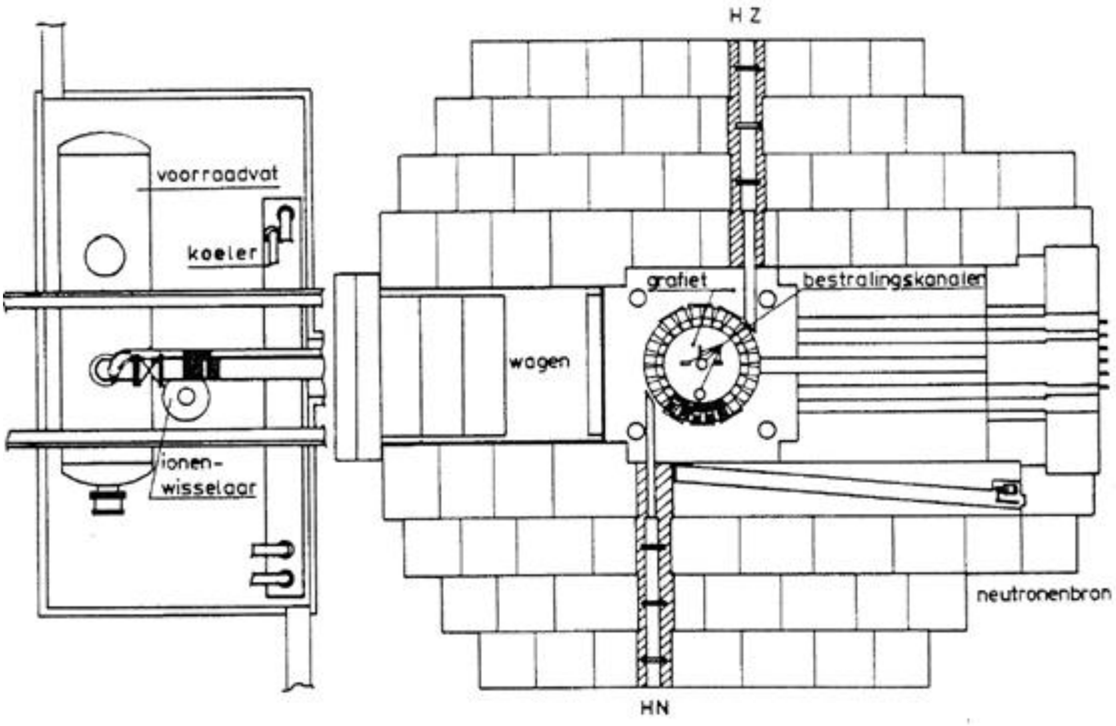

Figure 1-10: Horizontal cross section of the LFR

Location	Nuclide	Activity [Bq]
Calibration setup position 1	^{60}Co	2.0 E8
Calibration setup position 2	^{60}Co	8.0 E8
Calibration setup position 3	^{60}Co	5.3 E10
Calibration setup position 6	^{137}Cs	4.2 E9
BIBOP desk	$^{90}Sr/^{90}Y$	2.9 E8

Table 1-3: Inventory of sealed sources present in the LFR hall

1.2.3 Hot Cell Laboratories (HCL)

The Hot Cell Laboratories have been built at the beginning of the sixties and are designed for handling and testing of highly radioactive material. Over the last forty years extensive research programs have been carried out in the field of fuel and structural materials for application in different types of nuclear reactors. In 1995 the research laboratory (HCL-RL) has been expanded with a Molybdenum Production Facility (HCL-MPF) for the production of molybdenum-99. This isotope is being used for the production of technetium-99m, which is the most widely used isotope for medical diagnostics.

1.2.4 The Research Laboratory (HCL-RL)

The HCL-RL is used for carrying out materials research, for processing radio-isotopes for both medical and industrial purposes and for the processing of radioactive waste (Figure 1-11).

The building holds a central transport hall (area 42 and 43 in Figure 1-12), which is surrounded by concrete hot cells (cell A through E in Figure 1-12) at the south and lead hot cells (F-cells and G-cells) at the east and north side. The line of concrete cells, called High Activity cells, provide sufficient shielding to contain an activity of up to 3.7 PBq per cell. In the lead cells minor quantities of activity can be present. These cells contain equipment to measure mechanical properties and to perform metallographic procedures. A separate section of the building holds the actinides laboratory in which chemical research is done on non-irradiated samples of actinides in glove boxes.

1.2.4.1 Ventilation system

The ventilation system maintains a constant under-pressure in all the cells. An intake grille on the eastern side of the HCL-RL draws in 30,000 m^3 of outside air per hour and, after filtering and heating, distributes it throughout the peripheral areas through a system of ducts. The air then passes through the U-shaped corridor on the ground floor and from there is channelled to the lead cells hall (area 39 in Figure 1-12) and the transport hall (area 42 in Figure 1-12).

The High Activity (HA) cells take in approximately 5500 m^3 of air per hour, from the transport hall. This includes the air extracted from the area above the storage pool.

The flow rate in cells F1-F8 and G1-G8, as well as the other cells in the HCL- RL, collectively is about 500 m^3 of air per hour.

Figure 1-11: Impression the HCL-RL

The remainder – approximately 24,000 m^3 per hour – is extracted from the eastern (area 47 in figure 2) and western (area 48 in Figure 1-12) loading docks and the decontamination area by the building ventilation units and passes into the building air extraction duct. The cells are ventilated via grouped connections to extraction ducts in which an under-pressure is maintained. Several distinct subsystems perform this function:

- The HA extraction duct, situated in the basement underneath the HA cell line;
- The MA extraction duct, situated along the southern wall of the lead cells hall;
- The off-gas duct, with two branches, one running along the front side of the HA cells and the other along the front side of the MA cells.

Downstream, the air from the MA extraction duct feeds into the HA extraction duct. This combined flow then passes through a box of absolute filters. From there it feeds into the parallel cell air extraction units. The off-gas flow also passes through an absolute filter before entering the group of off-gas ventilator units.

The pressurised sides of the building air ventilators and the cell air ventilators are combined and fed through an isolating valve into the so-called “header”. This marks the starting point of the duct leading to the ventilation stack, the mouth of which is 45 metres above sea level.

The intake and extraction ducts from the actinides laboratory and the general chemistry laboratory are fitted with fire valves, and the extraction system also has an absolute filter.

The air extracted from the glove boxes in these laboratories feeds into a separate removal system. Each extraction duct is fitted with two absolute filters and a fire valve, all located outside the laboratory.

The intake filters ensure that only clean air enters the ventilation system. The extraction filters are there to prevent the release of hazardous or radioactive substance. The purpose of the isolating valves is to seal off the duct in the event of a ventilation unit failure, to stop any reverse flow of air.

1.2.4.2 Emergency power supply systems

The normal electrical power for HCL-RL is delivered from substation ECN 6 (see section 1.3.5.3.1). HCL-RL is connected to emergency power supply. In case of loss of off-site power, the following systems are supplied with electrical energy by emergency bus-bar A: ventilation systems for building and off-gas systems for hot cells, air compressor station, tele-manipulators wall sockets in cells, warning and monitoring systems, fire warning system, emergency lighting system and the NDO computer system.

1.2.4.3 Storage facilities

Radioactive materials are stored in the storage pool (58 m^3) and in 24 storage tubes, both located in the transport hall below ground level. The water-filled pool extends underneath cell AB in the High Activity (HA) line. Its floor and walls are made of concrete lined with watertight stainless steel. The pool (580 x 200 x 500 ; l x w x d) is used for the buffer storage of fuel rods and uranium containing waste. To hold these, the pool is fitted with racks configured in such a way using geometry and neutron absorbing materials, that criticality can be excluded. It is also possible to transfer materials directly between the pool and cell AB. Heat dissipation occurs by natural convection. The pool is fitted with a water purification system to maintain the optical quality of the water and to keep its level of radioactive contamination below 18 MBq/m^3 for beta/gamma emitters.

The storage-tube facility enables the dry temporary storage of radioactive objects and waste. It contains 24 steel tubes, galvanised inside and out, with a diameter of approximately 300 mm. These have a total capacity of 48 standard canisters. A tube is accessed from the transport hall by removing the 120 mm-long barite concrete plug with a sealing and shielding function. Canisters can then be transferred between the storage facility and HA cell C using a shielded transport container with vertical loading and unloading function. Air extracted from the facility passes through an absolute filter situated in the basement underneath the cells and into the cell air extraction duct.

1.2.5 The molybdenum production facility (HCL-MPF)

The HCL-MPF holds two independent lines of five lead hot cells each that are dedicated to the production of molybdenum (see Figure 1-13). In between the two lines a maintenance and transport hall is located. In each line the molybdenum is extracted from irradiated uranium containing targets in five chemical process steps. The pure molybdenum is transported in bulk to different plants where the product is loaded in technetium generators. These generators are shipped to hospitals where the technetium is used in medical, mainly diagnostic, applications.

1.2.5.1 Hot Cells

The hot cells of the HCL-MPF are all stainless steel lined at the inside. The first two cells have more shielding capacity than the remaining three because of the amount of radioactivity that they hold. The process equipment in the cells is controlled from the front of the cells using control panels and master-slave tele-manipulators.

1.2.5.2 Ventilation system

HCL-MPF is equipped with its own ventilation system. The ventilation system maintains two separate airflows, i.e. the building airflow and the cells- and glove box airflow. The system maintains a lower absolute pressure inside the cells in order to guarantee the outside-in direction of the airflow. Because of its safety function the active components are installed redundantly and the ventilators are connected to the emergency power supply. The off gas of the cells are equipped with delay bed filters for noble gasses, activated carbon and absolute filters. Because of its key safety function, the active components in the ventilation system are safeguarded against simple failure by redundancy duplication of the important ventilator units and filters and by supplying the ventilator units with emergency power in the event of a mains outage. Control systems ensure that the required conditions are maintained and prevent undesirable situations.

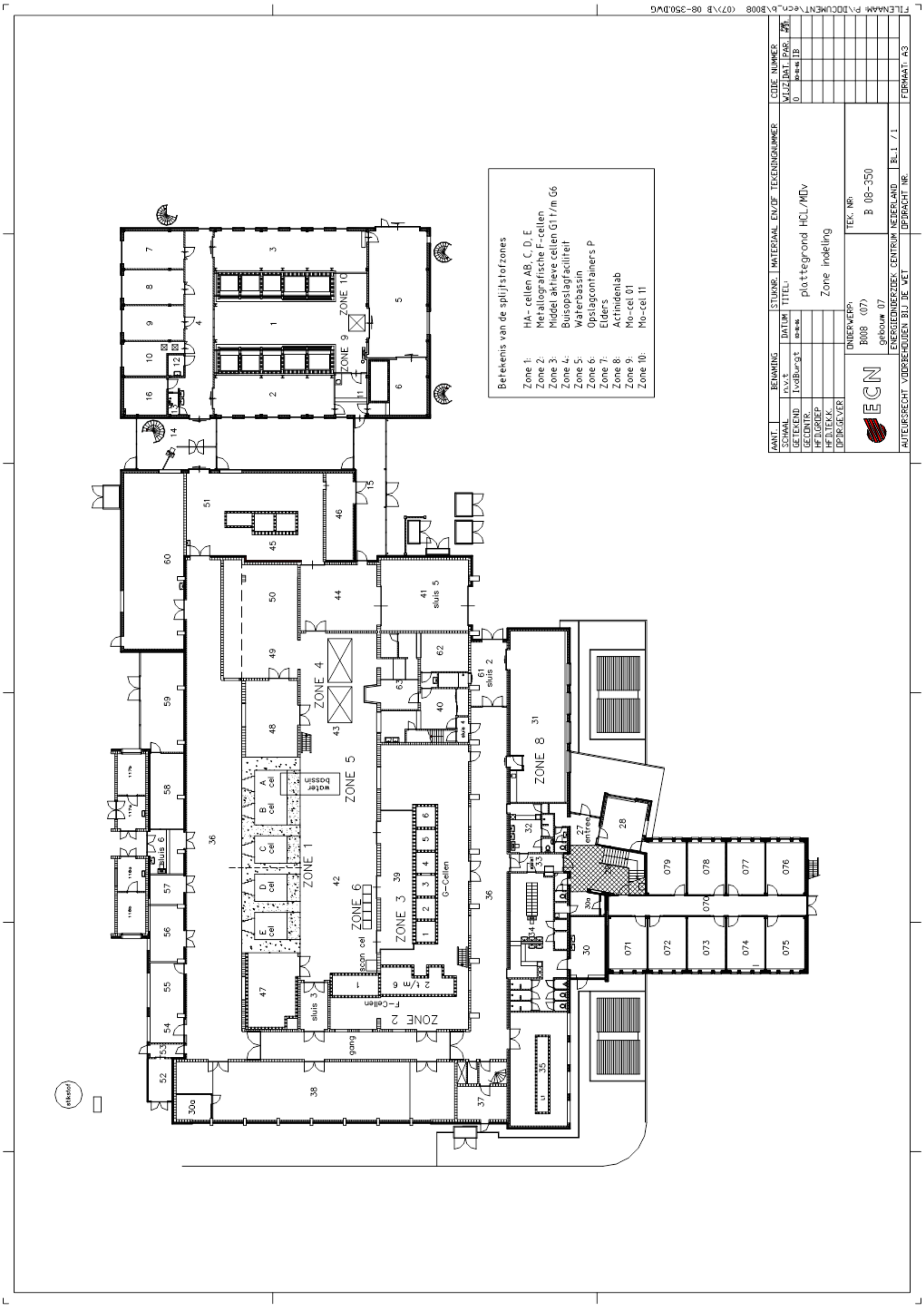

Figure 1-12: Ground floor plan of the Research Laboratory (HCL-RL)

1.2.5.3 Emergency power supply system

The normal electrical power for HCL-MPF is delivered from substation ECN 3 (see section 1.3.5.3.1). HCL-MPF is connected to emergency power supply. In case of loss of off-site power, the following systems are supplied with electrical energy by emergency bus-bar B: ventilation systems for building and off-gas systems for hot cells, air compressor station, tele-manipulators, pumps wall sockets in cells, warning and monitoring systems, fire warning system, emergency lighting system. Control panels of the hot cells, warning panels and radiation monitoring systems are equipped with no-break units.

1.2.5.4 Waste streams

The different types of waste produced are temporarily stored in the HCL- MPF before it is either transported to the HCL-RL for intermediate storage or is directly transported to the national radioactive waste storage facility (COVRA).

1.2.5.5 Uranium containing waste

The uranium from the targets remains in the first cell of the production line. The material from a number of batches is collected in a collect filter (UCW filter). A maximum of four collect filters may be present in every production line. The inventory of fissile material in the cell is controlled by technical measures and procedures to prevent criticality. The allowed maximum uranium mass is 700 g ^{235}U-equivalent in each production line. The loaded collect filters are transported to the HCL-RL for further intermediate storage. The transport of collect filters to COVRA is carried out once every two years.

1.2.5.6 Liquid waste

The high active liquid waste is collected in double walled storage tanks that are located in the basement (capacity 1200 or 350 litres). Low active liquid waste is stored underneath a number of cell chambers, inside their lead shielding (capacity 245 litres) of the HCL-MPF. The characteristics of the waste differ from cell to cell. After sufficient decay the liquid waste is transferred into transport containers and transported to COVRA.

1.2.5.7 Solid waste

Solid waste is generated during the production process. Examples are disposables like filters, separation columns, beakers, tubes, tissues and the like. The waste is collected in a waste drum inside the cell. A filled drum is temporarily stored inside the cell. After sufficient decay of short-lived radionuclides the drum is transported to the HCL-RL and Waste Storage Facility (WSF) for intermediate storage. Solid waste that is generated due to maintenance or replacement is disposed of through the same route.

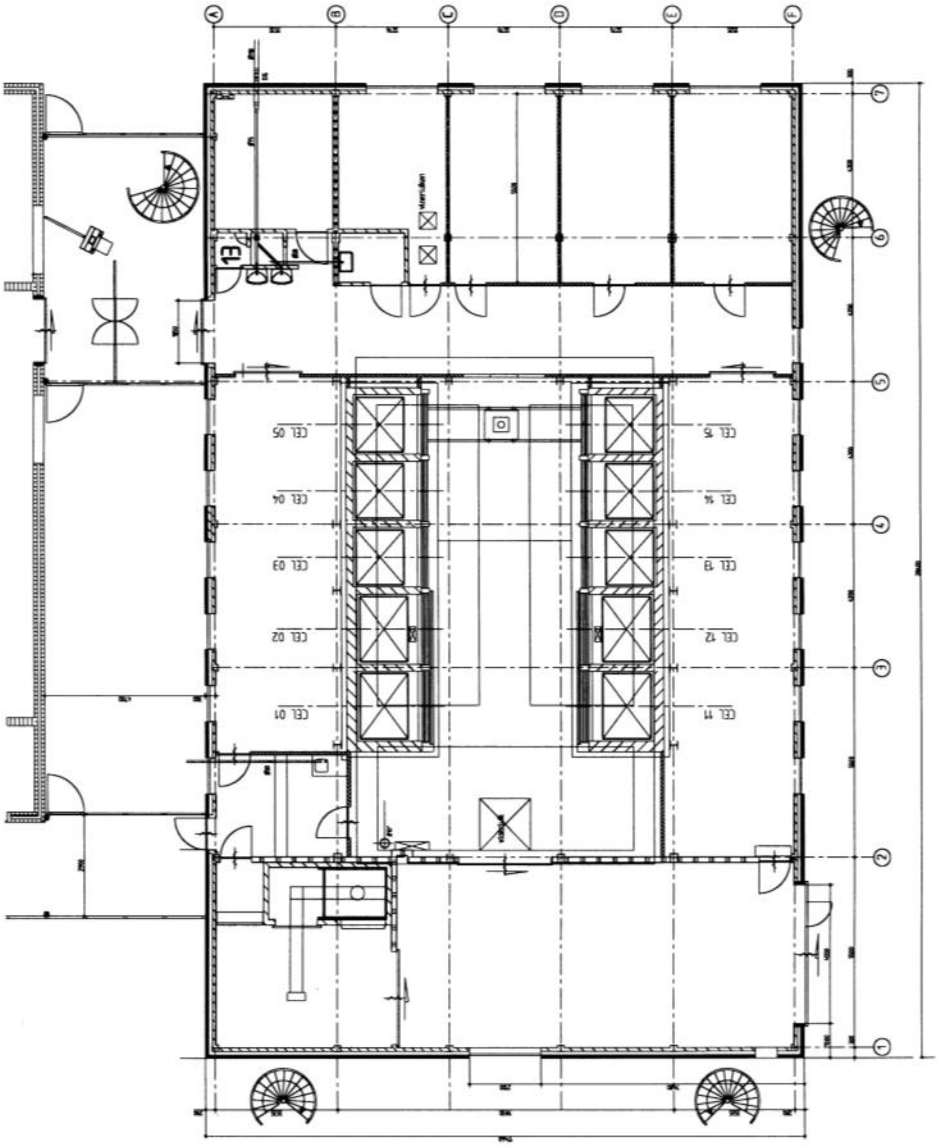

Figure 1-13: Ground floor plan of the molybdenum production facility (HCL-MPF)

1.2.5.8 Gaseous waste

Underneath a number of cell chambers, tanks with a capacity of 350 litres are fitted inside their lead shielding. These tanks are for temporary storage of radioactive gases, in order to allow the very short-lived isotopes to decay before the gas is piped to the delay beds. Each production line has a system of delay beds for the controlled capture, decay and discharge of noble gases released during the molybdenum production process. Such a bed consists of a series of interconnected carbon filters, designed to delay the passage of the gases for at least 70 days so that they decay to a radiation level within permitted discharge limits. This period incorporates a substantial safety margin.

The individual filters making up the system can be isolated using valves. The delay beds are located on the first floor and are shielded on all sides by 50 mm of lead. Underneath them, a steel base plate and the 30-cm-thick concrete floor prevent radiation from penetrating the working areas below.

1.2.6 Decontamination and Waste Treatment facilty (DWT)

Figure 1-14: Overview of the Decontamination and Waste Treatment facility

The Decontamination and Waste Treatment facility (DWT) is located in the western valley of the Onderzoekslocatie Petten (OLP). The DWT facility comprises three buildings and an area for temporary storage of contaminated equipment:

- Decontamination building (at the left hand side of the picture) is mainly used for (wet) decontamination of equipment from the nuclear and NORM industries. An enclosed area is used for decontamination of highly contaminated objects.
- Water treatment building (at the right hand side of the picture) is mainly used for storage, treatment and controlled release of waste water (via a storage basin and a 4.4 km long double contained waste water pipe) into the North Sea.

- Solid-waste treatment building (at the back-end at the picture) is mainly used for treatment such as compacting and packaging of solid waste and cleaning of contaminated steel tubes. A tube-cleaning system is installed in an enclosed area for cleaning oil & gas tubing. The building has shielded subterranean vaults with concrete lids for storage of active waste.
- Storage area with water-tight sloping pavement ("Stelcon plates") with controlled accumulation of (contaminated) drain and rain water that will be carried-off to a silt separator and oil-water separator and accumulated in a well. From here the contaminated water will either be transferred to the OLP sewage system or (in case of a level of contamination exceeding the nuclear permit) to be transferred to the water treatment system.

1.2.6.1 Ventilation system

All buildings are ventilated in order to establish an under-pressure regime inside these buildings. The ventilation exhaust is filtered and monitored. Some of the systems in the buildings have their own filters (for batch wise monitoring). The filters of the ventilation system of the solid-waste treatment building are HEPA filters and an activated carbon filter. Too small under-pressure triggers an alarm signal by the building management system, initiating measures by the building manager.

1.2.6.2 Emergency power supply system

The normal electrical power for DWT is delivered from substation ECN 4 (see section 1.3.5.3.1). DWT is connected to emergency power supply through emergency bus-bar A. In case of loss of off-site power, the following systems are supplied with electrical energy by emergency power supply: ventilation systems for building and off-gas systems , air compressor station, leakage monitoring systems, warning and radiation monitoring systems, fire warning system, and emergency lighting.

1.2.6.3 Radiation detection system

Radiation levels inside the DWT are monitored constantly by radiation sensors. To monitor possible airborne radioactivity within DWT, various continuous air dust contamination monitoring units are installed. This units consists of an air pump, a filter and a GM tube for continuous β and γ measurements. Possible α contamination is determined periodically by analysis of the filters.

1.2.6.4 Fire detection sytem

Ionisation fire detectors, connected to the on-site emergency power grid, are placed at various locations at DWT. These fire detectors are connected to the central alarm station (CAS) and the corporate fire brigade

1.2.6.5 Water detection system

Underground piping for waste water transport is either double-containment piping or placed in a tunnel with leakage detection.

1.2.7 The Waste Storage Facility (WSF)

In the Waste storage Facility (WSF) radioactive waste is stored in its basements (Figure 1-15;Figure 1-16). These are either 2.5 or 5 metres in depth, with the deeper ones fitted with storage tubes. Some of these are sealed using a steel or concrete plug; they are known as "plugged stores". The others are covered by steel bars and are called "tube stores". The shallower basements take the form of trenches, covered with slabs of concrete. The plugs, steel bars and concrete slabs form the floor of the WSF.

The WSF contains the following storage systems:

- Trenches, against the northern and southern walls, subdivided in smaller compartments;
- Plugged stores, consisting of both wide (A) and narrow tubes (B);
- Tube stores, divided in north and south cellar.

The basement walls and floors are watertight. The groundwater level at the WSF site is sufficiently low that no leakage into the basement storage areas can occur even if that sealing is lost. The ground surfaces around the facility are also watertight, so that even in heavy precipitation no rainwater can penetrate the basements. Any water that does enter them despite all these precautions is detected by the building management system and sets off an alarm.

1.2.7.1 Trenches

The trenches are used for the temporary storage of stable intermediate and high-level waste (ILW/HLW) and of non-irradiated nuclear fuel. As the amount of stored fissile material can exceed the safe mass dedicated storage systems are installed to prevent criticality of the material. The duration of storage here depends upon the composition of the material concerned, and in particular the half-lives of the composing radioactive materials. Sufficiently decayed waste is removed from the trenches and prepared for shipment to COVRA.

The trenches consist of two large long cellars (Figure 1-15). The floor and covering slabs are made of concrete, which is sufficiently thick to work in the WSF hall without further measures to shield direct radiation from the trenches.

1.2.7.2 Plugged stores

The plugged stores are used for the temporary storage of stable Intermediate Level Waste (ILW) and High Level Waste (HLW). Each tube here is covered by its own individual lid, the so-called “plug”. The waste itself, which includes radiated nuclear fuel, is packed in steel canisters. In most cases, the dose rate on the outer surface of a canister necessitates that it has to be transported in a shielded container. Using the hoisting system installed inside the storage hall, this container is suspended above the required tube so that the canister can be lowered directly into it, or lifted from the tube directly into the container.

The plug storage area contains two types of pipes:

- Eternit pipes (areas B and C), 99 pieces, with a diameter of 0.30 m (12 inches), covered with steel plugs. Each plug has a number and is provided with a hole to attach a hoisting bolt;
- Steel pipes (area A), 27 pieces, with a diameter of 0.23 m (8 inches), for storage of specific wastes, such as control rods. These pipes are covered with a plug of very thick, extra dense concrete.

The intermediate space between these pipes is filled with concrete. Their lengths amount to 3.85 m. Only in some assigned pipes a restricted amount of nuclear fuel may be stored. Additional facilities ensure restriction of the number of pipes for fuel storage to 33 out of 99 available pipes.

The floor, outer wall and covering slabs with plugs are made of thick and dense concrete, thus enabling work in the WSF hall without further measures to shield direct radiation from the plug storage area.

1.2.7.3 Tube stores

The tube stores are also used for the temporary storage of stable ILW and HLW. Underneath the central section of the WSF are two basement areas in the form of large basins with thick high-density concrete walls and floors. These house the tube stores. The tubes here are covered by steel bars. Again, this structure is so designed that no special protective measures against radiation from it are required when working elsewhere in the WSF.

The pipe storage area is divided into a north and south cellar. The south cellar contains 301, the north cellar 387 eternit pipes, each with a diameter of 0.30 m (12 inches), and is enclosed by 0.56 m (22 inches) thick cast steel beams. The pipes measure to 4.32 m each.

Pipes in the pipe storage area are filled by means of a loading trolley. This trolley fulfils two additional functions:

- Transfer of covering steel beams;
- Maintaining shielding of the filling zone. To that end the trolley is filled with dense concrete.

The movements of the loading trolley are physically restricted so that the wagon cannot be moved from uncovered pipes.

1.2.7.4 Ventilation system

The ventilation system is meant to keep the concentrations of airborne radioactive material as low as feasible. The ventilation system creates an under-pressure in the cellars below floor level. Too small under-pressure triggers an alarm signal by the building management system, initiating measures by the building manager. Air from the WSF hall is ventilated through small openings all around the storage areas into the cellars in the basement. From these cellars the ventilation air is conducted through a 500 x 400 mm conduct to a filter unit on top of the sorting cell in the neighbouring Waste Treatment building of the Decontamination & Waste Treatment (DWT) facility. That filter unit serves only the WSF ventilation installation and captures any airborne radioactive contamination in the ventilation air. The WSF ventilation exhaust air is sampled for α, β and tritium activity. To that end part of the exhaust air is led over an activated carbon filter and a condensation unit and measured periodically.

1.2.7.5 Emergency power supply system

The normal electrical power for the WSF is delivered from substation ECN 4 (see section 1.3.5.3.1). The WSF is connected to emergency power supply. In case of loss of off-site power, the following systems are supplied with electrical energy by emergency bus-bar A: ventilation systems for building and off-gas systems , warning and monitoring systems, fire warning system.

1.2.7.6 Radiation detection system

Radiation levels inside the WSF are monitored constantly by five sensors. These have a measurement range of 0-100 μSv/h and are fitted with lamp and sound alarms, indicating whether the value currently being detected is greater or less than 10 μSv/h. Both units are connected to the on-site emergency power supply.

1.2.7.7 Detection of air dust contamination

To monitor possible airborne radioactivity in the WSF, a continuous air dust contamination monitoring unit is installed. This unit consists of an air pump, a filter and a GM tube for continuous β and γ measurements. Possible α contamination is determined periodically by analysis of the filters.

1.2.7.8 Fire detection system

Ionisation fire detectors, connected to the on-site emergency power grid, are placed at four locations in the WSF. These fire detectors are connected to the central alarm station (CAS; "Centrale meldpost") and the corporate fire brigade via a substation in the entrance of the DWT solid waste halls of the Decontamination &Waste Treatment (DWT) facility.

1.2.7.9 Water detection system

The cellar floor contains gutters in which any water is collected in case of an unforeseen flooding due to leakages. The gutters lead to a collection pit equipped with a water detection system (water detector). This system is connected to the building management system. Furthermore, an installation is present to empty the water collection pit.

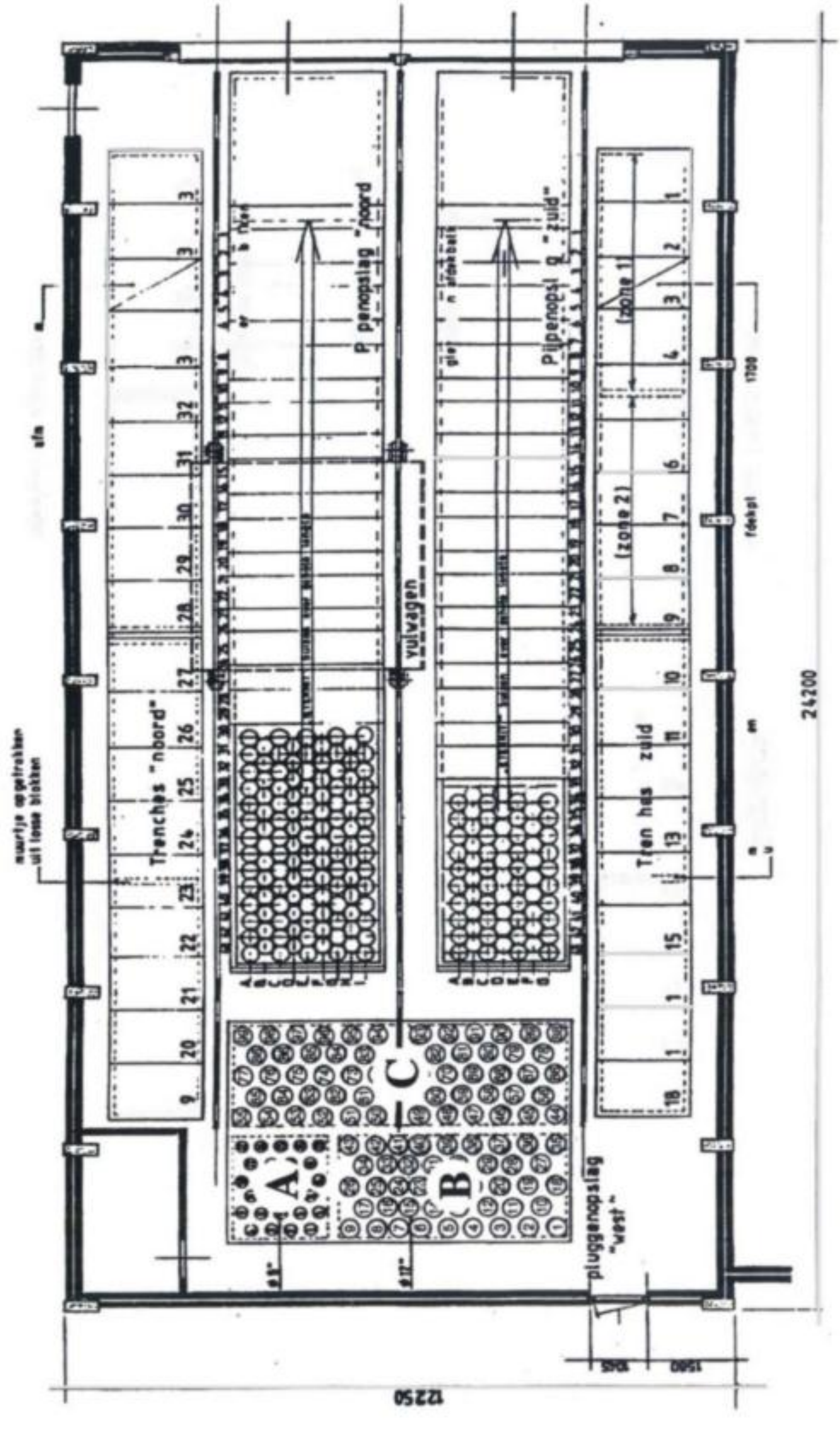

Figure 1-15: Floor plan of the Waste Storage Facility (WSF)

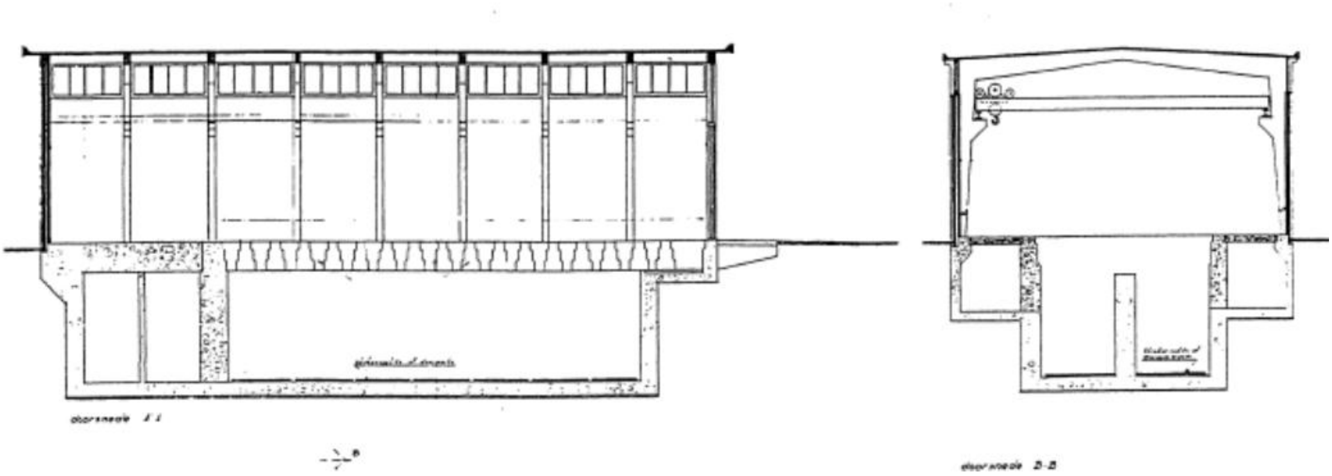

Figure 1-16: Cross sections through the Waste Storage Facility (WSF).

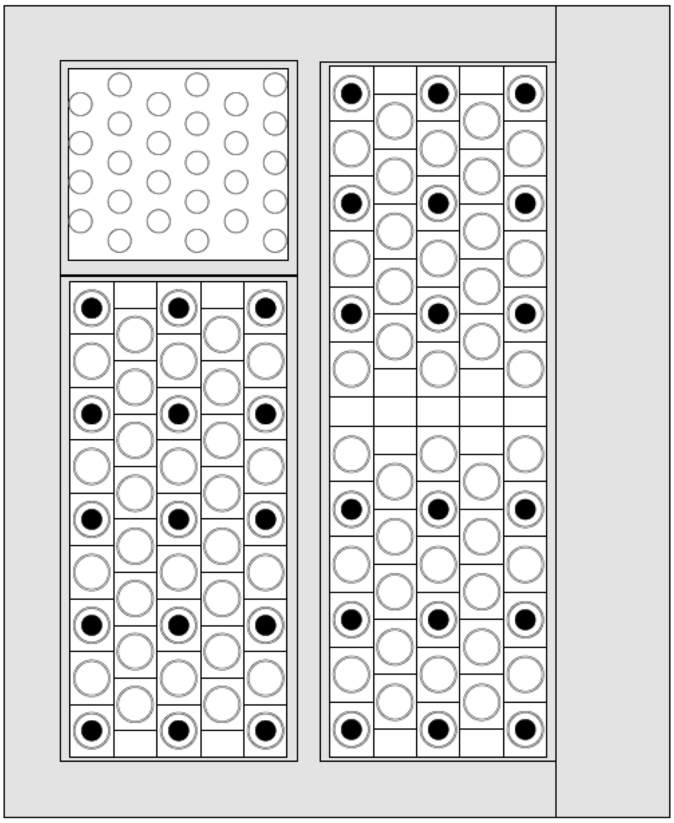

Figure 1-17: Lay-out of plug storage that are allowed to contain fissile material (indicated in black)

1.2.8 The STEK-hall

The Fermi Laboratory is situated in the western valley at the western side of the OLP (Figure 1-8). The intermediate building contains offices, laboratories, measuring rooms, a workshop and service rooms. This building is connected to the LFR hall by semi-gastight fire retardant doors at one side, and with gas-tight doors to the STEK hall, once designed to house experiments with critical reactor configurations.

Until the late 1970's the STEK-hall was used to house an experimental zero-power reactor. Between 1970 and 1990 a neutron generator was installed there. Both installations have been removed completely.

The construction of the hall makes it suitable for storage of packages containing radioactive materials. The STEK hall is a reinforced concrete construction, measures 7.50 x 11.00 m and is 13 m high. Up till 7.80 m height the outer walls measure 1 m, the wall facing the Fermi Laboratory even 1.5 m. Above 7.80 m height wall thickness drops to 50 cm. The roof of the STEK hall consists of a single 25 cm thick reinforced concrete slab, free lying on top of some 15 cm deep recesses in the building walls.

Presently, the STEK hall is mainly used to collect transport ready radioactive wastes from the Hot Cell Laboratories (HCL), both from the Molybdenum Production Facility (HCL-MPF) and the Research Laboratory (HCL-RL) awaiting transport to COVRA. In addition the STEK hall is used storage facility of short-lived intermediate level solid radioactive waste – providing time for decay – as well as for storage of neutron sources. The STEK hall is not used for storage of fissile materials.

The hall is equipped with a ventilation system containing absolute filters.

A semi-gastight door forms the entrance to the STEK hall from the Fermi building. Next to this connecting door the hall is directly accessible from the outside (east) by a large square 3 m high double steel door. For shielding purposes this door can be blocked from the outside by a 70 cm thick, 4 m wide and 3.5 m high dense concrete sliding door, now permanently moved aside.

The STEK hall contains the following installations:

- Hoisting installation;
- Ventilation system;
- Detection of air dust contamination;
- Fire detection and extinguish equipment.

1.2.8.1 Hoisting equipment

The STEK hall is furnished with an electric controllable bridge-crane at 10 m height with movable trolley. This crane moves on two steel rails. Its maximum allowable load amounts to 5 kN.

1.2.8.2 Ventilation system

The STEK hall is equipped with exhaust ventilation with absolute and activated carbon filters. In addition, exhaust air is monitored for radioactivity.

1.2.8.3 Detection of air dust contamination

To monitor possible airborne radioactivity in the STEK hall, a continuous air dust contamination monitoring unit installed. This unit consists of an air pump and a filter. Possible α contamination is determined periodically by analysis of the filters.

1.2.9 The Jaap Goedkoop Laboratory (JGL)

The JGL (Figure 1-18) is a type B radionuclide laboratory in operation since 2007. This means that the design and use of the JGL meets type B standards for radionuclide laboratories related to ventilation, workbenches, laminar air flow cabinets, storage of materials, etc.

The quantities of radionuclides and of fissile materials present in the laboratory are limited by applicable regulations for type B radionuclide laboratories. A typical value that is allowed in a glove box in the JGL (Figure 1-19) at maximum is 2,000 RE.

In the Jaap Goedkoop Laboratory (JGL) research is performed to various materials, e.g.:

- materials used for nuclear fission research;
- materials used for nuclear fusion research;
- materials used for the research to the shortening of the lifetime of nuclear waste.

Besides research also production and handling of medical radioisotopes is performed in the JGL.

1.2.9.1 Ventilation system

The ventilation system is accommodated with less (different) filters compared to the HCL ventilation systems. The building exhaust air is carried through absolute filters. When the building exhaust air ventilation stops, the supply fan will stop automatically in order to prevent overpressure. The ventilation system is connected to the emergency power supply. The valves of the ventilation system can be closed manually but this requires that staff is present on the second floor.

Figure 1-18: Photograph of the Jaap Goedkoop Laboratory (JGL)

The ventilation system of the JGL consists of:

- One air inlet system for the cellar and the ground floor;
- One air inlet system for the first floor and the second floor;
- A redundant local off-gas systems;
- A redundant off-gas system for the complete building.

This consists of two identical ventilation systems. Only one of these systems is in operation at a time. Each week one of the systems is stopped and the other is started. This in order to ascertain that both of these systems can be operational and that the wear of both systems is equal. The outlet air goes through an absolute filter and a measurement system that determines the activity concentration in the outlet air. In case concentration of radioactivity in the outlet air exceeds a predefined threshold level, an alarm signal is triggered.

1.2.9.2 Emergency power supply system

The normal electrical power for JGL is delivered from substation ECN 7 (see section 1.3.5.3.1). JGL is connected to emergency power supply. In case of loss of off-site power, the following systems are supplied with electrical energy by emergency bus-bar B: ventilation systems for building and off-gas systems for benches, flow hoods and glove-boxes, wall sockets in laboratories for active experiments, warning and monitoring systems and central intercom, fire warning system, emergency lighting system. The fire detection system is equipped with back-up batteries that have a stand-by time of 72 hours.

Figure 1-19: View on gloves boxes in the JGL

1.2.9.2.1

1.2.9.3 Waste storage

The JGL is equipped with two storage tanks located in the cellar of the building for potentially radioactive water coming from the laboratories. During normal operation of the JGL the radioactivity level inside these tanks is very low.

1.2.9.4 Control systems

At the entrance of the JGL building (outside the controlled entrance system) there are several panels located for monitoring and control of the JGL. These are the following systems:

- Radiological monitor system determining dose rates at various locations in the JGL. This monitor system also shows the activity concentrations in the air at various locations, including exhaust, in the ventilation system;

- A building control system “gebouw beheerssysteem”, capable of opening and closing valves (e.g. of the ventilation system) remotely from another location (currently mainly ECN office space);
- The fire alarm and valves panel. This panel shows the location of the monitors that detect fire. These valves can be switched in order to close the inlet and outlet valves.

1.2.10 Support facilities

In the context of the CSA the following support facilities have been identified that play a role in accident management:

- Telecommunication Centre;
- Emergency coordination centre, “INO-rooms” (2 locations on the OLP, Forum Grote vergaderzaal and room 111 Building 309, Joint Research Centre);
- The GBD building with extensive supply of radiological measurement systems;
- Fire department;
- Drinking water pump facility.

These facilities will be addressed in the (severe) accident management section (chapter 6).

1.3 Systems for providing or supporting main safety functions

1.3.1 Reactivity control

This section will describe only those facilities on the OLP where fissile material is used or stored.

1.3.1.1 Reactivity control of the HFR core assembly

The reactivity control system of the HFR has two functions:

- The control of reactor power during operating conditions.
- The fast shutdown of the reactor in the case of disruptions or accident conditions;

The reactivity control system usually consists of 6 control rods which govern both power control and reactor shutdown. One control rod is controlled by the automatic power control system. The position of the other control rods (known as manual control rods) is adjusted manually. Each control rod consists of 3 sections: at the top, a neutron-absorbing section, beneath which is a fuel section, followed by an aluminium extension piece connected to the control rod drive mechanism. The drive mechanisms, which are located beneath the reactor vessel, can move the control rods upwards and downwards so that the fuel section or the neutron-absorbing cadmium section is moved into the core, with a reactivity-increasing and -lowering effect, respectively.

The drive mechanism is connected to the control rod by means of a mechanical and electromagnetical coupling. The first connection works mechanically and ensures that the control rod cannot become detached upwards from the drive mechanism. This mechanical coupling is only broken when the control rods are removed from the reactor core during a stoppage period. The second connection is provided by means of an electromagnet and ensures that the control rod also ascends as the drive mechanism moves upwards. In the case of a reactor shutdown, the electromagnetic connection is broken and the control rod drops down so that the neutron-absorbing cadmium section enters the core.

If, following a reactor shutdown, a control rod does not descend (fully) into the lowest position, the mechanical connection then ensures that the control rod is still pulled downwards by the drive mechanism. In the case of a reactor shutdown, in addition to the disconnection of the electromagnet, all drive mechanism shafts are automatically drawn downwards by the drive mechanisms.

A fast reactor shutdown is initiated by the reactor safety system, which breaks the current from the electromagnets. This is done by three switch circuits which each connect the electromagnetic coupling of two control rods. As a result, a simple fault in the drive mechanism system can affect the decoupling of at most two control rods. The aim here is that the system can regulate the reactivity of the reactor and control it to ensure that the reactor can be shut down safely and remain shut down for any applicable core composition and period in the cycle. In order to meet this requirement at all times, the following demands are placed on the reactivity control system:

Besides the fast shutdown system action (RSA) there is an alternative reactor shut down system available to minimize the flow of neutrons and thus stop the fission process in case of beyond design accident where the fast shutdown system has failed. This system is manually operated in the context of accident management. The system consists of a Cd-plate device which can be placed on the outside of the reactor vessel to create a strong neutron absorber against the outer wall of the reactor vessel near the reactor core. The Cd-plate reduces the reactivity with 300 pcm which is enough to make the reactor subcritical which provides a time interval to fully insert control rods and avoid (prompt) re-criticality.

A third inherent safety feature is the negative temperature coefficient for both fuel and moderator. This is the self-inhibiting nature of the chain reaction by feedback to the fission process of the temperature in both the reactor fuel assembly and the cooling water.

In addition, after a reactor scram, power reduction or manual shutdown the start of core poisoning due to the disturbance of the Xe/Sm-equilibrium will occur and reduce reactivity up to 42 hrs.

The HFR is not equipped with a neutron absorber system in the coolant. Therefore re-criticality due to density differences will not take place.

1.3.1.2 Reactivity control of (spent) fuel elements

The standard storage racks for the HFR (spent) fuel elements consist of an array of 6x7 aluminium rectangular tubes, placed on the bottom of one of the High Flux Reactor storage pools. In the walls of each tube a cadmium sheet with a thickness of 1 mm is embedded. Each tube contains one fuel element. The storage racks for (spent) HFR LEU fuel and control elements in the HFR storage pool is arranged in such a way that the storage racks are neutronically decoupled when submerged, so that criticality safety is guaranteed for any given number of storage racks in the storage pools.

Fresh HFR LEU fuel and control elements are stored in 2 vaults in specific racks containing neutron absorbing material in the basement of the containment building. Sub-criticality is ensured under all operational as well as flooding conditions.

1.3.1.3 Reactivity control of HEU targets for isotope production

Fresh HEU targets currently in use for Mo production (tubular targets for IRE and rectangular plates for Covidien) are stored in vaults in specific racks containing neutron absorbing material in such a way that sub-criticality is ensured under all operational conditions including flooding.

1.3.1.4 Reactivity control LFR fuel

The geometry of the pit storage in the LFR Hall is such that criticality of stored LFR fuel elements is excluded in all operational and accidental situations, including inundation and additional reactivity control is not necessary..

1.3.1.5 Reactivity control in HCL

The HCL Laboratories is divided in a number of zones that may hold fissile material. To prevent criticality the allowed mass present in each zone is limited to the so called Safe Mass of Fissile Material equal to 750 g-equivalent[1] ^{235}U . This mass is that small that the amount of fissile material cannot become critical under any circumstance (i.e. optimum moderation and reflection of neutrons). The mass present in a zone is controlled by the administrative procedure that controls the transfer of fissile material on the site.

In case a larger mass than the Safe Mass needs to be present in a particular zone special storages systems are installed. By spatial design and use of neutron absorbing materials criticality is prevented.

1.3.1.6 Reactivity control in the WSF

The storage of fissile material in the trenches and tube storage in the WSF is arranged in such a way that sub-criticality is guaranteed.

1.3.2 Heat transfer from reactor to the ultimate heat sink

ENSREG defines the ultimate heat sink (UHS) as 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, supplemented by an alternate UHS, for example a lake, a water table or the atmosphere.

Characteristic for those media is that they are almost infinite and are in their bulk hardly affected by the heat discharge (this means slight local temperature increase).

In this definition, for the HFR the ultimate heat sink is established by the Noordhollands kanaal for the supply of water and the North Sea for the discharge.

An alternate ultimate heat sink, as it is indicated by ENSREG, does not exist at the Petten site. However alternate cooling is provided by the (reactor) pools, this means that for their water supply the reactor pools can be considered as back up.

1.3.2.1 Heat tranfers means/chains for the nuclear facility and operation states

The reactor cooling water system of the High Flux Reactor (HFR) transfers the heat generated by the reactor core to the environment. The reactor cooling system is divided into a primary, secondary and pool cooling system. The primary cooling water system (PKWS) removes the heat from the reactor core

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[1] Equivalent mass ^{235}U = mass ^{235}U + 1.75 * mass ^{239}Pu

assembly. This heat is transferred to the secondary cooling water system (SKWS), which then releases it into the North Sea. In addition, the heat is transferred from the pools using the pool cooling water system (BKWS). This heat is also released via the secondary cooling water system into the North Sea.

Using the PKWS, the cooling of the HFR reactor core assembly takes place by means of circulating high purity demineralized water in a closed primary loop, consisting of the reactor vessel, a decay tank, parallel ion exchangers, three main coolant water pumps and the space between the plates of the three heat exchangers (Figure 1-20, Figure 1-21). The coolant water flows from the top to bottom through the reactor core. The speed between the fuel plates is 6.8 $m.s^{-1}$. The nominal coolant water flow through the core is 1.14 $m^3.s^{-1}$ (4100 $m^3.h^{-1}$). The inlet temperature of the primary coolant water varies between 40 and 50 °C, the temperature increase of the coolant water over the core is about 9.2 °C, at a reactor power of 45 MW.

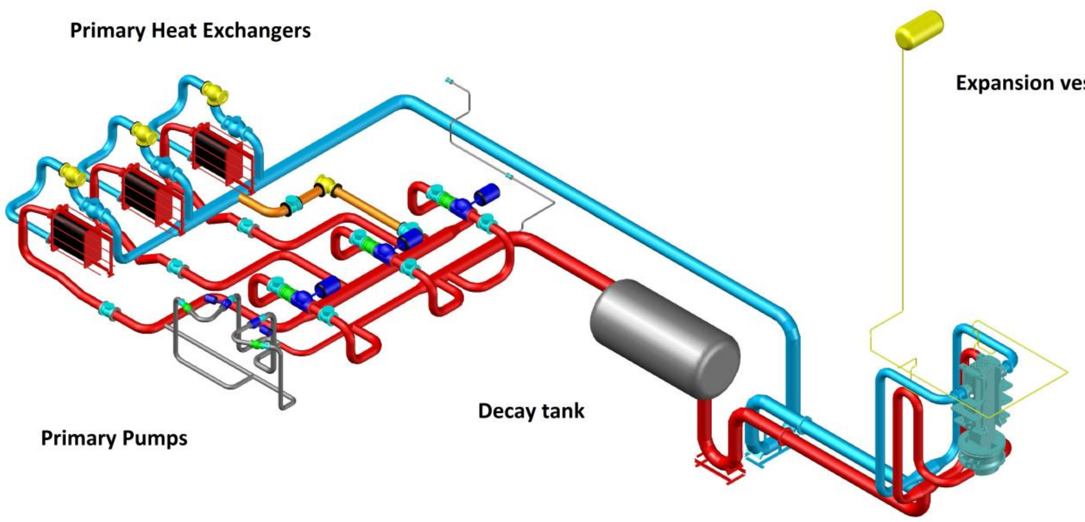
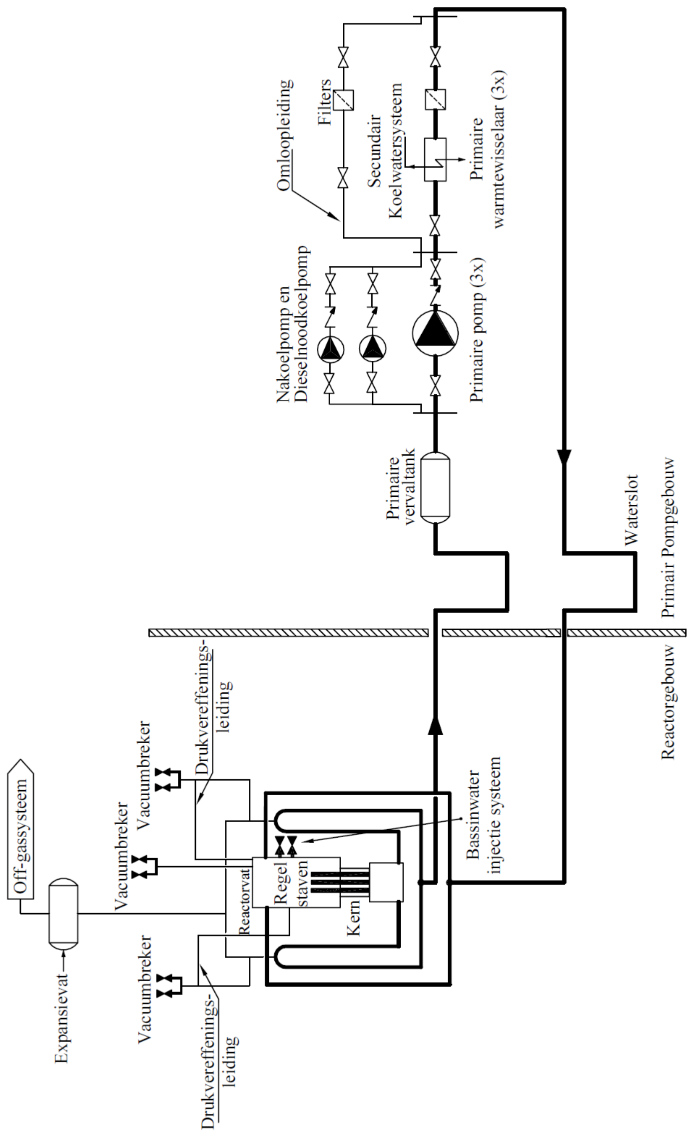


Figure 1-20: 3-Dimensional overview of the primary cooling water system (PKWS)

The heat generated in the primary system is transferred to the secondary coolant water using the SKWS, which is taken from the North Holland canal and after passing the heat exchangers discharged into the North Sea. The water flow varies between 1000 and 3125 $m^3.h^{-1}$ depending on the inlet temperature of the secondary coolant water.

The PKWS is equipped with a residual and decay heat removal pump system to maintain an adequate flow for the discharge of decay and residual heat from the reactor core, when the primary pumps will be stopped or have unexpectedly stopped. The residual and decay heat removal pump system consists of a set of



Key:

Off-gassysteem	Off-gas system	Dieselnoodkoelpomp	Emergency diesel cooling pump
Expansievat	Expansion vessel	Omloopleiding	Bypass
Vacuumbreker	Vacuum breaker	Filters	Filters
Drukvereffningsleiding	Pressure equalisation line	Secundair koelwatersysteem	Secondary cooling water system
Reactorvat	Reactor vessel	Primaire pomp	Primary pump
Regelstaven	Control rods	Primaire warmtewisselaar	Primary heat exchanger
Kern	Core	Waterslot	Water seal
Bassinwater injectiesysteem	Pool water injection system	Reactorgebouw	Containment Building
Primaire vervaltank	Primary decay tank	Primair Pompgebouw	Primary Pump Building
Nakoelpomp en	Decay heat removal pump and		

Figure 1-21: Schematic representation of the primary cooling water system (PKWS)

two pumps P-04-PMP and P-05-PMP (redundancy), with identical hydraulic properties, which are set parallel to the primary main pumps.

The primary facilities cooling system (PEKWS) provides cooling of collimators of the horizontal beam tubes and is supplied by and discharged to the PKWS. During operation of the reactor the cooling flow is generated by the pressure difference over the reactor assembly. During shutdown flow is generated with the facility pump P-07-PMP.

The pool cooling water system (BKWS) is a semi closed/open circuit in which demineralized water is circulated. The BKWS cools the vital reactor component like, the eastern and western core box wall, the outer core box cabinets and the reactor vessel bottom plug and removes heat generated by poolside isotopes production facilities connected to the pool experiments cooling water systems (BEKWS, U-BEKWS and HD-BEKW) with coolant in- and outlets into the reactor pool. In addition, the decay heat from spent fuel elements, the stored core components, irradiated experiments containing fuel or other radioactive objects is removed via BKWS.

Reactor shutdown condition	PKWS	SKWS	Decay heat removal pumps
Hot shutdown	3 primary pumps	Adjusted secondary cooling	--
Cooling down from hot to cold shutdown	3 primary pumps	Adjusted secondary cooling	--
Cold shutdown with closed PKWS	--	Adjusted secondary cooling	1
Cold shutdown with open PKWS	--	Adjusted secondary cooling	1

Table 1-4: The use of cooling systems under various reactor shutdown conditions

The BKWS has a cooling capacity of 3.2 MW at a flow rate of approximately 300 $m^3.h^{-1}$ and a pressure of 5 bar, provided by one of the two (redundant) parallel pumps. The heat absorbed by the pool cooling water system is discharged into the secondary cooling system.

The function of the three pool experiments basin cooling systems (BEKWS, UBEKWS and HDBEKWS) is to provide forced cooling with pool water for the experiments in or beside the reactor. The pump systems are installed on the second platform in the reactor building. Because the systems are used for cooling of fissile material, there are provisions for emergency cooling. A flow controller ensures a constant coolant water flow at slightly varying pressure in the pressure pipeline. On the contact pressure gauges alarms can be set, linked to the reactor protection and control system.

The use of the various cooling systems for a number of reactor shutdown conditions is given in Table 1-4.

1.3.2.2 Lay out of heat tranfer chains and protection against internal/external events

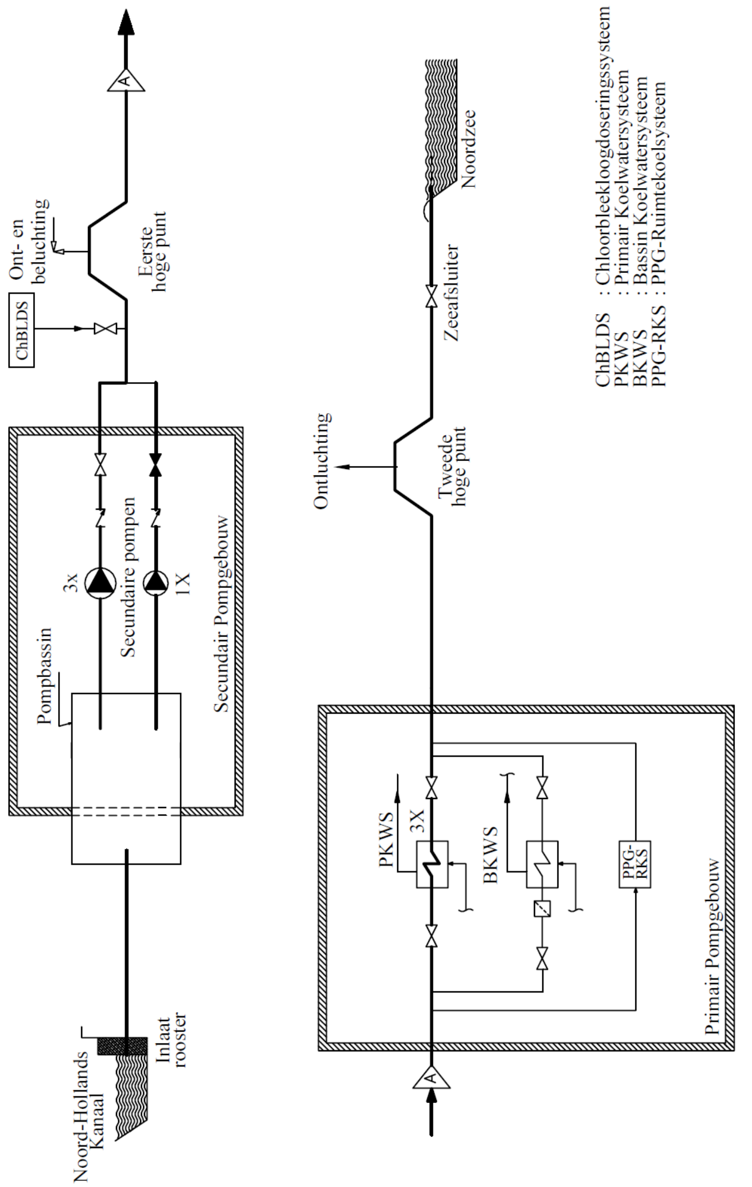
The primary cooling system (PKWS) is mainly installed in the primary pump building. The primary pump building is a semi-airtight building and is kept at a light under-pressure via a ventilation system. The three primary pumps and the three heat exchangers are each installed separately in a semi-airtight and protected cell in the primary pump building. Each cell has its own ventilation and cooling, as well as heating to prevent freezing during extended winter stoppages. Each heat exchanger has a cooling capacity of 22 MW. The three primary pumps are powered directly from HK-2.

The decay heat removal system is also installed in the primary pump building. The decay heat removal pumps are installed in a semi-airtight and protected cell in the primary pump building. This cell is provided with its own ventilation, cooling and heating. The pumps are physically separated from each other by a fire-resistant door between the two compartments of the cell (diverse location). The power supply to the motor is provided via control and operating cabinet RK1-NV1 for decay heat removal pump P-04-PMP in room 14A, and control and operating cabinet RK1-NV2 for decay heat removal pump P-05-PMP in room 14B. Both pumps are connected to an uninterruptible power supply with a battery back-up system (see section 1.3.5.4).

The two pumps and the pool water heat exchanger of the BKWS are installed in a separate semi gas-tight cell in the primary pump building and are fed directly from HK-2. The cell is provided with an air conditioning installation and a heating installation. The pool heat exchanger has a cooling capacity of 3.2 MW.

The secondary cooling water system (SKWS) runs from the Noord-Hollands Kanaal via the secondary pump building to the primary pump building, from where it runs out to the breakwater (longitudinal embankment) in the North Sea (Figure 1-22). The canal water enters the SKWS through an inlet grill. A inlet pipe of 1200 mm in diameter provides the connection between the inlet grill and the pit in which the rotating scraper grill is installed. From this pit a concrete pipe with a diameter of 1250 mm leads to the secondary pump building (SPG). The inlet pipe has a small gradient of 20 cm over a total length of 1600

m. Here the pipe ends into the intake put located outside of the secondary pump building. Two inlet slide valves give access to the separated north and south filtering basins, in which rotating filters are installed. Then the water flows through two inlet slide valves of the two filter basins into the pump building basin.



Key:

Noord-Hollands Kanaal	North Holland Canal	PKWS	Primary cooling water system
Inlaatrooster	Inlet grid	BKWS	Pool cooling water system
Pompbassin	Pump pool	PPG-RKS	Primary Pump Building room cooling system
Secundaire pompen	Secondary pumps	Primair Pompgebouw	Primary Pump Building
Secundaire Pompgebouw	Secondary Pump Building	Ontluchting	Ventilation
ChBLDS	Sodium hypochlorite dosing system	Tweede hoge punt	Second high point
Ont- en beluchting	Aeration and ventilation	Zeeafsluiter	Sea valve
Eerste hoge punt	First high point	Nordzee	North Sea

Figure 1-22: Schematic representation of the secundary cooling water system (SKWS)

In the SKWS secondary pump building, four pumps are installed in parallel in the secondary pump building and are fed directly from HK-1. Three pumps each with a rate of about 1450 $m^3.h^{-1}$ and one pump with a flow rate of 800 $m^3.h^{-1}$. The SKWS is equipped with a sodium hypochlorite dosing system to prevent growth of mussels and microbes in the secondary cooling

The three heat exchangers each with a cooling capacity of 22 MW are installed separately in a semi-airtight and protected cell in the primary pump building. Each cell is provided with heating to prevent freezing during long winter stops.

1.3.2.3 Availability of different heat transfer chains, constraints and possible extensions

The following alternative cooling systems/provisions are available.

- **The convection valves**. The reactor vessel is fitted with two convection valves located above and below the core. By opening these valves through controls on the third platform on the east side of the reactor pool, an open connection is established between the primary cooling water system (PKWS) and the reactor pool. In this way, the core can be passively cooled with pool water by natural convection. The direction of flow through the reactor core is then from bottom to top, in other words in the opposite direction to the original forced primary cooling water flow. The decay and residual heat from the reactor core is absorbed by the pool water and removed via the pool cooling water system (BKWS). If the pool cooling water system fails to work, the volume of pool water available is sufficient for at least 400 hours of cooling of the shutdown reactor.
- **The pool water injection system.** This system injects by gravity water from the pool into the reactor vessel to provide core cooling capacity in the case of “loss of coolant accident”. The controls for the pool water emergency injection system are located on the second platform on the north side of the reactor pool. The connections with the pool water injection system are located on the north side of the reactor vessel. The pool water emergency injection system is opened using key-operated keys from the control room. The two electrically operated pool water emergency injection valves are connected to an uninterruptible power supply (VZO).
- **Public water supplementation system.** The public water emergency supplementation system is used for providing tap water to the reactor pool during emergency situations. The water from the water ring mains net is channelled to the pool via the HFR pure water system (hydrophore system) and via a pipe which emerges above the water level of the pool. This system will only be used when the level in the reactor pool drops due to the use of the pool water emergency injection system for the reactor core. The drinking water will flow directly into the reactor pool upon opening of the supply valve. The drinking water for the primary pump building and the reactor building is taken off from the basement of the reactor auxiliary building via a drinking water pressure booster with a pressure of 5

bar. A second supply line, the so-called fall line, supplies water at a pressure of 0.6 bar and is directly connected to buffer basin 1 in the ECN clean water basements (Reinwaterkelder). Two pumps and a drinking water pressure booster tank are installed in the reactor auxiliary building. To ensure that the pressure booster water supply to the reactor building is guaranteed four buffer tanks are installed in the ECN clean water basement with a total capacity of 480 m^3. Buffer tank 1 with a capacity of 120 m3 is specifically reserved for the HFR.

1.3.2.4 Electrical power supply to heat transfer chains

The electrical energy supply to the HFR is provided by the following systems (See also section 1.3.5):

- Electrical power and lighting supply to HK-1 and HK-2 via the energy company's public grid, or upon the loss of the public grid, from the emergency power installation for HK-2 (exceptional);
- Emergency power K2A(NS) and K2R(NS), via public grid (normal operational status) or upon the loss of the public grid, from the emergency power installation;
- Emergency power (NV), via public grid (normal operational status) or upon the loss of the public grid, from two independent passive energy sources (NV-1 and NV-2);
- Uninterruptible Power Supply (Dutch abbreviation "VZO"), via public grid (normal operational status) or upon loss of the public grid, from two independent passive energy sources (VZO-A and VZO-B).

To assure operation during loss of off-site power the following systems are fed by the emergency power system:

- The pool cooling water system;
- The residual and decay heat removal system. This system is additional backed by batteries (2 pumps, design discharge capacity for 30 min per pump);
- The hydrophor system (Hydrofoor);
- One pump of the pool facilities cooling system supplying also water to the extended pool facilities cooling system in that situation;
- Two pumps of the high pressure pool facilities cooling system;
- Actuation of the pool water injection system and valves.

1.3.2.5 Cooling of heat transfer components

Not installed at the HFR.

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

The HFR is a tank-in-pool-type multipurpose reactor with interconnected pools. Therefore the heat chains and back-up systems described under the previous section are applicable for this section.

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

No special systems installed at the HFR since the HFR is a research rector and not a power plant.

1.3.5 AC power supply

1.3.5.1 Off-site power supply

1.3.5.1.1 Reliability of off-site power supply

The off-site power supply in the Netherlands is in general reliable. The mean failure probability of the 10 kV power supply to a customer in the Netherlands is 0.27 per year. The mean duration of a power supply failure is 70 minutes.

During the time period from 1997 up to and including 2002 nine occurrences of loss of offsite power took place (in most cases minor voltage drops causing a trip of the primary pumps). The plant specific loss of offsite power frequency is: 1.5 per year.

This plant specific number was much higher than the average number for the Netherlands 0.27 per year for that period. A Bayesian update was performed using the average number for the Netherlands as a prior distribution. The lognormal distribution was used as a prior distribution with a mean value of 0.27 and an error factor of 9. The results of the Bayesian update is a posterior distribution with a mean value of 1.27 per year and an error factor of 1.0.

In the period 2005 up to august 31st 2011, 5 occurrences of loss of offsite power took place occurred resulting in a reactor scram. The mean duration for the loss of off-site power was less than 1 minute. The plant specific loss of offsite power frequency for that period is: 0.8 per year.

1.3.5.1.2 Connections of the nuclear facilties with external power grids

The main 10 kV electrical distribution system of the OLP is presented in Figure 1-23. The preferred 10 kV supply is provided by two separate feeder cables from the Liander 50 kV public grid switchyard Schagen that is equipped with a 10 kV dual bus system. In the field the 10 kV feeders are rolled out in separate trace's at a considerable distance of each other (approx. 1 km) so sufficient physical separation is ensured. Both feeders are electrically separated from each other by several manual operated normally open (NO) circuit breakers.

1.3.5.2 Power distribution inside the nuclear facilties

1.3.5.2.1 Main cable routings and power distribution switchboards

In the diagram the preferred 10 kV supply circuit for the HFR is marked up in red. Tie-in at the OLP is in the Secondary Pump Building (SPG) where a 10 kV Magnefix substation is located. A 400 kVA transformer 10 kV/0.4 kV provides power to switchboard HK1 for the secondary pumps and the other SPG building loads. Via substation Covidien 3 a feeder cable runs to an open construction 10 kV substation ECN 2 in the Primary Pump Building HFR, serving four 630 kVA transformers 10 kV/0.4 kV for the HFR switchboards.

The second 10 kV feeder, marked up in blue, supplies electrical power to the remaining NRG nuclear facilities at the OLP. After tie-in at the ECN wind turbine test field it serves several ECN substations as indicated in the diagram.

Below an oversight is given of the division of these nuclear facilities over the substations:

ECN 3: HCL-MPF and GBD Building

ECN 4: LFR, STEK Hall, DWT and WSF

ECN 6: HCL-RL

ECN 7: JGL

In case of a long term LOOP at the HFR site caused by disturbance of the 10 kV feeder to the SPG, an alternative circuit can be established by first opening the circuit breaker to field 141 at the SPG and

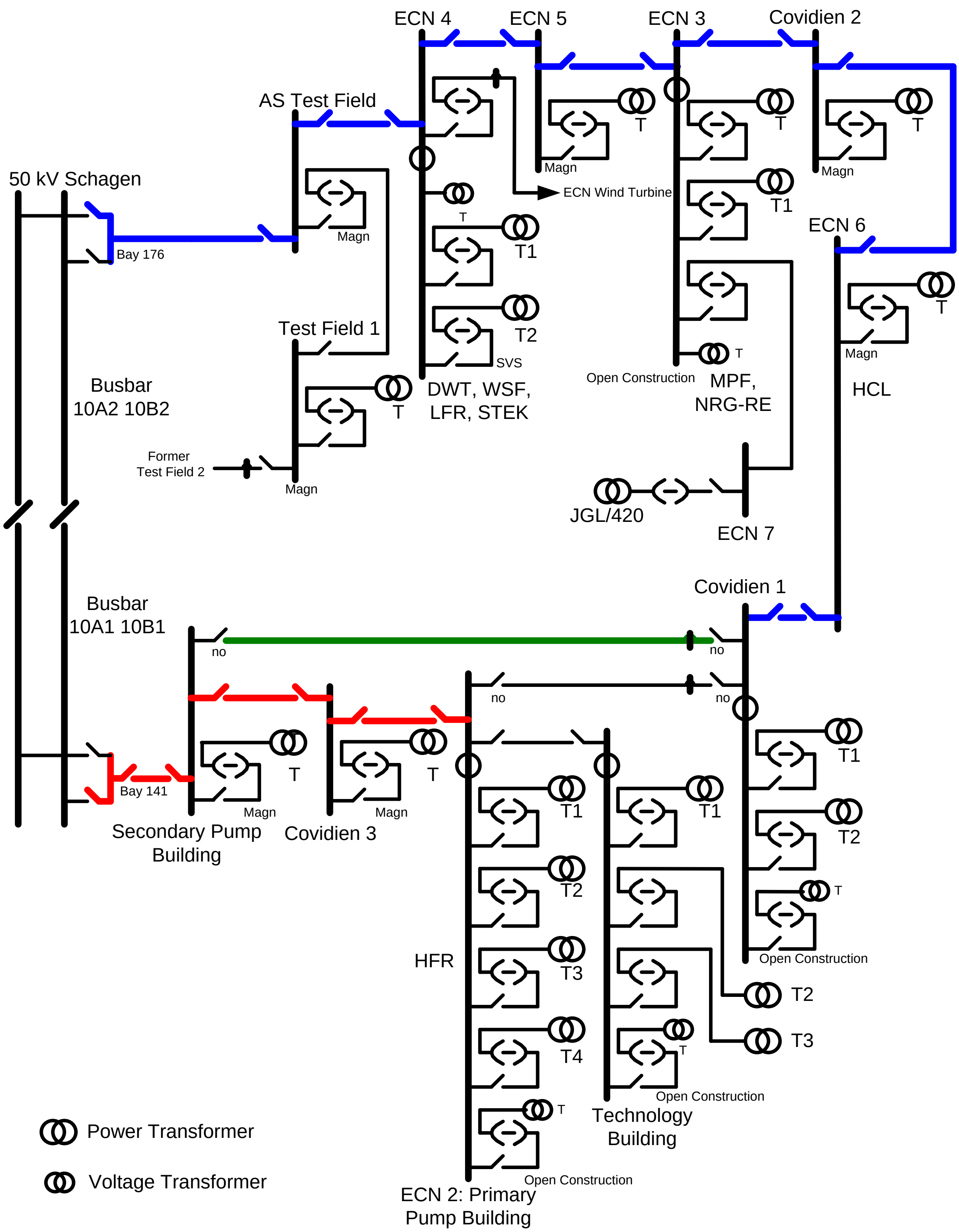


Figure 1-23: Main 10 kV electrical distribution Onderzoekslocatie Petten (OLP)

subsequently closing the normally open breakers in the SPG and the Covidien substations. This procedure has to be executed by Liander personnel that are on call. It can be done within a time frame of 3-4 hours. In this way long term emergency diesel generator operation of the OLP-EPS can be avoided. The alternative circuit is marked up in green.

The HFR electrical power supply system is designed to provide electrical power to the operational and safety systems in such a way that under all circumstances sufficient off-site and on-site power is available for the HFR. It is qualified for the earthquake response spectrum of the HFR. In Figure 1-24 a one-line diagram of the electrical main distribution system is given. In this diagram, the 0.4 kV switchboard HK1 in the SPG is not shown. Under normal circumstances 10 kV off-site power from the public Liander grid is transformed to 0.4 kV by four 630 kV transformers in the ECN 2 substation in PPG-015 and divided over two 0.4 kV main switchboards HK2 Bus 1 and HK2 Bus 2. Two transformers are able to carry the full HFR HK2 Bus 1 load. NRG staff is not allowed to execute switching actions in the ECN 2 substation. This is solely the responsibility of Liander personell.

Switchboard HK2 Bus 1 provides power to non-safety and safety related systems and, under normal circumstances, to the emergency power switchboards K2A (NS) in the RBG basement and K2R (NS) in the PPG. Switchboard HK2 Bus 2 provides power to the bypasses of the UPS's inverters (see 1.3.5.4) and is connected to the EPS diesel generator C back-up feeder. The cross-feed switch KS-1 between the switchboards is normally locked open, manual operation of this switch is subject to a procedure. The same procedure applies to the diesel generator C back-up connection to HK2 Bus 2.

Under normal operational circumstances the emergency power switchboards K2A and K2R are both powered by HK2 Bus 1. At LOOP the connection to this bus will be cut and will K2A be powered from diesel generator A and K2R from diesel generator C. The emergency power switchboards K2A and K2R power all safety related 0.4 kV HFR systems and components.

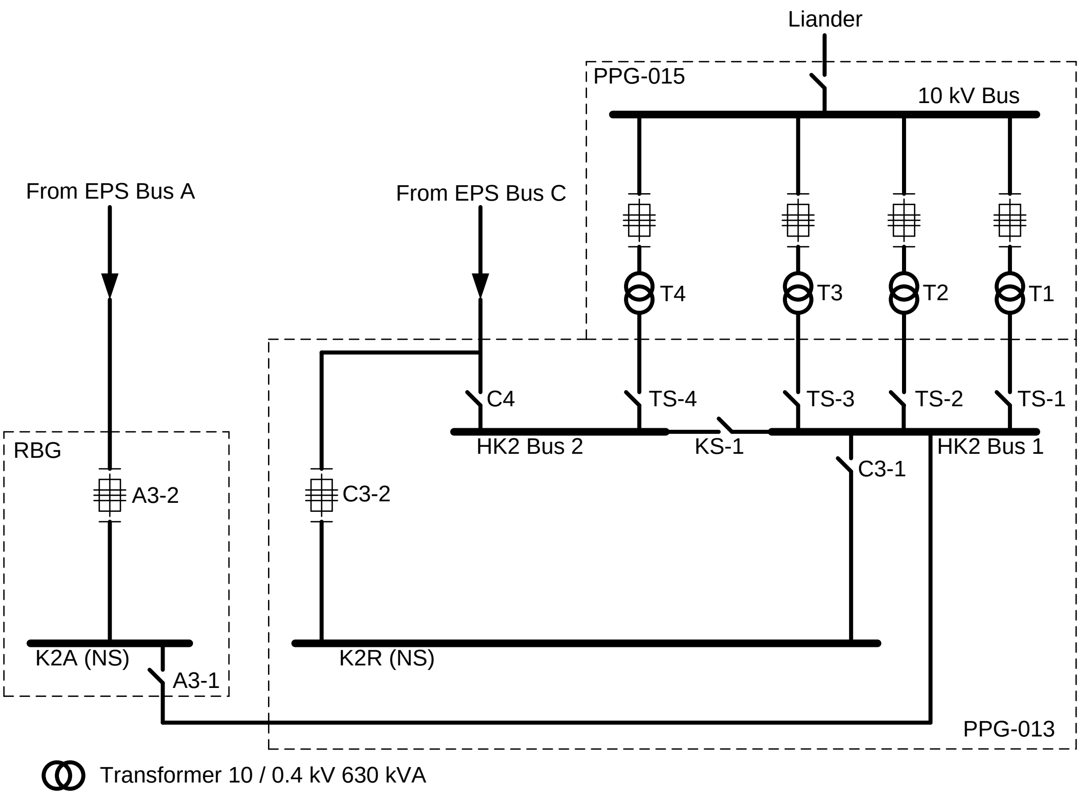


Figure 1-24: One-line diagram HFR electrical power supply system

1.3.5.2.2 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 of the nuclear facilities. 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.

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

1.3.5.3.1 On site Emergency Power Supply System

The on-site emergency power supply system (EPS) provides electrical power on a voltage level of 0.4 kV to its users in case of a loss of external electrical 10 kV power from the public grid. It is qualified for the earthquake response spectrum of the HFR. In Figure 1-25 an one-line circuit diagram of the EPS is given.

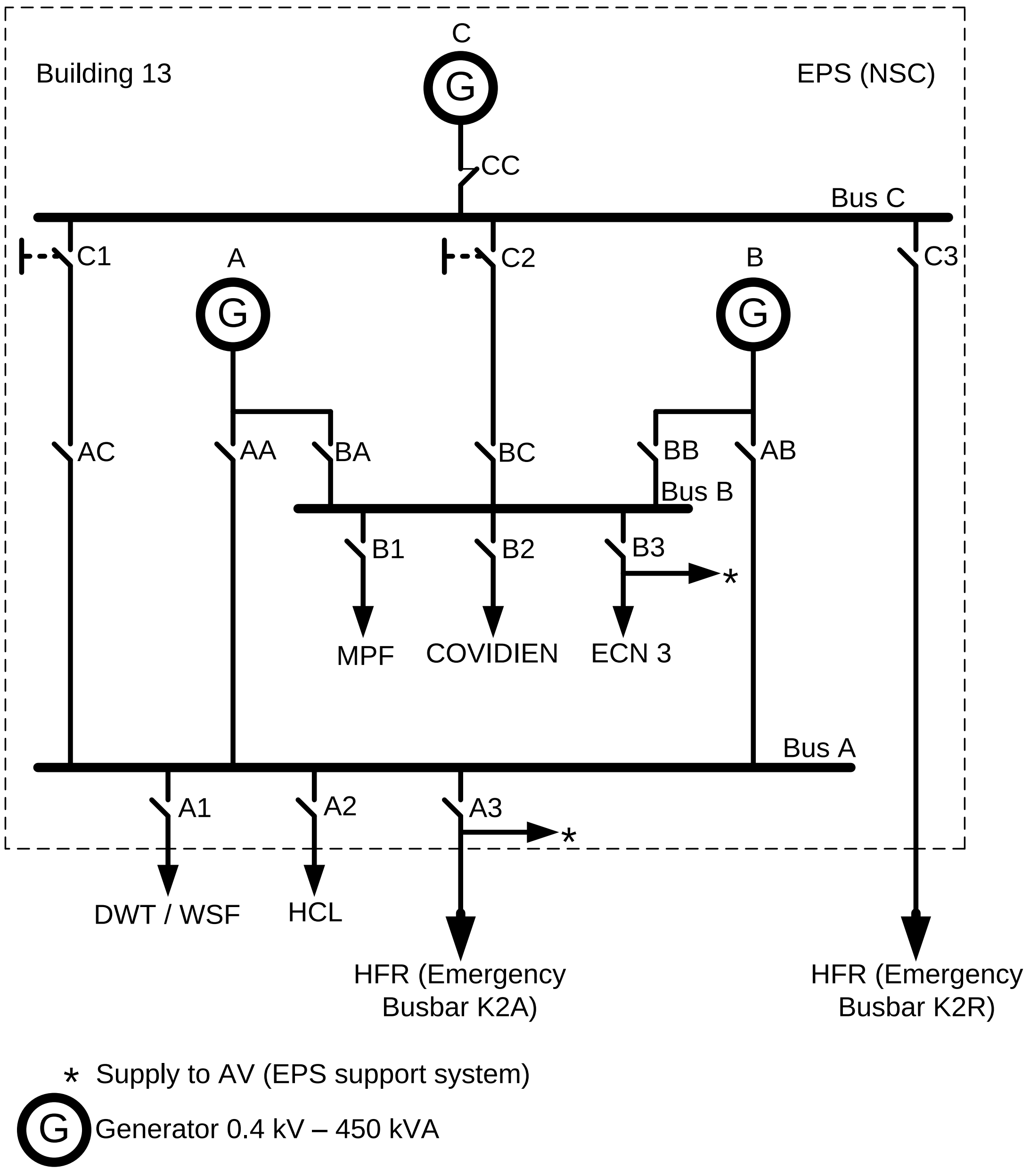


Figure 1-25: One-line diagram OLP Emergency diesel generator system (EPS)

1.3.5.3.2 Redundancy and separation of the EPS

The system consists of three 0.4 kV-50 Hz diesel generator sets A, B and C, each with a nominal design power of 450 kVA with a continuous overload capability of 10%. However, during operation the real connected load is rather low; it varies from 45-68 %.

The diesel generators sets are housed at ground level in a separate building on the OLP some 50 m from the HFR. Inside the building the diesel generators A and B are located in one room along with their day tanks and starter batteries. For fire protection reasons diesel generator C, its day tank and starter battery are located in an adjacent room. The same physical separation is applied to the rooms for the matching switchboards and control consoles. All three diesel generator sets are fuelled from a common underground main tank just outside the EPS building.

In case of a loss of off-site power from the Schagen 10 kV switch bay 141, the diesel generators A and C will start automatically by the under-voltage protection units on the users main bus bar systems and power up their bus bars within approx. 7 seconds. Should diesel generator A fail to start or fail to take on its load then generator A is disconnected from the system and generator C will take over its load on bus bar A.

Should diesel generator set C fail too, then after 1 minute the load of diesel generator B will be shed and generator B will be connected to bus bar A, thus restoring emergency power to its users. The possible effects of this load-shedding have been evaluated.

In case of LOOP from Schagen 10 kV switch bay 176 the diesel generators A, B and C will start and power up. Of course, when both 10 kV feeders fail at the same time or the 50 kV grid of Schagen is not available, establishing a complete LOOP, also all three diesel generators will start at once and power up.

Below an oversight is given of the division of the loads, including their magnitude in brackets, that are connected to the diesel generators:

- Generator A: HFR (emergency bus K2A), DWT/WSF and HCL (202 kVA);
- Generator B: HCL-MPF, Covidien, substation ECN 3 that serves JGL and GBD-Building (306 kVA);
- Generator C: HFR (emergency bus K2R) (10 kVA).

In Appendix A.3 a more comprehensive listing of the EPS emergency power loads is given.

1.3.5.3.3 Runtime and backup measures for EPS

The on-site emergency power system (EPS) is equipped with a common main tank with a capacity of 10 m^3. At a level of 4 m^3 a top-up order is given to the oil company. There is no formal procedure for this action. A minimum volume of 0.5 m^3 cannot be used for operation. This leads to a worst case net capacity of the main tank of 3.5 m^3.

Each diesel generator is equipped with a day tank with a capacity of 0.6 m^3. At an alarm on the 20 % level the diesel generator is shut down. So per diesel generator the effective capacity of its day tank is 0.48 m^3. At the full design load the fuel consumption per diesel generator is 0.1 $m^3 . h^{-1}$, the real connected loads however are considerable lower than the design loads. Taking these connected maximum loads, as listed above, into account, while as a minimum, fuel consumption of a running diesel is set for 20% of its maximum load, the following stand-by times after a LOOP are valid for the EPS:

- Diesel generator A: 37 h;
- Diesel generator B: 33 h;
- Diesel generator C: 50 h.

Additionally, at the DWT facility diesel fuel is stored in a tank. It is normally in use for cleaning purposes and for the fork-lift trucks and has a minimum stock of 0.5 m^3 and a maximum of 3.2 m^3. The tank can be picked-up by a fork-lift truck and transported to the EPS building in case of an emergency in order to top-up the EPS main fuel tank.

No specific arrangements are in place in case of infrastructural problems resulting from for example earthquake or flooding.

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

1.3.5.4.1 Diverse backup power supplies

The HFR is equipped with two un-interruptible power systems (UPS) NV and VZO. The NV system consists of 2 inverters 0.4 kV-50 Hz of 120 kVA with battery back-up. The VZO system consists of 2 inverters 0.4 kV-50 Hz of 60 kVA with battery back-up.

1.3.5.4.2 Redundancy and separation of the backup power supplies

The inverters, NV1 and NV2, and their matching battery sets are located in separate rooms on the ground floor of the PPG. In case of LOOP-SBO1 when no on-site AC power source is available, the battery back-up kicks in without interruption of the 0.4 kV supply to the connected loads. The design back-up time for each battery set is 30 minutes at a load of 24 kW. In case of failure of the K2A or K2R emergency buses bypass feeders to the HFR main switchboard HK2 bus are available for charging the

NV system. The NV inverters can be described as a redundant system with physical separation between the inverter trains.

Inverter NV1 gets its primary power from emergency bus K2A. Its output supplies decay heat removal pump P-04-PMP, two smaller experiment cooling pumps (BEKWS and PSF) and some safety related instrumentation like aerosol monitors. Inverter NV2 gets its primary power from emergency bus K2R. Its output supplies decay heat removal pump P-05-PMP.

The VZO system consists of 2 inverters 0.4 kV-50 Hz of 60 kVA with battery back-up. The inverters, VZO-A and VZO-B, hooked-up in parallel to a common distribution board, are located in a common room in the basement of the reactor outbuilding (RBG). Their matching battery sets are located in another common room in the RBG. In case of LOOP-SBO1 when no on-site AC power source is available, the battery back-up kicks in without interruption of the 0.4 kV supply to the connected loads. The design back-up time for each battery set is 30 minutes at 60 kVA. In case of failure of the K2A emergency bus a bypass feeder to the HFR main switchboard HK2 Bus is available. Despite their being located in common rooms, the VZO inverters and batteries can be described as a redundant system because of the physical separation between the inverter trains that meets the requirements of Standard IEEE 384.

The VZO inverters supply electrical power to safety and safety related systems, the most important among them are the reactor protection system, the control room annunciator system, the instrumentation in the HFR control room, the data-acquisition systems DAS-DACOS and the 110 V DC reactor control rectifiers DC-A and DC-B. These 110 V DC systems are equipped with battery back-up too.

1.3.5.4.3 Runtime and backup measures of the backup power supplies

The NV-system as described above has the following battery stand-by times based on the real installed load on this system:

- Battery NV-1: 34 minutes;
- Battery NV-2: 77 minutes.

Compared to the design stand-by time of 2x30 minutes there is a margin of 51 minutes here.

The VZO-system has the following battery stand-by times based on the real installed load on this system:

- Battery VZO-A: 90 minutes;
- Battery VZO-B: 90 minutes.

Compared to the design stand-by time of 2x30 minutes there is a margin of 2 hours here.

The VZO-system also supplies power to the 110 V DC rectifier system that supports the HFR control system for the 0.4 kV switchboards. This system consists of two rectifier-battery combinations feeding on a common distribution board. Each one has a stand-by time of approx. 5 hours. This means that for at least 10 hours after the VZO batteries are exhausted 0.4 kV control actions on the 0.4 kV switchboards can still be executed. This enables the operator to preset components (mainly valves) in a safe mode which will be resumed in case electrical power is restored.

1.3.5.5 External power sources

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

There are no connections to other power plants.

1.3.5.5.2 Transportable power sources

There are only limited transportable power sources present on the OLP. The 2 fire engines are equipped with 2.5 kW generators and a back-up generator is located in the fire department. In addition, a 3.5 kW portable generator is located in the basement of building 015. All these transportable generators use benzine as fuel.

There are currently no actions foreseen or procedures in place to get AC power from transportable off-site sources

1.3.5.5.3 Information on each power source

Not applicable

1.3.5.5.4 Preparedness to utilise an extenal power

There are currently no provisions to use external power.

1.3.5.5.5 Batteries for DC power supply

Not applicable

1.4 Significant differences between units

Not applicable

1.5 Scope and main results of Probabilistic Safety Assessments

1.5.1 Scope of the Probalistic Safety Assessment (PSA)

1.5.1.1 Methods and regulations

The probabilistic safety assessments as applied to the nuclear facilities at the OLP, are neither full scope nor based on a detailed event tree and fault tree analyses, although the Competent Authorities requires a full scope PSA (at all levels) for the Dutch Nuclear Power Plant. Since early 90-ties no practical guidelines on PSA level-3 were available, the Competent Authority did develope a PSA-level 3 guide to be applied to the nuclear facilities in the Netherlands. Since no full PSA level-1 and level-2 for non-power reactors were required by the Competent Authorities for the OLP facilities (before 2000 even not for already operational research reactors), guidance has been found in IAEA safety series G1, where an analysis based on a list of postulated initiating events (PIE's) was recommended. For external PIE's, a draft safety guide NVR 3.1 has been developed by the Competent Authority. For internal PIE's, guidance has been found in the generic list for research reactors presented by the IAEA. As far as PSA level-3, concerns, regulations on radiation protection have been adapted to Euratom requirements and dose limits and risk criteria were established by the Competent Authority for safety analyses. For risk assessment of regular operations with open and closed radioactive sources such as used in the OLP facilities, a guideline mr-AGIS has been developed by the Competent Authority.

For the nuclear facilities on the OLP, except the HFR, the dominating PIEs listed in Table 1-5 are fire, airplane crash (APC) or random system failure. The table also shows what the result was of the screening of the PIEs for earthquake, flooding and extreme weather, what the enveloping source of the PIEs was, and in which year the latest revision of the Safety Reports was published.

In consultation with the Competent Authority, NRG has performed a Risk Scoping Study (RSS) for the HFR that has been also reviewed by an International expert mission (IAEA/INSARR mission).This RSS was carried out as an integral part of the application in 2003 of a license renewal for the HFR to use LEU fuel. In addition, the results of the HFR-RSS are used as input for subsequent improvements directed to preventive measures and the effectiveness of mitigation measures. Compliant with regulatory requirements, the objective of the RSS was:

- The identification of initiating events with the potential to lead to core damage or unusual release of radioactive materials (such as failure of experiments);
- The estimation of the core damage frequency;

- The estimation of the frequencies containment failure and the resulting releases to the environment, the so-called source terms;
- The estimation of off-site consequences of these source terms resulting in estimates of public risks.

The level-1 RSS covered internal events categorised into 35 initiating events including internal flooding and fire under full power conditions. In previous studies (not part of the RSS) impacts of seismic events have been analysed.

The Level-1 RSS involved:

1) Identification and analysis of accident sequences, that may contribute to core damage, and;
2) Quantification of the frequency of core damage for these sequences, including a computerised plant model for the Level-1 analysis, using event and fault trees;
3) Identification of potential radioactive release the that result from events sequences without core damage, such as failure of experiments.

The level of detail in the systems analysis has been such that the effectiveness of potential hardware modifications can be demonstrated (one of the goals of this RSS). The HFR 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 unavailability’s. Consideration was given to dependencies and human failures; dependency matrices has been developed for all systems (front-line and support systems).

The level-2 RSS resulted in the quantification of the radiological source terms, i.e. the probability, magnitude and timing of radioactive releases to the environment. A two-step approach has been used: in the first step the effects of the non-successful end-states which result from the level-1 analyses, have been analysed. From this analysis, four representative accident scenarios have been obtained, including their frequency of occurrence. the second step of the level-2 RSS, the progression of the four accident scenarios have deterministically been analysed to determine the resulting source terms. Two of these scenarios involved severe core damage and were analysed using MELCOR

The level-3 RSS evaluates the radiological consequences of the source terms for members of the public as described in sections 1.5.1.2 and 1.5.1.3 .

Radiological consequences of postulated accidents at the nuclear facilities at the OLP, except the HFR, have been calculated semi-probabilistic, using the NRG NUDOS code and dose conversion coefficients developed by the ICRP (ICRP publications 60 and later ones).

The radiological consequences for the HFR were calculated with the computer programmes Microshield and COSYMA.

	Dominating PIE	Earthquake	Flooding	Extreme Weather	Source PIE
		represented by:	represented by:	represented by:	
LFR (2000)	APC	APC	screened out	screened out	**IAEA SS-35-G1**
HCL_RL (2000)	APC, Fire	fire	screened out	screened out	**IAEA SS-35-G1**
HCL_MPF (2004)	APC, Fire, random system failure	APC	random system failure	screened out	**IAEA SS-35-G1**
DWT (2007)	APC; Fire; random system failure	random system failure	screened out	random system failures	**IAEA SS-35-G1**
WSF (2005)	APC	APC	Screened out	APC	**IAEA NS-G-1.6 NS-G-3.5 NS-G-1.5**

Table 1-5: PIEs that dominate the radiological consequences of potential accidents.

1.5.1.2 Criteria for assessment of design basis accidents

Consequences of design basis nuclear accidents are assessed in the Netherlands on the basis of dose limits that depend on probability of the accident and the exposed group of the general population: i.e. children (till 16 year) and adults, see Table 1-6

This maximum dose should be calculated for so-called critical group of the exposed population, see also mr-AGIS. This critical group is selected as such that 95% of the exposed population would receive a lower dose than this critical group. This dose of the critical group depends also weather statistics. Certain weather conditions could lead to relatively higher doses than more common weather conditions do. Practically, the maximum dose (to be compared with the limits) is selected as the 95-percentile value of the calculated distribution of maximum doses due to an accidental release.

The calculation of the doses should encompass all exposure pathways (including ingestion) and no protective countermeasures should be taken into account.

1.5.1.3 Criteria for assessment of beyond design basis accidents

According the regulation, the risks due to beyond design basis accidents should comply with the following limitations:

- The probability of death due to the consequences of a beyond design basis accident at a facility should be lower than 1.0 E-06 per year for any person who is unprotected and permanently located at given distance from the facility.
- The probability of having at least 10 fatal victims due to a beyond design basis accident at a facility, should be less that 1.0 E-05 per year, or for a group of n-times more fatal victims, less than n^2 lower (for 100 victims less than 1.0 E-07 per year).

Maximum allowed effective dose		
Frequency [F] per year	**Adults**	**Children**
F ≥ 1E–1	0.1 mSv	0.04 mSv
1E–1 > F ≥ 1E–2	1 mSv	0.4 mSv
1E–2 > F ≥ 1E–4	10 mSv	4 mSv
F < 1E–4	100 mSv	40 mSv
	Maximum allowed effective dose on the Thyroid	
All design basis accidents	500 mSv	500 mSv

Table 1-6: Limits for maximum dose of design basis accidents.

1.5.2 Main results of the PSA of OLP nuclear facilities

1.5.2.1 HFR

1.5.2.1.1 Results of the level-1 RSS

Each accident starts with an initiating event. The list of initiating events in IAEA-TECDOC-400 is used as a starting point. This list was complemented with HFR operating experience and lists from other PSA's. For each initiating event its applicability for the HFR was investigated. This resulted in the inclusion of 35 initiating events, for which the annual frequency was determined. After an initiating event, the operator or (automatic) safety systems will act to mitigate the potential undesired negative effects. The responses of the operator and systems are analysed using event trees. Event trees show the combinations of system failures that potentially lead to core damage or to a release of radioactive material from other sources than the core, such as an experiment. The event tree reflects the level of defence and provides all necessary information about the various systems that will determine the plant response for a particular initiating event, based on success criteria. Figure 1-26 shows such an event tree. In this figure, the possible sequences after a blockage of flow (the initiating event) are shown. When no action, either by the operator or automatic system (which result in a SCRAM) is taken and the reactor could not be shut-down (manually) this sequence result in core damage (CD). The failure probabilities on demand for the required safety systems are calculated using fault trees.

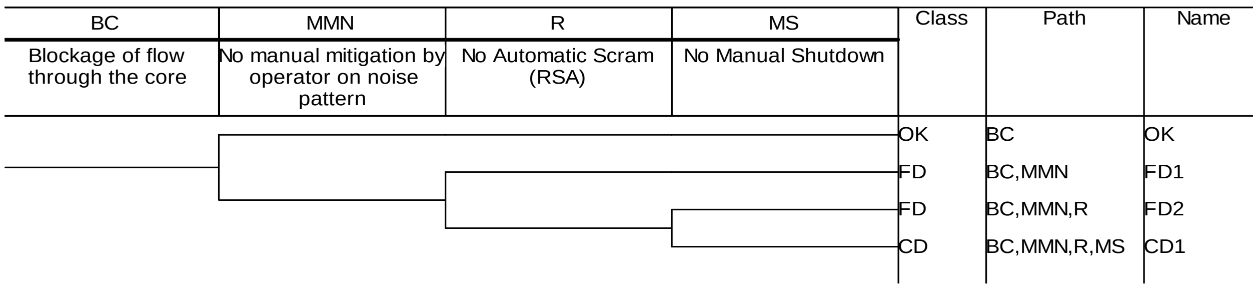


Figure 1-26: Example of an event tree

The analysis identified 54 different accident scenarios with 7 different end states varying from a safe end-state or end-states with limited to major damage to the core as presented here:

1. The state OK in which the plant is in a stable and save condition with long term cooling assured;
2. The state CD (core damage) encompasses various conditions:
    a. no adequate core cooling resulting in damage to the core;
    b. Reactor not scrammed on demand.

3. The state FD (fuel damage) with a limited damage to the reactor fuel and radioactive contamination limited to the reactor coolant system;
4. The state SR0 defined as a very small release inside the containment;
5. The state SR1 defined as a small release outside the containment;
6. The state SR2 defined as a small release inside the containment;
7. The state FR with a release of radioactive material from an experiment inside the containment.

The results of Level-1 RSS are presented in Figure 1-27.

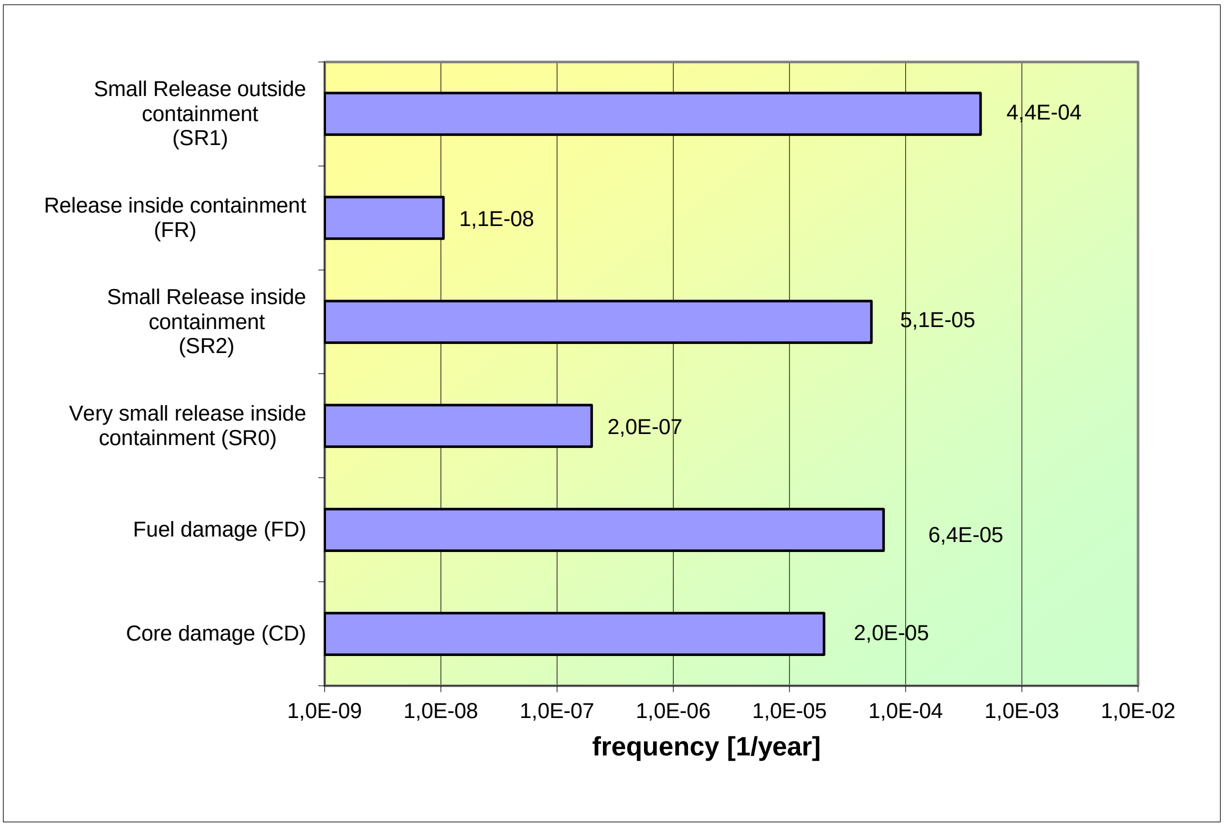


Figure 1-27: Quantified frequencies of the end-states.

End state CD: Core damage

The frequency of the end state Core Damage is 2.0 E-5 per year. The highest contributors to the Core Damage Frequency are Loss of Offsite Power (LOOP), Internal Fire and the Heavy Load Drop scenario that causes a Large LOCA outside the pool.

End state FD: Fuel damage

The frequency of the end state FD (local fuel damage) is 6.4 E-05 per year. Fuel damage is dominated by core bypass scenarios (flow through the core is blocked) and unbalanced rod position.

End-state SR0 (very small release inside the containment)

This end-state describes (frequency 2.0 E-07 per year) the situation in which pool water enters the piping corridor through the damaged bottom of the pool after impact of a heavy load. The primary system however is not damaged, preventing primary water to enter the corridor.

End-state SR1 (small release outside the containment)

Due to the assumption that the water of the primary circuit is contaminated and thus leads to a small release, the frequency of the end-state SR1 equals the sum of the LOCA frequencies for leakage's outside the containment: 4.4 E-04 per year.

End-state SR2 (small release inside the containment)

The total frequency of the end-state SR2 equals 5.1 E-05 per year. The relevant initiating events are a failed experiment and a container drop in the spent fuel pool.

End-state FR (release inside the containment)

The total frequency of end-state FR is estimated to be 1.1 E-08 per year, which is a low number in itself. The only contributor is one of the sequences following failure of an Experiment in the Pool Side Facility.

1.5.2.1.2 Results of level-2 RSS

The end results of the level-2 analysis (such as the frequency of core damage) are not used in the level-1 and level-3 analysis. The level-1 results that are used are the 54 identified unique accident sequences. For each of these 54 accident sequences the possible progression of the accident up to a release into the environment was evaluated. For this evaluation a Containment Event Tree (CET) has been set up, covering the most important events that influence the accident progression up to the release of radioactive material into the environment. These events are:

- Level of core damage (high/low/no);
- Level of fuel damage due failure of an experiment (high/low/no)
- Release path: from primary system to: 1) primary pump building, 2) reactor building or 3) ventilation building;
- Reaction of the ventilation, off-gas and stack valve system on activity monitoring;
- Containment failure: yes/no;
- Time between start and release to environment: before/after 12 hours.

The resulting CET has 97 end states, representing 97 different releases. These releases are grouped based on similar fission product release characteristics. This results in the definition of 9 Source Term Categories (STC). For each STC the frequency of occurrence and the dominant accident sequence was determined (see Table 1-7).

STC	Frequency [1/a]	Contribution [%]	Source term characteristics	Representative contributor
1	6.2 E-08	0.011	Core damage: high; Release path: through PPG within 12hrs	Large LOCA; outside containment
2	5.9 E-10	0.000	Core damage: high; Release path: through PPG within 12hrs	Large LOCA; outside containment
3	6.0 E-07	0.103	Core damage: high; Release path: through Reactor building within 12hrs	Intermediate LOCA; not towards pool; inside containment
4	1.6 E-05	2.808	Core damage: high; Release path: through Reactor building after 12hrs	Intermediate LOCA; not towards pool; inside containment
5	3.6 E-09	0.001	Core damage: high; Release path: through Reactor pool within 12hrs	LOCA towards pool; failed scram
6	4.3 E-09	0.001	Core damage: high; Release path: through Reactor pool after 12hrs	LOCA towards pool; failed scram
7	3.1 E-05	5.431	Core damage: low; Release path: through Reactor pool; primary cool water system closed	Damaged fuel containing experiment
8	6.4 E-05	11.167	Core damage: low; Release path: through Reactor pool; primary cool water system open	Partial blockage core; failed scram
9	4.6 E-04	80.478	No core damage	Intermediate LOCA; scram & decay heat removal successful; outside containment
Total	5.8 E-04			

Table 1-7: Frequency and characteristics of the 9 Source Term Categories (STC)

1.5.2.1.3

Based on the resulting frequencies of the 9 source terms and knowledge of the accident progression of the dominant accident sequences, the 9 STC's were further reduced to 4 accident scenarios. These final accident scenarios are representative for the calculation of the consequences for the environment. The resulting representative 4 accident scenarios are:

1. **Large LOCA into the Primary Pump Building (PPG) with a failed jacket pipe (STC 1 + STC 2) with a total frequency of 6.3 E-08/year.**

   This scenario is a large LOCA in the primary pump building (PPG) at the vessel inlet side. The break is at the lowest level of the primary system located in the Swan Lake. Due to the forces of the break the newly constructed jacket pipe surrounding the piping is bypassed and primary water is blown off through the break into the Swan Lake area. Due to the modifications of the primary system the water level in the reactor core will drop slowly. Because no water can be added to the primary system (failure to open pool cooling valves and convection valves) the reactor core will become uncovered after some time, resulting in extensive damage to the core and a release of radioactive material into the primary system. This radioactive material is released into the Swan Lake and from there into the environment due to leakage of the PPG.

2. **Large LOCA in PPG, with an intact jacket pipe and with direct release via the Reactor Building to the environment. Representative for STC 3, 4, 5 and 6, with a total frequency of 1.7 E-05/year.**

   The initiator and accident progression is almost identical to the first scenario, the difference between the scenarios is the fact that in this case the jacket pipe surrounding the primary piping in the Swan Lake stays intact. Therefore the break is quickly covered by 5 meters of water and the release of activity is through the expansion tank into the reactor building. Because the ventilation and stack valves are not closed, release to the environment does occur swiftly.

3. **Failure of a fuel containing experiment in the reactor pool. Representative for STC 7 + STC 8, with a total frequency of 9.5 E-05/year.**

   Accident scenario 3 concerns situations in which there is locally restricted damage to the fuel or failure of an experiment. In case of STC 7 it concerns damage to a fuel target containing experimental set-up, with release of radioactivity through the reactor pool into the reactor building. For STC 8 it concerns restricted damage to the reactor core, but without opening of the primary system. Although in this second case the activity is not directly transferred to the reactor building it is conservatively binned in the same accident scenario as the experimental failure.

4. **Loss of primary water into the Pump Building due to a large LOCA, representative for STC 9, with a total frequency of 4.6 E-04/year.**

Here all systems function, thereby preventing any damage to the core. The representative accident scenario is the loss of primary water into the primary pump building due to a Large LOCA. All systems function, thereby preventing any damage to the core.

The accident scenarios 1 and 2 have been analysed with the MELCOR code The representative release fractions for the accident scenarios 3 and 4 have been determined directly from the damage state and the radiological analyses for the enveloping PIE's (Postulated Initiating Event, so called design based accidents). All resulting information is applied in the Level-3 RSS, the off-site consequence analyses.

1.5.2.1.4 Results of the level-3 RSS

The maximum of the individual risk for children and adult members of the off-site population have been calculated for the 4 representative accident scenarios derived in the Level-2 RSS. The maximum values for the individual risks of members of the population are those for children. For each of the four representative accident scenarios, the conditional risk, i.e. the risk given the specific accident scenario, has been calculated and presented in the Risk Scoping Study HFR: LEU. By combining these results with the accident probabilities as given in Table 1-7, the total maximum individual risk (resulting from all accidents) has conservatively been estimated, see Table 1-8. The value of this sum is below the regulatory criterion of 1.0 E-06 per year. Therefore the individual risk for members of the population as a result of the operation of the HFR at 50 MW with LEU fuel, complies with the regulatory requirements as stated in accordance with Article 6 sub 1 and the risk criteria stated in Article 18 of the Decree for nuclear installations, fissile materials and ores.

In case of the Large LOCA scenario representing STC 1+2 may result in early fatalities in the area close to the HFR onsite the OLP. The resulting societal risk complies with the regulatory criterion. It is even below the 1% value of this criterion.

Source term category	Frequency [1/a]	Max. indiv. Risk (child)	Representative accident scenario
STC 1+2	6.3 E-08	3.4 E-09	LLOCA with a failed jacket pipe, open containment
STC 3+4+5+6	1.7 E-05	1.4 E-07	LLOCA with an intact jacket pipe, intact containment
STC 7+8	9.6 E-05	2.3 E-09	Failure fuel containing experiment
STC 9	4.6 E-05	1.9 E-09	Loss of primary coolant in PPG
All		**1.4 E-07**	

Table 1-8: Maximum individual risk for children as a result of the representative scenarios for the HFR

1.5.2.2 Hot-Cell Laboratories

The Hot-cell Laboratories (HCL) building complex encompasses the Research laboratory (HCL-RL) and the Molybdenum Production Facility (HCL-MPF), the licence of this complex is adapted after an license application in 2007. The HCL-RL building has various hot cells for material research, including research of irradiated fissile materials. The HCL-MPF has two production lines for the separation of Molybdenum-99 from irradiated targets of fissile materials.

1.5.2.2.1 HCL-RL

In the safety report and safety analysis report of the HCL-RL two design basis accidents and three beyond design basis accidents were identified and the postulated scenarios have been analysed. These accidents and their radiological consequences are described in the safety analysis report and doses and risks are presented in Table 1-9, Table 1-10 and Table 1-11, respectively. The beyond design basis accident scenarios, a plane crash both without or with kerosene burning is applied for all facilities at the OLP and is based on the external event described in the NVR 3.1 Guideline for protection against external events.

Typical releases for accidents in the hot-cells are releases of fission products such as iodine and caesium that would escape when irradiated fuel (investigated in the hot-cells) is heated by fire. In some accident scenarios an amount of radioactive source material (production of radioisotope) is assumed to be present in a hot-cell. In case of a fire in this hot-cell, part of this source material will be released, such as Sr-90, I-131 and Ir-192.

From the individual risk in these tables, the maximum risk for the hypothetical group of children that lives permanently outside the OLP site is estimated to be less than 3 E-9 per year. For the (also) hypothetical group of adults that lives permanently outside the OLP site, the maximum individual risk is less than 3 E-10 per year.

1.5.2.2.2 HCL-MPF

In the safety report and safety analysis report of the HCL-MPF two design basis accidents and two beyond design basis accidents were identified and the postulated scenarios have been analysed. These accidents and their radiological consequences are described in the safety analysis report and the doses and risks are presented in Table 1-12, Table 1-13 and Table 1-14, respectively. During the production process, the irradiated uranium targets are dissolved resulting in dissolved uranium oxide and dissolved fission products. During the process, the noble gasses will be captured in active carbon delay filters and the resolved Molybdenum is captured in ion-exchange columns. The residual liquid waste from this production process includes other fission products such as Caesium and Iodine and is stored in the waste

tanks. After a storage and decay period, the liquid waste is transported to COVRA for conditioning and storage.
The postulated design basis accidents are either leaking of radioactive gases and airborne material from the production line or leaking of liquid waste from the storage tanks in the basement that are well shielded with thick concrete slabs, which are part of the ground floor.

From the individual risk in these tables, the maximum risk for the hypothetical group of children that lives permanently outside the OLP site is estimated to be less than 5 E-10 per year. For the (also) hypothetical group of adults that lives permanently outside the OLP site, the maximum individual risk is less than 4 E-11 per year.

1.5.2.3 Waste Storage facility

In the safety report and safety analysis report of the Waste Storage Facility (WSF) only two beyond design basis accidents are identified and the postulated scenarios have been analysed. These accidents and their radiological consequences are described in the safety analysis report and the risks are presented in Table 1-15. In this storage facility solid radioactive waste including waste from irradiated fissile material is packed in canisters that are stored in several vertical tubes, which are mounted in a concrete subterranean vault. In addition, large irradiated constructions (such as old HFR reactor vessel) and non-irradiated fissile materials are stored in two trenches. Both tubes and trenches are covered by thick concrete slabs. The postulated beyond design basis accident scenario is a destruction of the tubes and trenches and fire in the stored waste. The postulated releases are either airborne radionuclides from activated materials (^{60}Co), fission products (^{137}Cs) from irradiated fissile materials and irradiated fissile materials (^{239}Pu).
The maximum risk for the hypothetical group of children that lives permanently outside the OLP site is estimated (Table 1-15) and is less than 4 E-9 per year. For the hypothetical group of adults that lives permanently outside the OLP site, the maximum individual risk is less than 9 E-10 per year.

Description scenario	Release (TBq)	Prob. value (1/a)	95-percent child (mSv)	Conditional risk (child)*	Max. indiv. risk (child)
Damage to spent fuel canister stored in HCL storage basin	4.3 (Kr-85); 4.4 (Xe-133); 0.05 (Xe-135)	2.0 E-03	5.0 E-05	2.9 E-10	5.8 E-13
Fire in one of the HA hot-cells	4.3 (Kr-85); 0.13 (Ru-106); 0.18 (I-131); 4.4 (Xe-133); 0.05 (Xe-135); 0.09 (Cs-134); 0.07 (Cs-137); 0.02 (Ir-192)	1.0 E-05	9.0 E+00	9.6 E-05	9.6 E-10
***) Assumed the person is present at his location 24 hours per day.**					

Table 1-9: Design basis accidents HCL-RL (target group: children)

Description scenario	Release (TBq)	Prob. value (1/a)	95-percent adult (mSv)	Conditional risk (adult)*	Max. indiv. risk (adult)
Damage to spent fuel canister stored in HCL storage basin	4.3 (Kr-85); 4.4 (Xe-133); 0.05 (Xe-135)	2.0 E-03	3.0 E-05	2.4 E-11	4.8 E-14
Fire in one of the HA hot-cells	4.3 (Kr-85); 0.13 (Ru-106); 0.18 (I-131); 4.4 (Xe-133); 0.05 (Xe-135); 0.09 (Cs-134); 0.07 (Cs-137); 0.02 (Ir-192)	1.0 E-05	1.0 E+01	8.0 E-06	8.0 E-11
***) Assumed the person is present at his location 24 hours per day.**					

Table 1-10: Design basis accidents HCL-RL (target group: adults)

Description scenario	Release (TBq)	Maxim. prob. value (1/a)	Condit. risk (child)	Max. indiv. risk (child)	Condit. risk (adult)	Max. indiv. risk (adult)
Fire in one of the HA hot-cells +failure of filters	4.3 (Kr-85); 13 (Ru-106); 1.8 (I-131); 4.4 (Xe-133); 0.05 (Xe-135); 8.7 (Cs-134); 6.8 (Cs-137); 3.9 (Ce-144); 2.1 (Ir-192)	1.0 E-07	7.2 E-03	7.2 E-10	6.0 E-04	6.0 E-11
Plane crash with limited fire:	20 (Sr-90); 13 (Ru-106); 47 (I-131); 8.7 (Cs-134); 6.8 (Cs-137); 3.9 (Ce-144); 2.1 (Ir-192); 0.02 (Pu-241)	2.0 E-08	9.6E-02	1.9E-09	8.0 E-03	1.6 E-10
Plane crash with burning kerosene:	52 (Sr-90); 65 (Ru-106); 47 (I-131); 44 (Cs-134); 34 (Cs-137); 390 (Ce-144); 2100 (Ir-192); 0.08 (Pu-239); 22 (Pu-241)	2.0 E-08	1.2E-02	2.4E-10	1.0 E-03	2.0 E-11

Table 1-11: Beyond design basis accidents HCL-RL

Description scenario	Release (GBq)	Prob. value (1/a)	95-percent child (mSv)	Conditional risk (child)*	Max. indiv. risk (child)
large leak in the installation of one Mo-production train	8200 (Kr-85m); 3200 (Kr-88); 0.6 (Sr-89); 5.6 (Mo-99); 5.3 (Tc-99m); 1.8 (I-131); 25000 (Xe-133); 11000 (Xe-135); 0.1 (Ce-144); 2.0 (Pr-143); 1.0 (Nd-147)	3.0 E-02	8.0 E-02	5.4 E-07	1.62 E-08
large leak in one of the waste tanks in the basement of one Mo-production train	0.3 (Sr-89); 0.01 (Sr-90); 0.007 (Ru-106); 0.01 (Cs-134); 1.2 (Cs-137); 0.2 (Ce-144); 0.3 (Pr-143); 0.1 (Nd-147)	1.0 E-04	3.7 E+00	4.6 E-06	4.6 E-10
***) Assumed the person is present at his location 24 hours per day.**					

Table 1-12: Design basis accident in the HCL-MPF (target group: children)

Description scenario	Release (GBq)	Prob. value (1/a)	95-percent adult (mSv)	Conditional risk (adult)*	Max. indiv. risk (adult)
large leak in the installation of one Mo-production train	8200 (Kr-85m); 3200 (Kr-88); 0.6 (Sr-89); 5.6 (Mo-99); 5.3 (Tc-99m); 1.8 (I-131); 25000 (Xe-133); 11000 (Xe-135); 0.1 (Ce-144); 2.0 (Pr-143); 1.0 (Nd-147)	3.0 E-02	5.0 E-02	4.5 E-08	1.35 E-09
large leak in one of the waste tanks in the basement of one Mo-production train	0.3 (Sr-89); 0.01 (Sr-90); 0.007 (Ru-106); 0.01 (Cs-134; 1.2 (Cs-137); 0.2 (Ce-144); 0.3 (Pr-143); 0.1 (Nd-147)	1.0 E-04	3.6 E+00	3.8 E-07	3.8 E-11
***) Assumed the person is present at his location 24 hours per day.**					

Table 1-13: Design basis accident in the HCL-MPF (target group: adults)

Description scenario	Release (TBq)	Maxim. prob. value (1/a)	Condit. risk (child)	Max. indiv. risk (child)	Condit. risk (adult)	Max. indiv. risk (adult)
Plane crash with limited fire:	1.1 (Sr-89); 0.007 (Sr-90); 11 (Mo-99); 11 (Tc-99m); 0.03 (Ru-106); 3.7 (I-131); 0.008 (Cs-137); 0.25 (Ce-144); 4.1 (Pr-143); 2.1 (Nd-147)	2.0 E-08	4.3 E-03	8.6 E-11	3.6 E-04	7.2 E-12
Plane crash with burning kerosene:	1500 (Sr-89); 11 (Sr-90); 1100 (Mo-99); 3800(Tc-99m); 40 (Ru-106); 2800 (I-131); 0,03 (Cs-134); 13 (Cs-137); 380 (Ce-144); 4400 (Pr-143); 1700 (Nd-147)	2.0 E-08	2.2 E-02	4.4 E-10	1.8 E-03	3.6 E-11

Table 1-14: Beyond design basis accidents HCL-MPF

Description scenario	Release (TBq)	Maxim. prob. value (1/a)	Condit. risk (child)	Max. indiv. risk (child)	Condit. risk (adult)	Max. indiv. risk (adult)
Plane crash with limited fire:	3.3 (Co-60); 0.26 (Cs-137); 0.014 (Pu-239)	2.0 E-08	1.9 E-01	3.8 E-09	4.1 E-02	8.3 E-10
Plane crash with burning kerosene:	16.3 (Co-60); 1.3 (Cs-137); 0.025 (Pu-239)	2.0 E-08	2.2 E-03	4.4 E-11	2.5 E-04	5.1 E-12

Table 1-15: Beyond design basis accidents WSF

1.5.2.4 Decontamination and Waste Treatment (DWT) facility

In the safety report and safety analysis report of the Decontamination and Waste Treatment facility (DWT) for each of the three buildings of this facility design as well as beyond design basis accidents have been analysed. The radioactive inventories of these building are less than those of the other facilities. There are four types of waste handled at the DWT:

- NORM sludge from the oil & gas industry containing isotopes of radium and their daughter nuclides;
- Liquid waste and solid waste containing HFR resins, containing mainly Cobalt and cadmium isotopes;
- Waste from radioisotope productions, containing ^{99}Tc and ^{131}I;
- Liquid waste and solid waste containing depleted and enriched uranium;

There are two types of causes for accidents and releases in the DWT:

- Airborne radioactive material from evaporated liquids released after failure of basins and tanks with liquid waste;
- Airborne radioactive material from burned solid waste or from evaporated fluids heated by a fire.

Radioactive contaminated water from all nuclear facilities at the OLP is collected at the DWT. After treatment and monitoring, the cleaned waste water is released in batches via a long (double-walled) pipe into the North Sea. Failure of a single wall of this pipe is a design basis accident. Failure of booth walls is a beyond design basis accident.

The potential doses for children due to design basis accidents at the DWT are presented in Table 1-16, Table 1-18, Table 1-20 and Table 1-22. The potential doses for adults due to design basis accidents at the DWT are presented in Table 1-17, Table 1-19, Table 1-21 and Table 1-23.

The maximum risk for the hypothetical group of children that lives permanently outside the OLP site is estimated from Table 1-25, Table 1-26 and Table 1-27. The maximum individual risk for this hypothetical child is less than 5 E-8 per year as a result of beyond design basis accident at the solid-waste treatment building. These accidents cause also the maximum individual risk for the hypothetical adult, less than 2 E-8 per year.

1.5.2.5 Low Flux Reactor

In the safety report and safety analysis report of the Low Flux Reactor (LFR) only beyond design basis accidents have been analysed. The risks due to these accidents are presented in Table 1-28. The potential releases from these accidents consist of fission products such as I-131 and Cs-137. The Low Flux Reactor (LFR) is permanently shut down. In preparation of the decommissioning of this facility the fuel is removed from the reactor vessel and stored in store pits in the floor of the reactor hall.

However for the analysis, the fuel was considered to be present in the reactor. In addition, the (also stored) calibration sources, the neutron filter system containing bismuth and daughter products, and the convertor plate containing LFR fuel plates were included in the source term.

The maximum individual risk due to beyond design basis accidents for a hypothetical child living permanently near the OLP fence is less than 3 E-9 per year. For his hypothetical parents, the maximum individual risk is less than 2 E-10 per year (Table 1-28).

In conclusion, no full scope PSA has been performed for the nuclear facilities at the OLP. A semi deterministic approach has been chosen using a set of postulated initiating events. Frequencies are obtained from a list of general probabilities of failures of different type of equipment and systems. The frequency of beyond design basis accidents were either determined from multiplication of probabilities of subsequent failures or from the probability of an external event, such as a plane crash (statistics of crashes in the Netherlands of military planes and the total area of the Netherlands).

Based on the results presented it can be concluded that individual risk as well as societal risk resulting from the operation of the HFR at 50 MW with LEU fuel fully complies with the regulatory criteria.

Since the safety assessments have been performed during a period of more than 7 years, the last performed analyses (for the DWT) is more detailed than earlier analyses for facilities such as the LFR and WSF. In the process of 10-EVA a more detailed PIE analyses has been performed for the HCL.

Description scenario	Release (MBq)	Max. prob. Value (1/a)	95-percent child (mSv)	Conditional risk (child)*	Max. indiv. risk (child)
Fire in sludge-processing unit	1.68	0.04	2.26 E-01	1.08 E-05	4.34 E-07
Fire in decontamination hall (NORM)	0.34	0.04	4.67 E-02	2.24 E-06	8.97 E-08
Fire in decontamination hall (Nuclear Industry)	132.4	0.04	4.90 E-02	2.35 E-06	9.41 E-08
Fire in decontamination hall (Depleted U)	0.64	0.04	5.00 E-03	2.40 E-07	9.60 E-09
Fire in decontamination hall (Enriched U)	0.38	0.04	5.80 E-03	2.78 E-07	1.11 E-08
Fire flushing liquid supply: Depleted U (deco-cell)	6.4	0.04	5.00 E-02	2.40 E-06	9.60 E-08
Fire flushing liquid supply: Enriched U (deco-cell)	3.8	0.04	5.80 E-02	2.78 E-06	1.11 E-07
Fire at HFR-resin storage	636	0.04	2.35 E-01	1.13 E-05	4.51 E-07
***)Assumed the person is present at his location 24 hours per day**					

Table 1-16: Design basis accidents in DWT decontamination building (target group: children)

Description scenario	Release (MBq)	Max. prob. Value (1/a)	95-percent adult (mSv)	Conditional risk (adult)*	Max. indiv. risk (adult)
Fire in sludge-processing unit	1.68	0.04	1.52 E-01	3.04 E-06	1.22 E-07
Fire in decontamination hall (NORM)	0.34	0.04	3.09 E-02	6.18 E-07	2.47 E-08
Fire in decontamination hall (Nuclear Industry)	132.4	0.04	4.24 E-02	8.48 E-07	3.39 E-08
Fire in decontamination hall (Depleted U)	0.64	0.04	3.00 E-03	6.00 E-08	2.40 E-09
Fire in decontamination hall (Enriched U)	0.38	0.04	3.60 E-03	7.20 E-08	2.88 E-09
Fire flushing liquid supply: Depleted U (deco-cell)	6.4	0.04	3.00 E-02	6.00 E-07	2.40 E-08
Fire flushing liquid supply: Enriched U (deco-cell)	3.8	0.04	3.60 E-02	7.20 E-07	2.88 E-08
Fire at HFR-resin storage	636	0.04	2.04 E-01	4.08 E-06	1.63 E-07
***) Assumed the person is present at his location 24 hours per day.**					

Table 1-17: Design basis accidents in DWT decontamination building (target group: adults)

Description scenario	Release (MBq)	Max. prob. Value (1/a)	95-percent child (mSv)	Conditional risk (child)*	Max. indiv. risk (child)
Fire in waste storage room (metal vessels)	200	0.04	7.40 E-02	3.55 E-06	1.42 E-07
Fire in waste storage room (plastic vessels)	10.5	0.04	3.90 E-03	1.87 E-07	7.49 E-09
Fire at sludge dryer/storage unit (NORM)	0.0025	0.04	3.30 E-04	1.58 E-08	6.34 E-10
Fire at sludge dryer/storage unit (silt OLP)	0.0385	0.04	7.00 E-05	3.36 E-09	1.34 E-10
Fire at sludge dryer/storage unit (resin- U-mixture)	0.0578	0.04	4.30 E-04	2.06 E-08	8.26 E-10
Leakage after failure of silt basin	0.095	0.5	9.00 E-05	4.32 E-09	2.16 E-09
Leakage after failure of HFR-waste water basin	3.0	0.5	1.10 E-03	5.28 E-08	2.64 E-08
*) Assumed the person is present at his location 24 hours per day.					

Table 1-18: Design basis accidents in DWT Water-treatment building (target group: children)

Description scenario	Release (MBq)	Max. prob. Value (1/a)	95-percent adult (mSv)	Conditional risk (adult)*	Max. indiv. risk (adult)
Fire in waste storage room (metal vessels)	200	0.04	6.40 E-02	1.28 E-06	5.12 E-08
Fire in waste storage room (plastic vessels)	10.5	0.04	3.40 E-03	6.80 E-08	2.72 E-09
Fire at sludge dryer/storage unit (NORM)	0.0025	0.04	2.20 E-04	4.40 E-09	1.76 E-10
Fire at sludge dryer/storage unit (silt OLP)	0.0385	0.04	7.00 E-05	1.40 E-09	5.6 E-11
Fire at sludge dryer/storage unit (resin- U-mixture)	0.0578	0.04	3.10 E-04	6.20 E-09	2.48 E-10
Leakage after failure of silt basin	0.095	0.5	9.00 E-05	1.80 E-09	9 E-10
Leakage after failure of HFR-waste water basin	3.0	0.5	9.60 E-04	1.92 E-08	9.6 E-09
*) Assumed the person is present at his location 24 hours per day.					

Table 1-19: Design basis accidents in DWT Water-treatment building (target group: adults)

Description scenario	Release (MBq)	Max. prob. Value (1/a)	95-percent child (mSv)	Conditional risk (child)*	Max. indiv. risk (child)
Fire in tube cleaning facility (NORM)	0.035	0.04	4.70 E-03	2.26 E-07	9.02 E-09
Leakage of drum with HFR-resin	8	0.04	2.90 E-03	1.39 E-07	5.57 E-09
Fire at Tc-generator storage	20000	0.04	9.10 E-02	4.37 E-06	1.75 E-07
Fire at storage location of drums with HFR-waste	300	0.04	1.10 E-01	5.28 E-06	2.11 E-07
***) Assumed the person is present at his location 24 hours per day.**					

Table 1-20: Design basis accidents in DWT Solid-waste treatment building (target group: children)

Description scenario	Release (MBq)	Max. prob. Value (1/a)	95-percent adult (mSv)	Conditional risk (adult)*	Max. indiv. risk (adult)
Fire in tube cleaning facility (NORM)	0.035	0.04	3.20 E-03	6.40 E-08	2.56 E-09
Leakage of drum with HFR-resin	8	0.04	2.50 E-03	5.00 E-08	2.00 E-09
Fire at Tc-generator storage	20000	0.04	7.60 E-02	1.52 E-06	6.08 E-08
Fire at storage location of drums with HFR-waste	300	0.04	9.60 E-02	1.92 E-06	7.68 E-08
***) Assumed the person is present at his location 24 hours per day.**					

Table 1-21: Design basis accidents in DWT Solid-waste treatment building (target group: adults)

Description scenario	Release (MBq)	Max. prob. Value (1/a)	95-percent child (mSv)	Conditional risk (child)*	Max. indiv. risk (child)
Failure of drum with sludge (NORM)	0.07	0.5	9.50 E-03	4.56 E-07	2.28 E-07
Fire at drum with sludge (NORM)	0.77	0.04	1.04 E-01	4.99 E-06	2.00 E-07
Fire at stored components with NORM	1.03	0.04	1.39 E-01	6.67 E-06	2.67 E-07
Fire at stored components from the nuclear industry	66.2	0.04	2.50 E-02	1.20 E-06	4.80 E-08
Fire at stored components with Depleted U residues	0.32	0.04	2.50 E-03	1.20 E-07	4.80 E-09
Fire at components with Enriched U residues	0.19	0.04	2.90 E-03	1.39 E-07	5.57 E-09
Fire at filter train	20 + 0.1 (I-131)	0.04	1.90 E-02	9.12 E-07	3.65 E-08
*) Assumed the person is present at his location 24 hours per day.					

Table 1-22: Design basis accidents in DWT facilities outside (target group: children)

Description scenario	Release (MBq)	Max. prob. Value (1/a)	95-percent adult (mSv)	Conditional risk (adult)*	Max. indiv. risk (adult)
Failure of drum with sludge (NORM)	0.07	0.5	6.30 E-03	1.26 E-07	6.30 E-08
Fire at drum with sludge (NORM)	0.77	0.04	6.90 E-02	1.38 E-06	5.52 E-08
Fire at stored components with NORM	1.03	0.04	9.30 E-02	1.86 E-06	7.44 E-08
Fire at stored components from the nuclear industry	66.2	0.04	2.10 E-02	4.20 E-07	1.68 E-08
Fire at stored components with Depleted U residues	0.32	0.04	1.50 E-03	3.00 E-08	1.20 E-09
Fire at components with Enriched U residues	0.19	0.04	1.80 E-03	3.60 E-08	1.44 E-09
Fire at filter train	20 + 0.1 (I-131)	0.04	1.90 E-02	3.80 E-07	1.52 E-08
*) Assumed the person is present at his location 24 hours per day.					

Table 1-23: Design basis accidents in DWT facilities outside (target group: adults)

Description scenario	Release (MBq)	Maxim. prob. value (1/a)	Condit. risk (child)	Max. indiv. risk (child)	Condit. risk (adult)	Max. indiv. risk (adult)
Plane crash with limited fire:		3.00 E-08	1.30 E-06	3.90 E-14	4.20 E-07	1.26 E-14
Dispersed NORM-sludge material	0.28					
Dispersed silt containing uranium (depl. and 3%-enr.)	3.8					
Dispersed silt containing HFR-resins	318					
Plane crash with burning kerosene:		3.00 E-08	1.90 E-08	5.70 E-16	4.00 E-09	1.20 E-16
Dispersed NORM-sludge material	3.42					
Dispersed silt containing uranium (depl. and 3%-enr.)	3.8					
Dispersed silt containing HFR-resins	768					

Table 1-24: Beyond design basis accidents at the DWT decontamination building

Description scenario	Release (MBq)	Maxim. prob. value (1/a)	Condit. risk (child)	Max. indiv. risk (child)	Condit. risk (adult)	Max. indiv. risk (adult)
Fire in waste storage room (metal vessels) no filter	20000	4.00 E-04	4.00 E-05	1.60 E-08	1.40 E-05	5.60 E-09
Fire in waste storage room (plastic vessels) no filter	105 (I-131)	4.00 E-04	6.90 E-07	2.76 E-10	2.10 E-07	8.40 E-11
Fire at (NORM) sludge dryer unit and failed filter	0.3	4.00 E-04	1.90 E-07	7.60 E-11	5.50 E-08	2.20 E-11
Leakage after failure of silt basin and failed filter	9.5	1.00 E-02	4.80 E-08	4.80 E-10	1.90 E-08	1.90 E-10
Leakage after failure of buffer basin of HFR waste water	300	1.00 E-02	5.90 E-07	5.90 E-09	2.20 E-07	2.20 E-09
Plane crash with limited fire:		3.00 E-08	6.60 E-07	1.98 E-14	2.40 E-07	7.20 E-15
Dispersed NORM-sludge material	0.023					
Dispersed silt from water treatment	10					
Dispersed HFR-resins	300					
Plane crash with burning kerosene:		3.00 E-08	3.00 E-07	9.00 E-15	8.70 E-08	2.61 E-15
Dispersed NORM-sludge material	0.3					
Dispersed burning waste with I-131	420					
Dispersed HFR-resins	80300					
Dispersed silt from water treatment	105					

Table 1-25: Beyond design basis accidents in the DWT water-treatment building

Description scenario	Release (MBq)	Maxim. prob. value (1/a)	Condit. risk (child)	Max. indiv. risk (child)	Condit. risk (adult)	Max. indiv. risk (adult)
Fire in tube cleaning facility (NORM) and failed filter	3.52 E+00	4.00 E-04	2.70 E-06	1.08 E-09	7.60 E-07	3.04 E-10
Fire in storage of Tc-generators and failed filter	2.00 E+06	4.00 E-04	5.00 E-05	2.00 E-08	1.70E-05	6.80 E-09
Fire at stored drum with HFR-waste and failed filter	3.00 E+04	4.00 E-04	5.90 E-05	2.36 E-08	2.20 E-05	8.80 E-09
Plane crash with limited fire:		3.00 E-08	3.10 E-05	9.30 E-13	1.20 E-05	3.60 E-13
Dispersed heated silt from water treatment	3.20 E-01					
Dispersed heated HFR-resins	1.56 E+04					
Dispersed burned Tc-generators	2.00 E+03					
Plane crash with burning kerosene:		3.00 E-08	4.10 E-07	1.23 E-14	1.20E-07	3.60 E-15
Dispersed heated silt from water treatment	3.20 E+00					
Dispersed heated HFR-resins	9.12 E+04					
Dispersed burned Tc-generators	2.00 E+06					

Table 1-26: Beyond design basis accidents in the DWT Solid-waste treatment building

Description scenario	Release (MBq)	Maxim. prob. value (1/a)	Condit. risk (child)	Max. indiv. risk (child)	Condit. risk (adult)	Max. indiv. risk (adult)
Double-failure of pipe for releases into the sea	6.450	4.00 E-04	1.40 E-05	5.60 E-09	1.10 E-06	4.40 E-10
Plane crash with limited fire:		3.00 E-08	5.50 E-07	1.65 E-14	1.50 E-07	4.50 E-15
Fire at NORM-contaminated equipment and NORM waste	0.70					
Plane crash with burning kerosene:		3.00 E-08	4.50 E-08	1.35 E-15	9.60 E-09	2.88 E-16
Fire at NORM-contaminated equipment and NORM waste	17.9					
Dispersed heated HFR-resins	662					
Dispersed heated silt and burned waste with HF resins + uranium traces	5.1					

Table 1-27: Beyond design basis accidents in the DWT facilities outside the buildings

Description scenario	Release (TBq)	Maxim. prob. value (1/a)	Condit. risk (child)	Max. indiv. risk (child)	Condit. risk (adult)	Max. indiv. risk (adult)
Plane crash with limited fire:	0.22 (Cs-137); 2.2 (I-131). etc.	2.00 E-08	1.20 E-01	2.40 E-09	1.00 E-02	2.00 E-10
Plane crash with burning kerosene:	1.1 (Cs-137); 3.6 (I-131)	2.00 E-08	9.60 E-04	1.92 E-11	8.00 E-05	1.60 E-12

Table 1-28: Beyond design basis accidents in the LFR

2 Earthquakes

2.1 Design basis

2.1.1 Earthquake against which nuclear facilities on OLP are designed

The Petten research location (Onderzoekslocatie Petten, OLP) is located in the dunes on the North Sea ("Noordzee") coast in the northwest of The Netherlands. Until 1990 there was no hard evidence of any seismic activity in the region. In the early 1990's and 2001 small earthquakes were observed. Dutch institutes, responsible for the evaluation of earthquake data and collection of geological information of the underground, considered the earthquakes induced by the exploration of natural gas in the Bergermeer. In 1998, based on the US American code NRC 100 the German standard KTA 2201.2 and the collected seismic data for the region, the Design Basis Earthquake (DBE) for the HFR on the OLP site was defined.

Earthquakes are characterized by their magnitude on the Richter scale or by the intensity on the modified Mercalli Intensity scale (MMI) or the European Macro-seismic Scale (EMS). The magnitude on the Richter scale is a measure for the energy released by the earthquake. The intensity scales are based upon what people in the affected area feel and their observations of damage to structures around them. A rough comparison between the intensity (MMI) and the magnitude (Richter) for natural earthquakes is given in Table 2-1. The comparison is only valid at the epicentre of the earthquakes. The MMI and EMS scales are identical for the range of the Richter scale, up to MMI intensity IX.

2.1.1.1 Characteristics of the design basis earthquake (DBE)

In the late 1980's and early 1990's several small earthquakes occurred in the region near the OLP site. As a first estimate the Royal Meteorological Institute of the Netherlands (KNMI) concluded for the mentioned earthquakes the maximum acceleration at approximately 0.5 m/s^2 near the epicentre (magnitude <3). Based on new seismic related requirements, in 1994 the owner of the HFR decided to evaluate the safety of the HFR under seismic loadings. For the evaluation of the seismic loadings it is necessary to define the Design Basis Earthquake (DBE). The DBE for seismic response analysis of nuclear installations is defined as:

Intensity (MMI or MMS)	Observation	Magnitude (Richter)
I	Felt by very few people; barely noticeable.	1.0 to 2.0
II	Felt by a few people, especially on upper floors.	2.0 to 3.0
III	Noticeable indoors, especially on upper floors, but may not be recognized as an earthquake.	3.0 to 4.0
IV	Felt by many indoors, few outdoors. May feel like heavy truck passing by.	4.0
V	Felt by almost everyone, some people awakened. Small objects moved and trees and poles may shake.	4.0 to 5.0
VI	Felt by everyone. Difficult to stand. Some heavy furniture moved and some plaster falls. Chimneys may be slightly damaged.	5.0 to 6.0
VII	Slight to moderate damage in well built, ordinary structures. Considerable damage to poorly built structures. Some walls may fall.	6.0
VIII	Little damage in specially built structures. Considerable damage to ordinary buildings, severe damage to poorly built structures. Some walls collapse.	6.0 to 7.0
IX	Considerable damage to specially built structures, buildings shifted off foundations. Ground cracked noticeably. Wholesale destruction. Landslides.	7.0
X	Most masonry and frame structures and their foundations destroyed. Ground badly cracked. Landslides. Wholesale destruction.	7.0 to 8.0
XI	Total damage. Few, if any, structures standing. Bridges destroyed. Wide cracks in ground. Waves seen on ground.	8.0
XII	Total damage. Waves seen on ground. Objects thrown up into the air	8.0 or larger

Table 2-1: Comparison between MMI and Richter scale, valid at epicentre of natural earthquakes

‘The earthquake for which the safety systems of a nuclear reactor are designed to remain functional both during and after the earthquake, thus assuring the ability to shut down and maintain a safe configuration in the nuclear installation”

The geological formations of the underground of the OLP site are characterized from bottom to top by the following layers:

- << - 950 m Mesozoic rock layers;
- 950 m - 350 m Tertiary sediment (marine sand, clay and rock layers)
- 350 m - 25 m Quaternary sediment (shell rich sand, fluvial sand, clay layers);
- 25 m - 10 m Pleistocene sand deposits with clay and loam layers;
- 10 m - zero Holocene coastal dune sand layer.

Taken into account the seismic history and the geology of the OLP site and the region, the Rijks Geologische Dienst (RGD) characterizes the OLP site as a site with low seismic activity. For these sites the NRC 100 Code requires for the DBE that the maximum vibratory ground accelerations (PGA) at the foundation of the nuclear reactor shall be assumed to be at least 0.1 g (1.0 m/s^2). In 1994 the first DBE for the HFR was characterized by a horizontal PGA of 1 m/s^2 and a spectrum ranging from 0.1 Hz to 33 Hz. In accordance with common practice, the vertical PGA was fixed at 2/3 of the horizontal PGA.

In the period 1994 – 1998 new information with respect to seismic loadings on nuclear installations, research data and experimental data of natural and non-natural (natural gas induced) earthquakes, and the results of an EU coordinated seismic re-evaluation of the operating plants in Europe came available. Based on this information the current DBE for the HFR site was defined as a combination of natural earthquake and non-natural earthquake response spectra.

- Natural earthquakes: The HFR site is characterized as a site with low seismic activity and was compared with nuclear sites in northwest Germany. For these sites the earthquake response spectrum is based on the recommendations from NRC 100 and the recommendations and requirements from KTA 2201.2. Compared to the response spectrum, defined in NRC 100 for areas with low seismic activity, KTA 2201.2 adopted modifications in the low frequency range and the maximum acceleration amplitude was increased to 2 m/s^2. The response spectrum of DBE related to the "natural" earthquake for the HFR site is determined to be identical to the DBE defined for a characteristic nuclear power plant in northwest Germany. In Figure 2-1 the response spectra are presented for the horizontal and vertical PGA of the maximum natural earthquake on the HFR site.
- Non-natural earthquakes: In 1998 KNMI estimated for natural gas induced earthquakes at the HFR site the maximum horizontal PGA at 4 m/s^2 and the maximum vertical PGA at 1.0 m/s^2. The corresponding frequency spectrum ranged from 5 Hz to 30 Hz. This estimation of the maximum gas induced earthquake was confirmed in 2006. For the DBE, in line with common practice, the vertical PGA was set at 2/3 of the horizontal PGA. This response spectrum corresponds with an earthquake of intensity VI to VII on the MMI scale and a Richter magnitude of 3.9. Earthquakes induced by natural gas are characterized by generating a high PGA for a short time interval and a relatively low energy release. Therefore the relation between the Mercalli intensity and the Richter magnitude is different from the relationship indicated in Table 2-1. Gas induced earthquakes have a relatively high Mercalli intensity related to the Richter magnitude. Figure 2-1 presents, separately, the response spectra for the horizontal and vertical PGA.

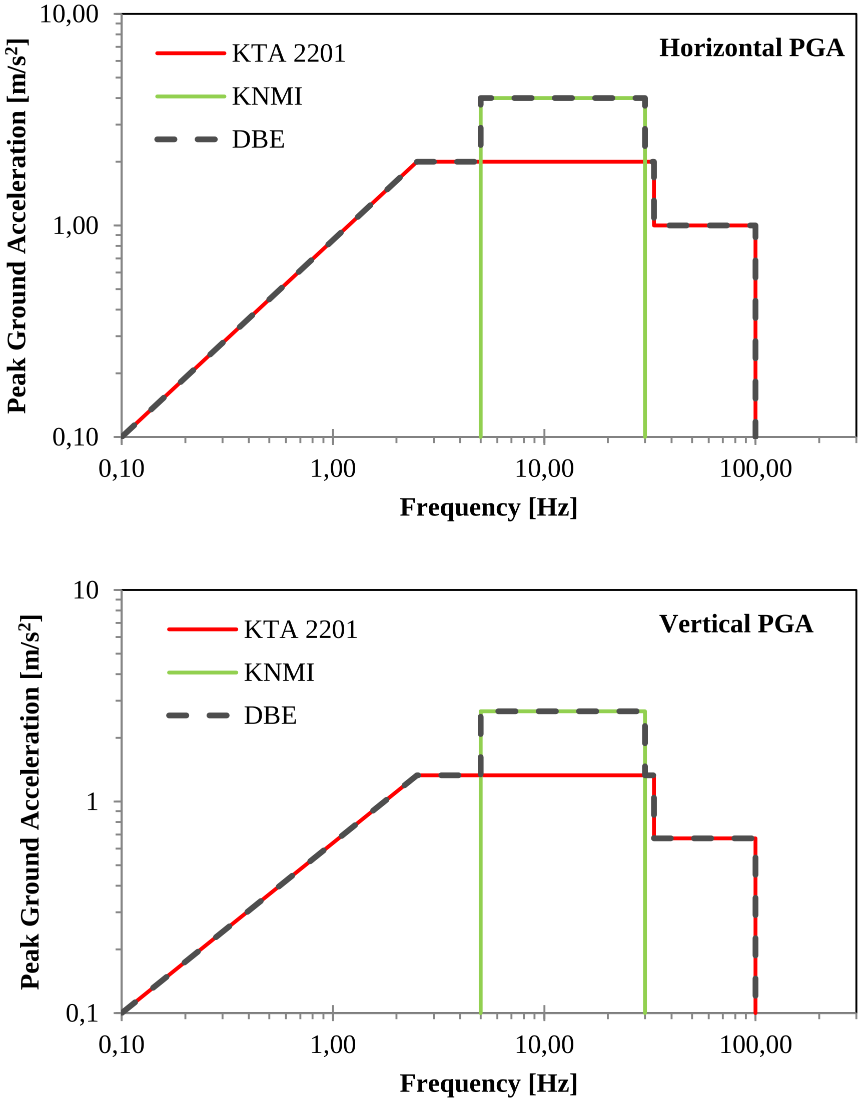


Figure 2-1: DBE spectrum for horizontal and vertical ground level acceleration

The DBE defined for the evaluation of the earthquake response of the HFR is the enveloping curve of the "natural" and "non-natural" earthquake spectra, characterized by the 5% damping earthquake response spectrum at foundation level. Licensing authorities agreed that the horizontal and vertical PGA have been cut off at 33 Hz because the HFR structure is practically not affected by frequencies between 33Hz and 100 Hz. However, for all the structural analyses, as a conservative approach, the full frequency spectrum from 0.1 to 100 Hz was included in the DBE. The DBE for the HFR and OLP is presented in Figure 2-1.

In 2003, based on the analysis of new experimental data, the HFR DBE was re-evaluated and it was concluded that the DBE defined in 1998 was still valid. In 2006 KNMI confirmed the earlier evaluation results of the gas induced seismic data.

The DBE is evaluated in particular for the HFR which is located on the OLP site. However, due to the limited size of the OLP site, the DBE defined for the HFR site is also applicable for all the nuclear installation on the OLP site.

2.1.1.2 Methodology to evaluate the design basis earthquake

OLP is located in the dunes on the North Sea coast in the northwest of The Netherlands. Seismicity of tectonic origin in The Netherlands is related to the Lower Rhine Graben and only observed in the southeast of The Netherlands. Until 1986 there was no hard evidence of any tectonic events in the north of The Netherlands. In 1986 a seismometer network was installed and the seismicity was recorded. All the observed seismicity after 1986 in the north of The Netherlands occurred at shallow depths (between 2500 m and 4000 m) corresponding to the depth of the natural gas reservoirs. Moreover, all events so far have been localised in or around the existing gas fields. Based on the experimental data, the geology and the history, KNMI concluded that all the seismic activity observed in the north of The Netherlands descends from gas induced earthquakes.

The DBE for the HFR is characterized by the enveloping curve of the maximum possible PGAs generated by the natural and non-natural earthquakes on the HFR site. For the HFR site, characterized as a low seismic area, according to NRC 100 the required maximum PGA for a natural earthquake should be set to 1.0 m/s^2. Lack of experimental data and to ensure a conservative approach, in 1998 it was decided to take the DBE, developed for a characteristic nuclear power plant in northwest Germany, as a basis for the natural earthquake at the HFR site.

In 1994, after the earthquakes in the Bergermeer area, KNMI started a large project to collect and evaluate data on gas induced earthquakes in the north of The Netherlands. In this area the main gas fields are the Groningen (one of the largest in the world) and the Bergermeer, Camperduin, and Roswinkel gas fields. The OLP site is located approximately 20 km north of the Bergermeer gas field and 5 km north of the Camperduin gas field. Due to the distance to other gas exploration locations, the seismic data from the Bergermeer and Camperduin gas fields are a basis for the seismic activity at the OLP site (Figure 2-2).

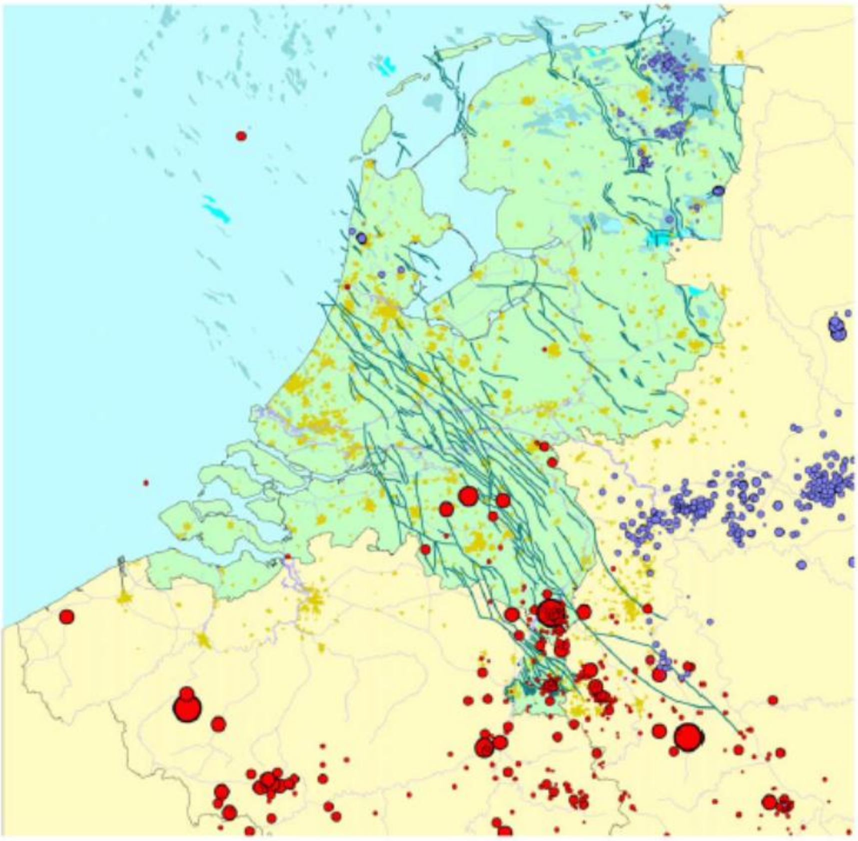


Figure 2-2: Observed earthquakes in The Netherlands

Table 2-2 presents the intensity and magnitude of the recorded earthquakes until 2005 in the Bergermeer and Camperduin gas fields. KNMI qualified the recorded earthquakes as gas induced earthquakes. This explains the relatively high intensity on the MMI scale compared to the magnitude of the maximum expected earthquake. Until 1998 the highest earthquake related intensity recorded in the Bergermeer (September 21, 1994) was approximately V on the MMI scale, caused by a natural gas induced earthquake with a magnitude of 3.2 on the Richter scale. Based on geological characteristics of the OLP site and experimental data from the Bergermeer and other gas fields, KNMI concluded in 1998 that the maximum possible gas induced earthquake in the Bergermeer is a gas induced earthquake with a maximum horizontal and vertical PGA of 4 m/s^2 respectively 1.25 m/s^2. The corresponding frequency spectrum was estimated at 5 Hz to 30 Hz. In accordance with common practice, the vertical PGA was increased to 2/3 of the horizontal PGA. This postulated natural gas induced earthquake corresponds to an earthquake with an intensity of VI to VII on the MMI scale and a Richter magnitude of 3.9. This maximum non-natural earthquake was included as the gas induced earthquake in the DBE for the HFR site.

Date	Magnitude (Richter)	Intensity (Mercalli)	Location
6/8/1994	3.0	4,5	Bergermeer
21/9/1994	3.2	5	Bergermeer
9/9/01	3.5	6	Bergermeer
10/9/01	3.2	5	Bergermeer

Table 2-2: Recorded gas induced earthquakes in the OLP site area

The DBE for the HFR is characterized by the enveloping curve of the maximum possible PGAs generated by an assumed natural and non-natural earthquake on the HFR site. Including in the enveloping curve are the previous described estimates of the maximum PGAs for natural and non-natural earthquakes, as shown in Figure 2-2. In 1998 it was agreed with representatives of the Dutch licensing authorities and the German institute GRS to define this as the DBE for the HFR site and consequently for the OLP site.

After 1998, the strongest earthquake (September 9, 2001) had an intensity of approximately VI on the MMI scale caused by an earthquake of 3.5 on the Richter scale. In 2003 it was concluded that, taken into account the new seismic data and recent geological investigations of natural gas induced earthquakes in the Netherlands, the DBE of 1998 is still valid for the HFR site.

The non-natural earthquake part of the DBE is developed based on experimental data at the epicentre of the earthquake. The distance between the OLP site and the Bergermeer gas field is approximately 20 km. The ground movement at locations near the epicentre depends on the distance to the epicentre and composition of the soil. Studies from KNMI show that at the OLP site the PGA is less than 5% of the PGA at the epicentre. It supports the assumption that the DBE covers the maximum earthquake that can occur at the OLP site.

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

The seismic activity on the OLP site is characterized by low natural earthquake activity and gas induced earthquake activity. The DBE for OLP site is defined as the enveloping curve of the horizontal en vertical PGA of the maximum natural and non-natural earthquake that is postulated to occur at the site. The probability of a natural earthquake in the region is very low, however, the gas induced earthquake in the region has an occurrence probability of approximately once per ten years. The response spectrum of the DBE is an adequate load that represents the most severe earthquake condition at the OLP 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 safe shutdown state and are most endangered during an earthquake Evaluation of their robustness in connection with DBE and assessment of potential safety margin.

This section identifies the SSCs that are required for achieving a safe shutdown state and are most endangered during an earthquake. Moreover, in connection with a DBE, the robustness of the SSCs is evaluated and the potential safety margins are assessed.

The nuclear facilities of the OLP site are described in Chapter 1.

In the next sections, separately for the HFR and the other nuclear facilities, the SSCs will be identified that are necessary for a safe shutdown, respectively safe situation after DBE at the OLP site.

High Flux Reactor (HFR)

The current safety philosophy behind the design of nuclear power plants against earthquakes did not exist at the time the HFR was designed and built (1958 – 1961). The overall safety philosophy was a conservative design with simple technical solutions and where possible makes use of natural mechanisms to control the system processes after incidents and accidents (inherently safe approach). During the lifetime of the reactor modifications, components and systems were implemented according to design requirements. The protection of the HFR against incidents and accidents, including earthquakes, is based on the following four levels of defence (defence in depth principle):

1. Prevention of incidents through the conservative reactor design, quality assurance and a well-considered plant operation.
2. Detection and correction of plant operation parameters before the system parameters exceed the operation limits.
3. Accident management through the safety systems. Before system parameters exceed the safety limits the safety systems will take corrective actions to control accident.
4. Mitigation of the effects of an accident for the public and the environment.

The defence in depth concept for the HFR, as defined in the HFR Safety Report, combines within the third level "accident management" the fourth level (DBA) and fifth level (beyond DBA) as defined in the IAEA Safety Standard on research reactors.

Based on this approach the following three fundamental safety functions of the reactor are guaranteed:

1. Safe shutdown of the reactor and maintain sub-criticality.
2. Removal of decay heat from the core and from spent fuel.
3. Confinement of radioactivity, control of radioactive releases and shielding against radiation.

The protection of the HFR against the consequences of an earthquake is ensured by satisfying the fundamental safety functions at all levels of defence.

First level of defence prevention. The protection of the reactor against the consequences of the DBE starts with a proper site selection and an engineered design of the SSCs. As mentioned, the OLP site is characterized by very low natural seismic activity. There is no historical evidence of tectonic earthquakes and the magnitude of non-natural earthquakes is small, with a relative low energy release. The buildings of reinforced concrete and steel structures satisfy the Dutch building standards. With the finite element program ANSYS, the displacement, stresses, and forces of the different structures were determined and assessed against the Dutch building code TGB 1990. The (dynamic) analyses showed that after the DBE relevant HFR buildings are not expected to lose their function following the DBE. More details about the structural analysis including seismic walk-downs with NRG experts are presented under the third level of defence.

Regarding the mechanical, electrical and instrumentation and control (I&C) systems, including cabling, are not expected to lose their function following DBE. Worldwide industrial earthquake response experience (among others the Albstadt earthquake in Germany in 1978) showed in general little damage to steel and reinforced concrete structures and mechanical equipment. Damage to electrical and I&C systems occurred where unrestricted displacements were possible or where equipment was damaged by debris from unqualified structures. Similar results were obtained in studies of earthquakes with PGAs up to 0.5g in the Pacific region (USA and Japan). An IAEA study that summarizes various studies of earthquake response concluded that piping systems survive earthquakes very well. In 1998, during a seismic walk-down, in particular the safety relevant instrumentation and electrical cabinets, including connecting lines, were observed. The findings of the walk-down are documented and implemented.

Second level of defence detection and correction. As a consequence of the previous, the detection and corrective actions, necessary to keep the reactor within its operation limits, are assured after the DBE. In case the DBE causes that detection and corrective actions do not work properly, accident management actions are the next step.

Third level of defence accident management. The design of the HFR has provisions that control the consequences of design basis accidents. The initiating events for the postulated accidents are an internationally agreed list of internal and external events, including earthquakes. After an earthquake a proper operation of the safety systems relies on seismic qualification, while conservatively it is assumed that SSCs, which are not designed for DBE loads, fail. The seismic qualification of the SSCs under DBE loads has been established by design, analysis and walk-downs etc. The defence against design basis accidents relies on different diverse and redundant systems. For the HFR the following functions have been identified that are necessary to satisfy the three fundamental safety functions.

1. **Safe shutdown of the reactor and maintain sub-criticality.** Emergency shutdown (scram) of the HFR is initiated by the reactor protection system when specific system parameter set points are exceeded. Scram is ensured by the fast insertion of 6 neutron absorbing control rods into the core. The long term sub-criticality is ensured by the manual action to move a neutron absorber (alternative shutdown system) towards the reactor core, and the use of the ~40 h time interval to fully insert control rods and avoid (possibly dangerous) recriticality. The position of the experiments and the molybdenum production facilities do not change after DBE and the sub-criticality of the fissile material in the experiments is maintained after shutdown. More details about the sub-criticality are presented in Section 6.3.4 of this report (Prevention of re-criticality).
2. **Removal of heat from core and from spent fuel**. Decay heat removal from the core after a design basis accident is assured by the following cooling modes, as described in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink):
   - The decay heat removal system. This cooling mode relies on integrity of the primary system pressure boundary, availability of power supply including emergency diesel generators and batteries, and integrity of the pool. Availability of the decay heat removal system provides circulation through the primary system and complete heat up. The secondary cooling is not seismic qualified and therefore assumed to be not available. Without secondary cooling decay heat removal from the core is possible for about 12 hours. Hereafter the next cooling mode is needed.
   - Through manually opening of the convection valves on the reactor vessel natural circulation is established in the core region and the decay heat is transferred to the water in the pool. This cooling mode relies on the operability of the convection valves, integrity of the pool and the availability of coolant in the pool. The heat capacity of the available coolant in the pool and the evaporation of coolant from the pool is sufficient to ensure long term (700 hours until core uncovery) decay heat removal from the core.
   - If the former modes fail part of the primary coolant heats up and boiling starts. The steam generated in the core region is released through expansion vessel. After about 8 hours core heat-up starts.

The long term (700 hours) cooling capacity of the cooling modes is ultimately ensured by the heat capacity and the evaporation of the water in the pool. The possibility to supply additional water to the pool requires availability of the “hydrofoor” emergency pool supply system, including the water make up building (“reinwaterkelder”). The ultimate water resource is the water in the KOTS storage next to the reactor building. A pump is available to supply water from the KOTS into the pool. Details about the cooling modes including available time and resources are addressed in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink). However, the supply of water from the water make up building (“reinwaterkelder”) as well as the supply of water from the KOTS to the reactor building are not seismic qualified.

Cooling of the molybdenum production facilities in the HFR is established by separate cooling systems. The pumps in the circuits may fail because they are not seismic qualified. Decay heat from the facilities is removed by boiling on the surface of the targets and heat transfer to the pool. Only the fuel of the Mykonos irradiation facility may become damaged. The spent fuel racks are stored in the pool and long term cooling (600 hours until spent fuel uncovery) is provided by the water in the pool.

3. **Containment isolation.** Containment isolation is required in case of failures that results in release of activity into the reactor building. In case the radioactivity in the exhaust line of the ventilation system exceeds the safety limits containment isolation is initiated by the reactor protection system. The reactor confinement is a leak tight steel cylinder with isolation valves, material and personnel locks. Included is a ventilation system to keep releases of radioactivity within the safety limits. The ventilation system maintains a little under-pressure in the containment. The ventilation system turns to recirculation mode when under-pressure cannot be maintained. Until 0.5 bar over-pressure the containment is leak tight[2]. The maximum over-pressure after a design basis accident remains well below 0.5 bar. Containment isolation remains functional after the DBE, even in case of an increased containment pressure.

Currently for a design of a nuclear power plant all SSCs are grouped in three or more categories in terms of their importance to safety during and after an earthquake. All SSCs of the nuclear plant should be assigned to one of the following categories:

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[2] To satisfy the national requirements the accepted leak rate of the reactor building is less than 0.1% per 24 hours at 0.5 bar over-pressure. The requirement is ensured by periodically testing, inspection and maintenance.

Category 1: All SSCs required to withstand the consequences of ground motions associated with DBE. It covers all SSCs that are important to safety and includes the following SSCs:

- SSCs whose failure could directly or indirectly cause accident conditions as a consequence of DBE;
- SSCs required for shutting down the reactor, maintaining the reactor in a shutdown condition, removing residual heat, and monitoring parameters essential to these functions;
- SSCs required to prevent or mitigate radioactive releases exceeding the permissible limits in the event of any design basis accident.

Category 2: This category includes all SSCs whose failure after DBE have an impact on the SSCs included under seismic category 1.

Category 3: This category includes all SSCs that are not related to the reactor and that could pose a radiological hazard (e.g. spent fuel and the radioactive waste building).

Category 4: All SSCs that are not included in category 1, 2 or 3.

At the time the HFR was commissioned the seismic categorization was not required because the installation was located in a low seismic area. In 1998, based on new safety requirements related to seismic events, an extensive project started to evaluate the earthquake response of buildings and SSCs of the HFR against the DBE. The main objective of the project was to determine the maximum response of the installation due to seismic and static loadings and show that the reactor can be operated without undue risk to the public after a postulated DBE. The project had two main activities:

- Seismic walk-down of the facility;
- Seismic response analyses of selected buildings and SSCs.

Seismic walk-down: In the seismic walk-down exercise the safety relevant instrumentation, electrical cabinets, the cables of the emergency power, the shutdown instrumentation, and the hydraulics for the control of the convection valves were considered. The findings and recommendations of the walk-down were documented and implemented. A separate seismic walk-down of the emergency power supply building confirmed the availability of at least one of the diesel generators and the integrity of the building after the DBE.

Seismic response analyses: The earthquake, assumed for the analytical evaluation of the seismic response of the HFR SSCs, is defined by the DBE graphically presented in Figure 2-1. Included in the seismic evaluation process are the following SSCs:

- Steel dome structure and the safety relevant components;

- Stack of air treatment building;
- Concrete substructures of the reactor building;
- Primary pump building including safety relevant SSCs;
- Primary cooling system;
- Connecting duct between primary pump building and reactor building;
- Reactor vessel and internals.

Based on drawings, material, material properties, and expert judgment finite element models were developed for the considered structures. With the finite element program ANSYS, the displacement, and stresses and forces of the different structures were determined during and after the DBE. The analysis results for components and systems were assessed against ASME III requirements and the building analysis were assessed against the Dutch building code TGB 1990. The analyses showed that after DBE all considered SSCs remained functional and assured that the fundamental safety function could be satisfied. After the results of the analysis were documented, based on analysis results and new seismic information, the following analyses were re-evaluated and new analyses were performed.

- Based on the first analysis a steel frame was installed around the lead/basalt shielding walls in the basement of the reactor building to reinforce the walls and prevent collapse during an earthquake.
- Based on new information, the stack of the air treatment building was re-analysed. In 2004, although not required, the concrete pile group of the stack was reinforced.
- In 2003 the seismic response of connection duct between the reactor building and the primary pump building (support girders and cabling) on the DBE was evaluated. It was concluded that the considered structures remain functional.
- In 2005 new pumps were installed for the pool cooling system. Seismic evaluation shows that the pumps and the connecting pipes will remain functional after the DBE.
- In 2004 the reactor vessel and its internals have been analysed, with respect to seismic loadings after the DBE, for high enriched fuel as well as low enriched fuel. The analysis showed that the control rods remain functional during and after the DBE. The reactor vessel and its internals remain functional after DBE because the stresses in the vessel and its internals are predominantly determined by thermal loading.

Safety function / safety system	Seismic qualified			Remarks/availability/operability
	yes	no	category	
Safe Shutdown				
Shutdown systems • **Scram** • **Alternative shutdown**	 √ √		 1 1	 Manual operation from reactor building
Heat Removal				
Primary systems boundary	√		1	
Pools • **Pool cooling system** ○ **Capacity** ○ **Boundary** • **Boundary** • **Hot cell panels** • **Spent fuel** • **Supply of emergency water**	 √ √ √ √ √ 	 √	 1 1 1 2 3 1	 Integrity Integrity Supply of water from the water make up building ("reinwaterkelder") and KOTS are not included in the seismic qualification analyses
Decay heat removal system • **Cooling capacity** • **Boundary**	 √ √		 1 1	 Integrity
Convection valves	√		1	Manual from reactor building
Pool injection valves	√		1	Manual from control room/reactor building
Secondary cooling system		√	4	Secondary water after DBE not available
Confinement of radioactivity				
Ventilation system	√		1	
Off gas system	√		1	
Containment • **Steel dome** • **Isolation valves** • **Pool bridge, crane**	 √ √ √		 1 1 2	 Stability of containment isolation not included in analyses
Stack	√		2	
Other				
Emergency power supply • **Diesel generators** • **Battery back-up**	 √ √		 1 1	 Available for 33 hours
Reactor protection system	√		1	Building is not seismicly qualified
External power supply		√	2	Availability is not ensured
Control room		√	1	

Table 2-3: Seismic qualification of the HFR safety systems

Table 2-3 gives an overview of the seismic classification of the safety (related) systems of the HFR within the relevant safety function. Included in Table 2-3 is also information about the availability, integrity and operability of the different systems. The decay heat removal system and the pool cooling system are safety systems necessary for accident management. The two systems circulate coolant and heat is removed from the primary system and the pool in the heat exchangers (transfer to the secondary system). However, heat transfer of the two systems to the secondary system is not required to fulfil the safety function of the systems. The heat capacity of the coolant in the primary system and the pool is sufficient to keep the core cooled for the long term (700 hours until core uncovery, Chapter 5 of this report). This explains the qualification seismic category 4 for the cooling function of the decay heat removal and the decay heat removal system. Based on the results of the walk-down and analysis there are some issues that require attention:

- In case of DBE in the area the availability of external power supply is not ensured. Loss of this function results in transfer to emergency power. This source is available for 33 hours based on the diesel fuel storage inside the emergency power building. However, efficient use of the diesel generators can increase this time significantly.
- In case of a DBE the availability of cooling water from the Noordhollands Kanaal cannot be ensured. However, the secondary cooling function of the decay heat removal system and the pool cooling system is not required during the first 600-700 hours since other cooling modes are available as mentioned earlier.
- In case of DBE in the area the availability of water for the “hydrofoor” emergency pool water supply system cannot be ensured. Although the water in the pool is sufficient for long term (about 600-700 hours) cooling of the core (Chapter 5 of this report) and make-up of pool water is not ensured.
- The alternative shutdown system and the actuation of the convection valves on the reactor vessel are based on manual actions inside the reactor building.
- Actuation of the pool injection valves is based on a manual action from the control room or reactor building.

Fourth level of defence mitigation. This level of defence is addressed in the emergency plans which are in place for the HFR, the OLP site and the region. These plans are categorized as follows:

1. Internal emergency measures (bedrijfsvoorschriften en noodprocedures);
2. Emergency measure NRG (Bedrijfsnoodplan NRG);
3. Emergency measure OLP (Intern Noodplan Onderzoek Locatie Petten);
4. Disaster Emergency Plan region Zijpe (Rampenbestrijdingsplan Gemeente Zijpe en Duinbedrijven OLP);

5. National emergency plan nuclear accidents (Nationaal Plan Kernongevallenbestrijding).

The first two documents describe the measures to respond to incidents, accidents, design basis accidents and beyond design basis accidents. The objective is to keep the reactor within its operation limits, and if that is not possible, to keep the reactor in a safe shutdown state and prevent release of radioactivity to the environment. In case confinement of radioactivity inside the reactor building fails, the next emergency plans become effective. The other three documents are general site, regional and national emergency plans to protect the public and environment against the consequences of a nuclear disaster. Education and training programs for the emergency plans at different levels (operators, public and authorities) are organized regularly. Evaluation of the training and education programs results in an emergency preparedness of the public and improvement of the different emergency plans.

Other nuclear facilities on OLP site

The other facilities on the OLP site are radiological laboratories, an isotope production facility, and storage and preparation of nuclear waste. The safety measures in the facilities are based on well-designed SSCs, which satisfy the requirements of codes and regulations. In none of the facilities energy production based on a sustained fission reaction takes place. The ventilation and air conditioning systems in the buildings are active systems, the other safety systems (barriers) are passive systems with the objective to confine the radioactive material.

The protection of the nuclear facilities is based on the following four levels of defence:

1. Prevention of incidents through regulations and good workmanship;
2. Detection and correction if system parameters exceed the normal operation limits;
3. Accident management if system parameters exceed safety limits;
4. Mitigation of the consequences of an accident.

The three fundamental safety functions that are defined for nuclear reactors are basically also valid for the facilities on the OLP site.

1. Ensure sub-criticality of the nuclear material;
2. Removal of heat from nuclear material;
3. Confinement of radioactivity, control of radioactive releases and shielding against radiation.

Sub-criticality of the fissionable material of the nuclear facilities on the OLP site is ensured by design of the storage facilities, technical specification for operation and work procedures and instructions. The safeguard officer of NRG is in charge of the fission product administration. The correct implementation of the requirements for working with and storage of fission material is the responsibility of the Reactor Manager and Location Manager. The construction of the nuclear facilities and the location of the nuclear materials prevent a serious relocation of the nuclear material after DBE. The mentioned measures make that for the nuclear facilities after DBE sub-criticality is ensured. The first fundamental safety function is therefore satisfied for all nuclear facilities on the OLP site. More details about the sub-criticality of the fissionable material inside the nuclear facilities are given in Section 6.3.4 of this report (Prevention of re-criticality).

The nature of the nuclear material present in the different buildings is such that no significant heat is generated in the material. Natural convection inside the buildings and structures is sufficient to remove any generated heat from the nuclear material. Removal of decay heat from the nuclear material is not safety relevant for the nuclear facilities and the second fundamental safety function is satisfied for the nuclear facilities on the OLP site.

For the nuclear facilities on the OLP site, with exception of the HFR, the first and second fundamental safety functions are satisfied, the third fundamental safety (confinement of radioactivity, controlled release of radioactivity and shielding against radiation) will be addressed below.

The first and second level of defence (prevention, detection, correction) and the fourth level of defence (mitigation) are identical for all nuclear facilities. The third level of defence (accident management) will be addressed separately for the selected nuclear facilities.

First level of defence prevention. This level of defence is identical for all the nuclear facilities. Due to the proper site selection (low seismic active area) and the engineered design the SSCs are expected to remain functional after DBE. This level of defence is supported by the evaluation of the seismic response of the selected SSCs of the facilities. This evaluation has two main activities:

- Seismic walk-down of the facilities;
- Analytical approach.

Seismic walk-downs are a requirement of the 10-yearly safety evaluation of the OLP site. The objective of the seismic walk-down is to check if the safety relevant SSCs comply with the seismic related safety requirements. The findings are documented and implemented. During seismic walk-downs in 1999 and 2010 in particular the following details were verified:

- Construction of the buildings, cranes, crane supports etc.;
- Routing of electrical cables and ventilation systems;
- Possible damage by falling objects;

- General judgment of the location and storage of nuclear material.

The analytical approach started with an analysis based on simple boundary conditions and assumptions. Finite element analyses were introduced where simple analyses were inconclusive. More details about the walk-downs and the analytical evaluation are included in the section where the third level of defence is addressed.

Additional to the walk-downs and the analysis, experience and evaluation of natural earthquakes show that the mechanical, electrical and control (I&C) systems in buildings and installations are not expected to lose their functionality after the DBE. Worldwide industrial earthquake response experience (among others the Albstadt earthquake in Germany in 1978) showed in general little damage to steel and reinforced concrete structures and mechanical equipment. Damage to electrical and I&C systems occurred where unrestricted displacements were possible or where equipment was damaged by debris from unqualified structures. Similar results were obtained in studies of earthquakes with PGAs up to 0.5g in the Pacific region (USA and Japan). An IAEA study that summarizes various studies of earthquake response concluded that piping systems survive earthquakes very well.

In conclusion, the SSCs of the nuclear facilities are expected to remain functional after the DBE on the OLP site.

Second level of defence detection and correction: For all activities inside the nuclear facilities on the OLP site work instructions and control mechanism are in place to detect and correct deviations from normal operation. In case, after the DBE, corrections fail accident management actions are the next process step.

Third level of defence accident management: The different nuclear facilities on the OLP site have provisions to prevent incidents and accidents. The remaining fundamental safety function valid for the nuclear facilities on the OLP site is:

- Confinement of radioactive material and limit and control radioactive release.

Under normal operation conditions the release of radioactivity is controlled by ventilation and air conditioning systems. These systems function as an active barrier against uncontrolled release of radioactivity. However, although the ventilation systems are connected to the emergency grid on the OLP site, these ventilation systems are not seismic qualified.

The remaining safety function is based on the integrity of the different barriers that prevent release of radioactive material into the environment. After an accident a proper functioning of the barriers prevents violation of the fundamental safety function. This means that after an earthquake the functionality of the barriers must be ensured by a proper seismic qualification.

For the different nuclear facilities the following barriers have been identified that are necessary to satisfy the fundamental safety function:

Low Flux Reactor (LFR)

The LFR stopped operation in 2010. The fuel is removed from the reactor and, together with the lifetime spent fuel, stored in vertical pipes in the floor of the building. A seismic walk-down of the LFR building took place 2000. In 2000 additional structural analysis of the LFR building and the LFR vessel support and fuel storage showed that after DBE the vessel and fuel storage structures inside the building remain functional.

Confinement of the radioactivity after the DBE is ensured by the following barriers:

- Cladding of the fuel;
- Vertical steel pipes integrated in the concrete floor including plug (seismic qualified);
- LFR building. Ventilation valves close in case of loss of power.

Decontamination and Waste Treatment facility (DWT)

The decontamination and treatment of nuclear waste takes place in three buildings on the OLP site. The main activities are (1) decontamination of equipment, (2) treatment and storage of water from the other nuclear facilities, (3) treatment and storage of resin filters and containers from the isotope production, and (4) laundry. All buildings of the DWT have a ventilation system, which maintains under-pressure inside the buildings. The ventilation system is connected to the emergency power grid on the OLP site.

Waste water is stored and treated in large tanks and basins. The basins are an integral part of the building and remain functional after DBE. The confinement of radioactivity after the DBE in the Water Treatment Building is ensured by the following barriers (latter is an active system, the other two are passive barriers):

- Tank and pool walls;
- Water Treatment Building: ventilation valves close in case of loss of power;
- Under-pressure in the building.

The resin filters are stored in sealed canisters in trenches and the isotope containers are prepared and stored in racks inside the building. The trenches are covered by concrete slabs. The confinement of radioactivity after DBE in the Waste Storage Building is ensured by the following barriers (latter is an active system, the other three are passive barriers):

- Canisters and containers;
- Concrete slabs on top of the trenches;
- DWT Waste Storage Building: ventilation valves close in case of loss of power;
- Under-pressure in the building.

The waste removed from drilling pipes is temporarily stored in stainless steel containers inside the building. After the DBE the confinement is ensured by the cleaning installation. The containment of radioactivity is ensured by the following barriers (latter is an active system, the other two are passive barriers):

- Stainless steel containers;
- Waste Treatment Building: ventilation valves close in case of loss of power;
- Under-pressure in the building.

From the laundry building the water is directly transferred to the pools inside the Water Treatment Building.

<u>Waste Storage facility (WSF):</u>

The findings of seismic walk-downs in 2000 and 2010 are documented and implemented. Additional to the walk-downs structural analysis of the fibre cement-concrete storage pipes and the covers on top of the trenches are performed. The analyses confirm that the functionality of the pipes and covers is maintained after DBE.

The nuclear waste is stored in steel vessels in cellars below floor level. The outer barrier against release of radioactivity is the ventilation system that keeps under-pressure in the storage cellars under the floor of the building. The system is connected to the emergency power grid on the OLP site. The confinement of radioactivity after DBE is ensured by the following barriers:

- Storage containers (not always complete barrier as some containers have deteriorated);
- Eternite-cement vertical storage pipes (seismic qualified category 1);
- Beams above the storage pipes, concrete slabs above the trenches (seismic qualified category 1);
- WSF Waste Storage building: ventilation valves close in case of loss of power;
- Under pressure in the under floor cellars (this is an active barrier).

Jaap Goedkoop Laboratory (JGL):

JGL is a radiological laboratory (level B laboratory) for a wide range of radiological activities (isotope preparation, neutron measurements etc.). Small amounts of nuclear material are treated and handled in cells. A ventilation system maintains under-pressure inside the building. The ventilation system is connected to the emergency power grid of the OLP site. Confinement of radioactivity is ensured by the following barriers (latter is an active system, the other two are passive barriers):

- Containers of the nuclear material.
- Jaap Goedkoop laboratory building: ventilation valves close in case of loss of power.
- Under-pressure in the building.

Snel Thermisch Experiment Krito facility (STEK):

Until 1980, the STEK building was the location of an experimental zero power reactor. After decommissioning of the reactor the building has been used as temporarily storage for low, medium, and high level radioactive waste which is ready for transportation to the national nuclear waste disposal COVRA. Moreover neutron sources are stored in the building. A ventilation system in the building controls the release of radioactivity within the required limits. Confinement of radioactivity is ensured by the following barriers (latter is an active system, the other two are passive barriers):

- Containers of the nuclear waste;
- Transport canisters;
- LFR building: ventilation valves close in case of loss of power.

Hot Cell Laboratory (HCL):

The HCL facility has two main buildings:

- HCL laboratory, research laboratory;
- MPF, isotope production facility.

In this section only the HCL radiological research laboratory will be considered. The MPF is considered separately.

In the framework of the 10-yearly safety evaluation of the nuclear facilities on the OLP site, in 2000 and 2010 seismic walk-downs of the HCL laboratory took place. The results of the walk-downs are documented and implemented.

Based on the experience with the seismic analysis of the stack of the HFR a finite element analysis was performed for the HCL stack. Re-enforcement of the foundation of HCL stack was recommended and implemented.

Uranium filters from the isotope production process in the HFR and fissile materials from the HFR are stored in the pool of the HCL. The main function of the water of the HCL pool is radiation protection. The proper construction of the racks and the canisters was reviewed in the seismic walk-downs of the HCL facility.

Under normal operation conditions, under-pressure is maintained inside the HCL laboratory by the ventilation and air conditioning system. A second separate ventilation system maintains under-pressure in the hot cells. The ventilation systems are connected to the emergency power grid of the OLP site. This is the outer barrier to prevent release of radioactivity after the DBE. The following barriers ensure also confinement of radioactivity:

- Container with the nuclear material;
- Hot cells and the structure of the location where the nuclear material is placed;
- Under-pressure in hot cells (active system);
- HCL building (seismic qualified by walk-down): ventilation valves close in case of loss of power;
- Under-pressure in building (active system);

Molybdenum Production Facility (MPF):

A first evaluation of the earthquake response of the MPF facility, based on NRC100 requirements, was performed prior to the construction of the facility in 1994. In 2000, in the framework of the 10-yearly safety evaluation, the MPF was re-evaluated against the DBE for the OLP site. The findings of the seismic walk-down of the MPF facility were documented and implemented. Next to the seismic walk-down also the seismic response of the building and the stability of the lead cells were analysed. It was concluded that the seismic safety margins of the building were sufficient and the stability of the lead cells was ensured after DBE.

Recently the floor/ceiling between the first and second floor of the MPF building showed little cracks. The damage to the structure is caused by a onetime overload. Analysis showed also that currently the floor is loaded at its maximum. Re-enforcement of the floor will take place in the near future (2012-2013).

Under normal operation conditions, under-pressure is maintained inside the MPF by the ventilation and air conditioning system. A second system maintains under-pressure in the cells of the isotope production process. The ventilation systems are connected to the emergency power grid of the OLP site. The following barriers ensure confinement of radioactivity:

- Container of the nuclear material;
- Structures of cells where nuclear material is placed (seismic qualified);
- Under-pressure in the cells (active system);
- HCL building (seismic qualified category 1: ventilation valves close in case of loss of power;
- Under-pressure in the building (active system);

During design and commissioning of the facilities on the OLP site there was no formal seismic qualification of the different SSCs. According to the SSCs of the facilities should be classified as category 3. This category must have safety margins consistent with their potential for radiological consequences. For the facilities on the OLP site, the seismic qualification of the SSCs in category 3 has been established by design or by assuring continued performance under DBE loads confirmed analysis and seismic walk-downs. The seismic evaluation of the nuclear facilities on the OLP site showed that after DBE sufficient seismic qualified barriers are in place to satisfy the fundamental safety function that radioactive material is confined inside the nuclear facilities on the OLP site.

Fourth level of defence mitigation: This level of defence is covered by severe accident management guidelines and the emergency plans for the location, the site, the region, and on national level. These plans are categorized as follows:

1. Internal emergency procedures (“bedrijfsvoorschriften en noodprocedures”).
2. Emergency plan NRG (“Bedrijfsnoodplan NRG”).
3. Emergency plan OLP (“Intern Noodplan Onderzoek Locatie Petten”).
4. Regional disaster emergency plan Zijpe (“Rampenbestrijdingsplan Gemeente Zijpe en Duinbedrijven OLP”).
5. National plan for the Control of Nuclear Accidents (“Nationaal Plan Kernongevallenbestrijding”).

The first two documents contain the measures to respond to incidents, accidents, design basis accidents and beyond design basis accidents. The objective is to keep the nuclear facility within its operation limits, and if impossible, maintain sub-criticality and prevent release of radioactivity to the environment. Included in the other three documents are more general site, regional and national emergency plans to protect the public and environment against the consequences of a nuclear disaster. Education and training programs of the emergency plans at different levels (operators, public and authorities) are organized regularly. Evaluation of the training and education programs results in an emergency preparedness of the public and improvement of the different emergency plans.

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

The first level of defence is for all facilities on the OLP site established by a robust design of the buildings and operational equipment[3].

The second level of defence is for all nuclear facilities on the OLP site covered by operating manuals, working procedures and control systems. The availability of the necessary systems is ensured by a proper design.

The third level of defence is for the HFR ensured by the cooling capacity of the water in the pool. The availability of the SSCs to address the design basis accidents is ensured by design and confirmed by analysis and walk-downs. Maintenance, testing, inspection programs, and operator training to guarantee the availability of the systems are in place.

For the other nuclear facilities this level of defence is ensured by a proper design of the different confinement barriers of the nuclear material[4]. Proper use of operation procedures is part of extensive training programs.

The fourth level of defence is for the HFR as well as the other nuclear facilities is ensured by the different emergency plans on different levels. Training of all levels of personnel and authorities involved is implemented to ensure correct identification and implementation of the situation.

[3] This includes the HCL: detected haircracks in the foundation have been analysed extensively in the framework of plans to work with very heavy containers (beyond design specs) at this location. The analysis revealed that the floor could not bear these containers, but that the HCL could continue to operate in compliance with the specified maximum floor load.

[4] The barrier of the canisters is sometimes not complete for the WSF, as some canisters have deteriorated. Efforts for repackaging are underway. The remaining barriers are sufficient to ensure the third level of defence.

2.1.2.3 Protection against indirect effects of the earthquake

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

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

Internal flooding

Internal water pipes with a potential to cause internal flooding should be qualified as seismic category 2. The design and construction of these systems are addressed in the seismic walk-downs and in 10-yearly safety evaluation. Details about the internal flooding scenarios and its the consequences are addressed in Chapter 7 of this report (Other extreme events).

Internal fire

Protection against the risk of fire is ensured by a proper design of the safety systems and proper housekeeping. The walk-downs of the nuclear facilities address among others the fire-fighting systems and moreover the systems are checked on functionality on a regularly basis. The fire-fighting systems are supplied with water from the water make up building (“reinwaterkelder”). The water supply lines from this building to the nuclear facilities are not qualified as seismic category 1. Details about the consequences of internal fires are addressed in Chapter 7 of this report (Other extreme events).

High energy equipment

High energy equipment like tanks, high pressure gas cylinders installed inside the buildings of the HFR or the other nuclear facilities are fixed to walls etc. or physically separated to prevent in case of failure damage to other safety related SSCs. This is confirmed by good workmanship and seismic walk-downs.

LOCA

A LOCA in the HFR is excluded as a consequence of the DBE since the primary system and its components are designed to withstand the DBE. In case a LOCA occurs as a consequence of an earthquake, controlled opening of the pool injection valves on the reactor vessel will ensure core cooling. Details about the cooling modes are presented in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink).

Loss of ultimate heat sink

The ultimate heat sink after a DBE is the heat capacity and the evaporation of the water in the primary system and in the pool. If the level in the pool decreases refill is possible by opening of the “hydrofoor” pool emergency supply system starts water supply into the pool. If this fails the ultimate measure is to supply water into the pool from the KOTS storage tanks. However, these systems are not seismicly qualified. Details this scenario are addressed in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink).

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

Normally, loss of offsite power can be mitigated by the diesel generators that are installed in the emergency power building. HFR equipment that is essential for a safe shutdown of the reactor in case of accidents is connected to the emergency power grid (diesel generators and batteries). The diesel fuel stored in the emergency power building is sufficient for approximately 33 hours (Chapter 5 of this report). Additional diesel fuel is stored in a tank next to the emergency power building. Loss of power could be mitigated by an efficient use of the diesel generators. Details about the loss of power scenario are addressed in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink).

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

Based on the intensity of the DBE, the on- and offsite infrastructure remains functional after DBE.

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

Fire and explosions

The consequences of external fires and the very unlikely event of an external explosion are addressed in in Chapter 7 of this report (Other extreme events).

Liquefaction

Recently, the liquefaction on the OLP site was evaluated based on the seismic load caused by the DBE for the OLP site. Conservatively, for the analysis, a natural earthquake (magnitude 5 and duration 5 s.) and a gas induced earthquake (magnitude 3.9 and duration 3 s.) were assumed. According to the KTA requirements, liquefaction of the soil at the OLP site, caused by the assumed natural earthquake, is covered. The duration of the maximum PGA of a gas induced earthquake is very short 0.1 s and the energy released is small compared to natural earthquakes of identical magnitude. Based on the released energy and the distance to the OLP site liquefaction at the OLP site caused by a gas induced earthquake is very unlikely.

2.1.3 Compliance of nuclear facilities with 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.

Compliance of the plant with its current licensing basis is documented in the HFR Safety Report and the Technical Documentation. The availability of the safety related SSCs is ensured through maintenance and inspection programs, among others the 10-yearly safety evaluation program.

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.

Currently there exist no procedures and processes to ensure that mobile equipment and supplies that could be used are available after an earthquake.

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

No deviations with respect to the licensing base have been established.

2.2 Evaluation of the margins

The DBE for the OLP site (PGA 4 m/sec^2) is the enveloping curve of the maximum natural and non-natural earthquakes in the northwest of The Netherlands. The conservatism within the DBE is based on the following approaches:

- There is no evidence of any tectonically induced (natural) earthquake in the north of The Netherlands in the past. For a region with low seismic activity the US American regulatory code NRC 100 requires for the evaluation of the seismic response of a nuclear facility a DBE with a PGA of 1.0 m/s^2. However, for the OLP site, in 1998, it was agreed with the licensing authorities to compare the OLP site with nuclear sites in northwest Germany. The DBE response spectrum of a characteristic nuclear power plant in northwest Germany (maximum PGA 2 m/s^2) was defined as a conservative approach for a natural earthquake for the OLP site. Moreover an additional safety factor was used to set the DBE on 4 m/sec^2, making it twice more conservative.

- Gas induced (non-natural) earthquakes in the north of The Netherlands are related to the gas fields in the area. The gas fields in the OLP site region are the Bergermeer and the Camperduin gas fields. The observed natural gas induced earthquakes near the OLP site occurred all in the Bergermeer gas field. The epicentre of the earthquakes was in a natural fault in the centre of the gas field. Theoretical considerations about induced earthquakes show that these earthquakes do most likely not exceed a maximum magnitude. This is supported by statistical interpretation of the measured seismicity in the north of The Netherlands. KNMI estimated the maximum PGA of a gas induced earthquake in the north of The Netherlands at 4 m/s^2 with an intensity of VI to VII on the MMI scale and a Richter magnitude of 3.9.
- Based on experimental data, KNMI confirmed that the relation between the Richter magnitude, the maximum PGA and the distance to the epicentre as proposed by K.W. Campbell predicts the maximum of the gas induced earthquakes in the north of The Netherlands reasonably well. Applying the relation to the maximum gas induced earthquake (PGA 4 m/s^2, magnitude 3.9) to occur in the Camperduin and Bergermeer gas fields results in PGA's at the OLP site of 1.0 m/s^2 and 0.17 m/s^2 respectively (distance 5 km and 20 km respectively). Taking into account that an increase of intensity level by one unit results in doubling the PGA, it can be concluded that the MMI intensity of a gas induced earthquake at the OLP site will not exceed intensity IV to V on the MMI scale. This is also confirmed by the observation that, at the OLP site, there are no reports or observations of any incident related to the earthquakes in the Bergermeer gas field. In order to verify the local magnitude of an earthquake with epicentre in the Camperduin or the Bergermeer gas fields it is recommended to install seismic equipment on the HFR to monitor seismic activity.
- The collected experimental data shows that the gas induced earthquakes are shallow events with a relatively large PGA and short time duration, often one cycle only and a consequential low damage.
- In 1998, the DBE for the OLP site was defined based on the limited available data about gas induced earthquakes and a conservative approach for natural earthquakes. When more experimental seismic data came available, it appeared that the gas induced earthquakes always occur on top of the gas field. Moreover these earthquakes do not exceed a maximum magnitude. Based on the current available data and theoretical considerations a maximum PGA of 2 m/s^2 is considered as a conservative approach for a gas induced earthquake at the OLP site. Combination of this gas-induced earthquake with the natural earthquake defined for northwest Germany

results in a modified spectrum for a DBE at the OLP site equal to the DBE for a nuclear power plant in northwest Germany. This modified spectrum, with a maximum PGA of 2 m/s^2, is still conservative in relation to the PGA of 1.0 m/s^2 as required in NRC 100 for low seismic areas.

Since 1998 finite element analysis are performed to evaluate the earthquake resistance of the relevant SSCs on the OLP site. The seismic load condition for the analysis is based on the DBE defined in 1998 for the OLP site. Table 2-4 gives a summary of the lowest safety factor of the SSCs for the assumed seismic load condition. The analysis show that for the hot cell panels inside the reactor building of the HFR the minimum safety factor is 1.11. For the other nuclear facilities on the OLP site a minimum safety factor of 1.10 is calculated for the floor beams of the MPF building. For other selected SSCs significant higher minimum safety factors are calculated. The minimum safety margins are calculated for a DBE with a PGA of 4 m/s2. If the previous mentioned modified spectrum (maximum PGA 2 m/s2) is assumed as seismic load the safety margins will double. For this modified spectrum, which is still a conservative spectrum for the OLP site, the safety margins on the OLP site are 2.2 and higher. Range of earthquake leading to severe fuel damage

Conservatively it is assumed that failing of the hot panels and supports on top of the pool could result in damaged pool walls and consequently loss of cooling capacity of water of the pool. The minimum safety factor calculated for the SSCs in the HFR is 1.11 for the hot cell panels and supports. Assuming that SSCs that are not seismic qualified in category 1 are not available, an earthquake with a PGA> 4.4 m/s^2 (1.11 x 4 m/s^2) could result in core heat-up.

In the other nuclear facilities the nature of the nuclear material cannot lead to severe fuel damage, except for the uranium filters in the pool[5] in the HCL building. For the pool in the HCL no quantitative margin is determined. However, it can be assumed that the margin is covered by the safety margin of the HCL/MPF building. Therefore the defined PGA of 4.4 m/s^2 also covers the uranium filters in the HCL.

2.2.1 Range of earthquake leading to loss of containment integrity

The containment of the HFR is seismic qualified as category 1. Structural analysis of the steel dome of the HFR show a minimum safety factor of 3.20. Containment integrity is therefore[6] ensured for an

[5] The HCL pool has a shielding function only, not a cooling function.

[6] Safety factor and PGA have a linear relationship. The DBE (4.0 m/s^2) resulted in a safety factor of 3.2 for the selected structure, therefore a PGA of 12.8 m/s^2 results in a safety factor of 1.0.

earthquake with a horizontal PGA up to 12.8 m/s^2. For the other nuclear facilities no quantitative margin is determined.

2.2.2 Earthquake exceeding the design basis earthquake for the OLP and consequent flooding exceeding design basis flood

An earthquake at a seaside location like the OLP may cause a tsunami, which in turn could cause damage to the dunes. The high water level, required for the flooding of the damaged dunes, may be caused by a tsunami or high tide. Details about these flooding scenarios and the consequences of the flooding are addressed in Chapter 3 of this report (Flooding).

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

There are two situations identified beyond which it would be impossible to avoid the development of a serious accident (cliff-edge effect):

Unavailability of shift personnel

There is a potential for a cliff-edge after a DBE if the control room is destroyed (control room is not qualified for DBE) and the site is inaccessible. This situation has the potential to develop in a core heat-up scenario followed by a core melt scenario. Personnel issues are elaborated in Chapter 6 of this report (Severe Accident Management).

Unavailability of long term emergency water supply to the pool of the HFR

The water supply from the water make up building (“reinwaterkelder”) and the KOTS to the reactor building are not qualified for a DBE. Loss of this function results in the long term loss of core cooling (after 600-700 hours). Moreover the water supply for the fire-fighting systems are not ensured.

Component / structure	Minimum safety margin for DBE load
High Flux Reactor	
• Reactor building concrete substructure	1.25
• Low density concrete walls of the pool	1.24
• Hot cell panels and support inside reactor building	1.11
• HFR steel dome structure incl. crane, expansion vessel	3.20
• Pressure boundary primary cooling system including decay heat removal system	4.80
• Reactor vessel including SCRAM system	1.89
• Stack	1.86
• Primary pump building	1.53
• Pumps pool cooling system in cell 8	1.40
Hot Cell Laboratory	
• Stack	1.88
• MPF building	1.10
Low Flux Reactor	
• Fuel elements	3.04
Waste Storage Facility	
• Lifting device	6.09

Table 2-4: Analytical determined minimum safety margins of buildings and structures[7]

The following modifications and/or investigations could be envisaged to increase the robustness of the nuclear facilities on the OLP site against earthquakes:

- A systematic seismic qualification of the SSCs of the HFR in agreement with international accepted requirements and guidelines. Included under this point are the recommendations from the recent INSARR mission to the HFR. In particular the following points need attention and may improve the robustness of the HFR against seismic loads:
    - Seismic evaluation of the control room and upgrading of the planned monitoring control room to an emergency control room.

[7] The minimum seismic safety margins as presented in this table should be viewed with caution. These safety margins are determined based on the results of code evaluations and also incorporate other loads (e.g. gravity, pressure), besides the seismic load. Therefore, the margins mentioned are conservative predictions for the seismic margins with respect to the already conservative DBE.

- Seismic evaluation and improvement of the waterline from the water make up building ("reinwaterkelder") to the HFR will increase the long term availability of emergency water supply to the pool and to the fire-fighting system.
- Update the seismic walk-down of the HFR in agreement with the current requirements (10-yearly safety evaluation).
- Evaluation of the possibility to install seismic instrumentation to initiate reactor scram after an earthquake and monitor the local magnitude of an earthquake at the HFR site.
- Increasing the seismic resistance of the diesel fuel tank outside the emergency power building should be considered.
- Currently actuation of the alternative shutdown system, the pool injection valves and the convection valves on the reactor vessel are initiated with manual actions inside the reactor building. The pool injection valves can also be activated from the control room. Moving those manual actions to the control room and the planned monitoring control room would increase the availability of the heat removal capacity.

- The currently used Design Based earthquake will be evaluated against the current IAEA requirements and recommendations established in the IAEA. This is also a recommendation from the recent INSARR mission to the HFR.
  - The current DBE is most likely a very conservative estimate for a DBE for the OLP site. Most likely a re-evaluation of the seismic history and geology of the OLP site will result in a less conservative DBE.
  - Soil-structure interaction effects and the behaviour of the subsurface soil after DBE are to be investigated (planned for the 10-yearly periodic safety review of the nuclear installations on the OLP site).
- The existing seismic analysis of the HCL building structure and related components will be extended.
- For the buildings other than those of the HFR it is recommended to evaluate for each building the barriers between the stored and/or handled radioactive material en the environment into detail.

3 Flooding

This chapter describes the outline of the analysis of the OLP with respect to flooding conditions. The flooding conditions considered are:

- high tides and storm surges;
- tsunamis;
- extreme rain fall;
- on site pipe rupture;
- credible combinations of the items mentioned above.

Each facility may consist of several buildings. Table 3-1 describes which buildings belong to each facility and specifies which buildings are not in scope of this assessment. Buildings not in scope are those that fulfil no safety function. These buildings are indicated with ‘N’ in Table 3-1. The height contours of the OLP can be seen in Figure 3-2. Information on elevation and mutual connections of the distribution bus bars and transformers of interest is given in the Table 3-2 and Table 3-3. Details can be found in Appendix A.3.

Facility code	Facility name	Building name	Safety function[8]
HCL-RL	Hot Cell Laboratory	LSO	Cn, ,R
HCL-MFP	Molybdenum Production Facility	LSO	Cn, ,R
WSF	Waste Storage Facility	Radioactive waste building	Cn
JGL	Jaap Goedkoop Laboratory	Jaap Goedkoop Laboratory	Cn
STEK	STEK hall	STEK hall	Cn
HFR	High Flux Reactor	Reactor building [9]	Cn, Co ,R
		Reactor outbuilding	Cn, Co
		Primary pump building Storage tanks Storage building	Cn, Co
		Secondary pump building	U
		Chlorine bleach building	N [10]
		Air treatment building	Cn
		Security lodge	N
		NDE building	N
LFR	Low Flux Reactor	Fermi building	N[11]
DWT	Decontamination and Waste Treatment	Decontamination building	Cn
		Water treatment building	Cn
		Waste treatment and recycling /pipe cleaning building	Cn
ECC [12]	Emergency Coordination Centre	Forum	SAM
		Building at Joint Research Centre (JRC)	SAM
	Communication / ICT	General laboratory [13]	SAM
EPS	Emergency Power System	Emergency power building	Cn, Co,

Table 3-1: Overview of the OLP buildings.

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[8] Cn = confinement, Co = cooling, R = reactivity control, U = ultimate heat sink, N = none, SAM: severe accident management

[9] The roof of the reactor building is a spherical dome that prevents the accumulation of precipitation.

[10] The “chloorbleeklooggebouw” may contain chemicals but fulfills no safety function.

[11] The LFR is out of operation

[12] The ECC does not fulfill safety functions itself, but may play a supporting role in fulfilling the safety functions of the other facilities.

[13] In the basement of the General Laboratory building, ICT servers are located. These servers play an important role in telecommunication.

3.1 Design basis

3.1.1 Flooding against which the plant is designed

The OLP buildings are not designed against a postulated design basis flood height.

3.1.1.1 Characteristics of the design basis flood (DBF)

Since no design basis flood was defined, there are no characteristics of the design basis flood.

3.1.1.2 Methodology used to evaluate the design basis flood

Since no design basis flood was defined, evaluation is not possible.

3.1.1.3 Conclusion on the adequacy of protection against external flooding

Conclusions on the adequacy of protection against external flooding cannot be drawn based on the previous paragraphs. The adequacy of the resistance against external flooding will be considered in the next sections, where the impact of on-site and off-site flood, including the possible flooding causes, on safety functions is analysed.

3.1.2 Provisions to protect the plant against the design basis flood

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

This section gives a brief overview of safety functions and SSCs of each facility.

Safety functions and SSCs of the HFR

- *Control of reactivity*: a reactor core and spent fuel are present plus (remains of) experiments, HEU fissile targets, and waste of isotopes production. Main systems or components are the control rods, the control rod drive mechanisms and activation by the reactor protection system. Spent fuel is kept in a safe subcritical configuration by design and procedures;
- *Cooling*: a reactor core and spent fuel are present. Main cooling systems are the closed loop primary coolant system, the decay heat removal system and pool cooling system. All these systems are cooled by the open loop secondary water cooling system;

- *Confinement:* in succession fuel matrix, fuel cladding, primary system boundary (in combination with the pool and the off-gas system) and the ultimate barrier: the containment building. To assure confinement, viz. control of releases of radioactive materials, the containment is under-pressurised.

Safety functions and SSCs of the HCL-RL

- *Control of reactivity*: spent fissile material is kept in a safe subcritical configuration by design and procedures;
- *Confinement*: radioactive materials are present. Main systems where this activity is present are: matrix and cladding (to a limited extent), filter-container, pool, hot-cells, ventilation system, HCL-RL containment; Confinement is ensured by maintaining under-pressure in the hot cells and the building itself, and by storing the radioactive material when outside the hot cells in special containers. Pools in which fuel containing experiments are stored only have a shielding function and not a cooling function.

Safety functions and SSCs of the HCL-MPF

- *Confinement:* a significant amount of the radioactive materials is present. Main systems are: hot-cells, ventilation, waste storage tanks. Confinement is ensured by maintaining under-pressure in the hot cells and the building itself, and by storing the radioactive material when outside the hot cells in special containers.

Safety functions and SSCs of the decontamination building

- *Confinement*: a limited amount of radioactive materials is present in the building. All contaminated fluids used for cleaning are collected in the sump of the basement and transported to the Water Treatment Building. The ventilation system provides for containment by maintaining under-pressure in the building.

Safety functions and SSCs of the Water treatment

- *Confinement*: a very limited amount of radioactive materials is present in the waste storage tanks. The tanks provide the confinement function.

Safety functions and SSCs of the DWT

- *Confinement*: radioactive materials are present. Confinement is ensured by maintaining under-pressure

Safety functions and SSCs of the Pipe Cleaning

The pipe cleaning building has no confinement function. Under the centrifuge is a sludge tank for collection of waste from the pipe cleaning.

Safety functions and SSCs of the WSF

- *Control of reactivity*: spent fuel and other fissile material are kept in a safe subcritical configuration by design and procedures;
- *Confinement:* a significant amount of radioactive materials and a limited amount of spent fuel are present. Main safety systems are filter-containment, pipes and trenches.

Safety functions and SSCs of the STEK

The STEK hall is out of service, but is used for storage of (water tight) containers with radioactive material. The building itself provides no containment function. However, although not providing a gas tight containment, its thick walls protect the environment from direct radiation and the containers inside from external hazards.

Safety functions and SSCs of the LFR

The reactor core and its spent fuel have been brought in a permanent shutdown configuration and the fuel has been removed from the core. Fresh and spent fuel have been placed is the LFR's own storage facility, where the fuel is kept in a safe subcritical configuration by design. In addition, the near term relocation of the spent fuel is being planned. Due to the status of the facility neither the control of reactivity nor the heat removal function are relevant safety functions.

Safety functions and SSCs of the Telecommunication Centre (in the General Laboratory)

Although this building has no direct relation to the safety functions under consideration, the building houses equipment that is important for communication on-site as well as off-site.

Safety functions and SSCs of the Emergency Coordination Centre (in the Forum building)

Although this building has no direct relation to the safety functions under consideration, the building houses equipment that is important for the coordination during an on-site accident with potential radiological consequences. The ECC in this building has the same function as the ECC at JRC.

Safety functions and SSCs of the Emergency Coordination Centre at JRC

Although this building has no direct relation to the safety functions under consideration, the building houses equipment that is important for the coordination during an on-site accident with potential radiological consequences. The ECC in this building has the same function as the ECC in the Forum building.

Safety functions and SSCs of the Jaap Goedkoop Laboratory

- *Confinement*: a very limited amount of radioactive materials is present; Confinement is preserved by maintaining under-pressure;.
- Main safety systems include ventilation system and glove boxes.

Safety functions and SSCs of the Electrical Power Supply

The electrical power supply on the OLP consists of 10kV distribution network (public grid, see Figure 23 (chapter 1)) and a back-up Emergency Power System consisting of 3 diesel generators in a separate building and battery packs in two of the HFR buildings (details can be found in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink)). Although the power supply systems are no front line systems themselves, they are important for safety as support systems for the front line systems.

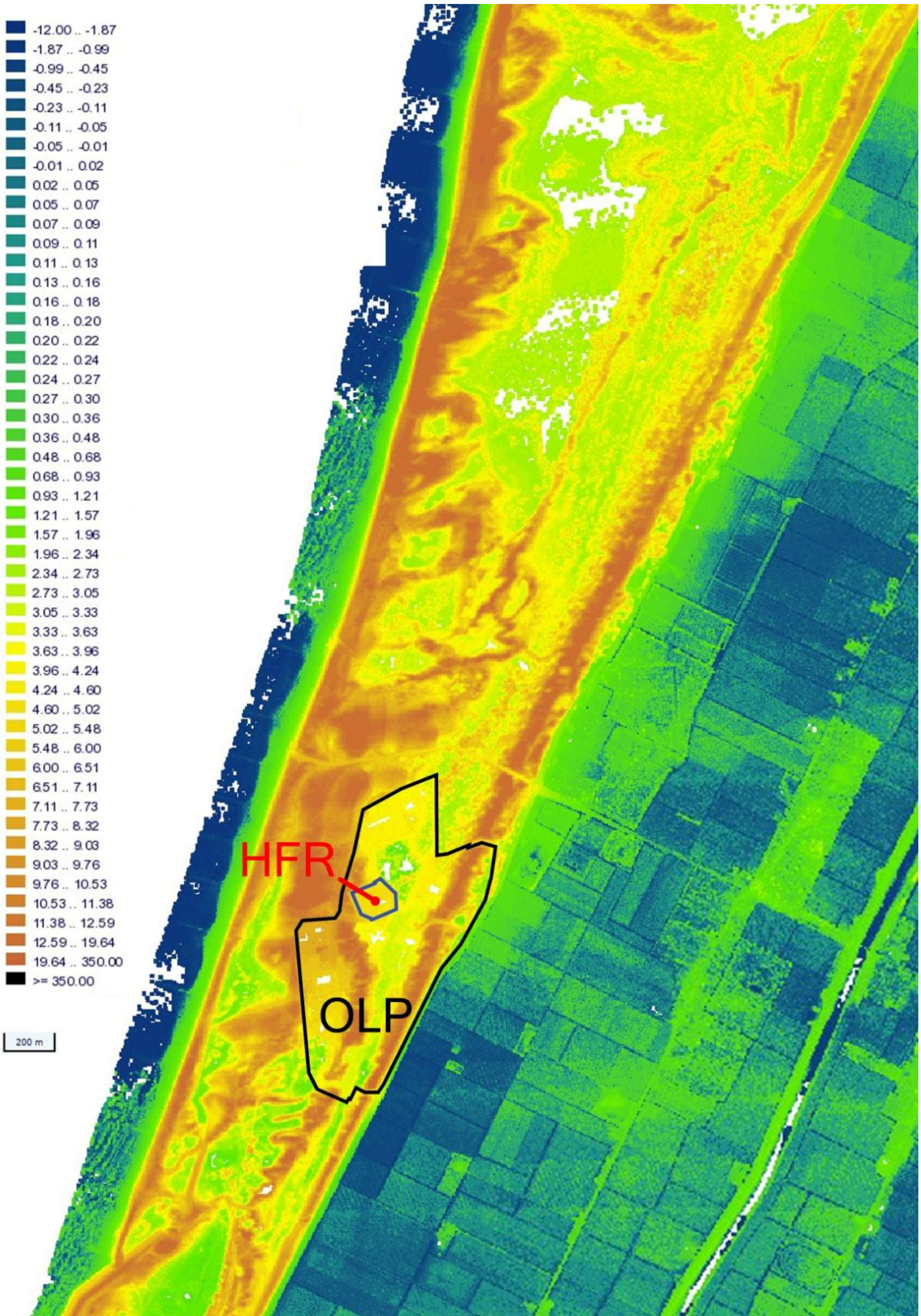

Figure 3-1: Height contours at OLP site and surroundings in meters NAP.

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

The following text is a summary of what is stated in the safety report of the HFR with respect to site characteristics. The OLP is located in a dune area. The North Sea is to the West and the Zijper Polder is towards the east. The distance between the North Sea and the relevant buildings lies between 250 and 500 m. A minimum of two rows of dunes separates the buildings from the sea. At the Zijper Polder side a third row of dunes marks the sharp transition towards the polder. Figure 3-1 gives an overview of the location including height differences. Details on the OLP are shown in Figure 3-2. Terrain heights vary between 2 and 15 m+NAP. Minimum elevation of the relevant buildings is approximately 4 m+NAP. The dunes itself have a height between approximately 8 and 16 m+NAP. The Zijper Polder is at 0 m+NAP.

Given the difference in elevation between large, very low hinterland (0 m+NAP) and the site, and the rows of dunes protecting the OLP a breach of the dunes or dikes not directly in front of OLP will not jeopardise the OLP.

A breach directly in front of the OLP, however, could threaten the OLP. The first row of dunes is kept in such a condition that the failure frequency is once in 10,000 years, as required by law ("Waterwet"). This return period corresponds to a storm surge with a mean water level of 4 m+NAP. The highest measured water level at the OLP location is 3.20 m NAP, during the February storm disaster of 1953.

The "Waterwet" also ensures that with regular inspections and a five-yearly review of the boundary conditions, the condition of the dunes is kept up to the mark. As a result of this, the first dune row will be improved in the period 2012/2013.

However, this does not signify that the OLP will be flooded with a mean frequency of once every 10,000 years, while the second row of dunes and the sand masses between the two dune rows will still protect the site. At least 250 m of dunes has to be eroded away by seawater, before the first buildings are reached. This requires an attack time far longer than during normal storm surges or it requires storm surges with much higher water levels. In both cases the return period of such events will be much longer than 10,000 years. A mean water level of 6 m+NAP corresponds with a return period of over 1 million year.

Saturation of the sand will not impact the most important building (from a safety point of view), namely the reactor building, as it is placed on piles.

Given the nature of the site, all buildings do not have provisions to protect them against flooding or water entering, other than measures to prevent precipitation entering the buildings. This means that flooding of (parts of) buildings by extreme rain cannot be excluded. For the specific case of the HFR, flooding caused by failure of the secondary cooling pipe is also possible.

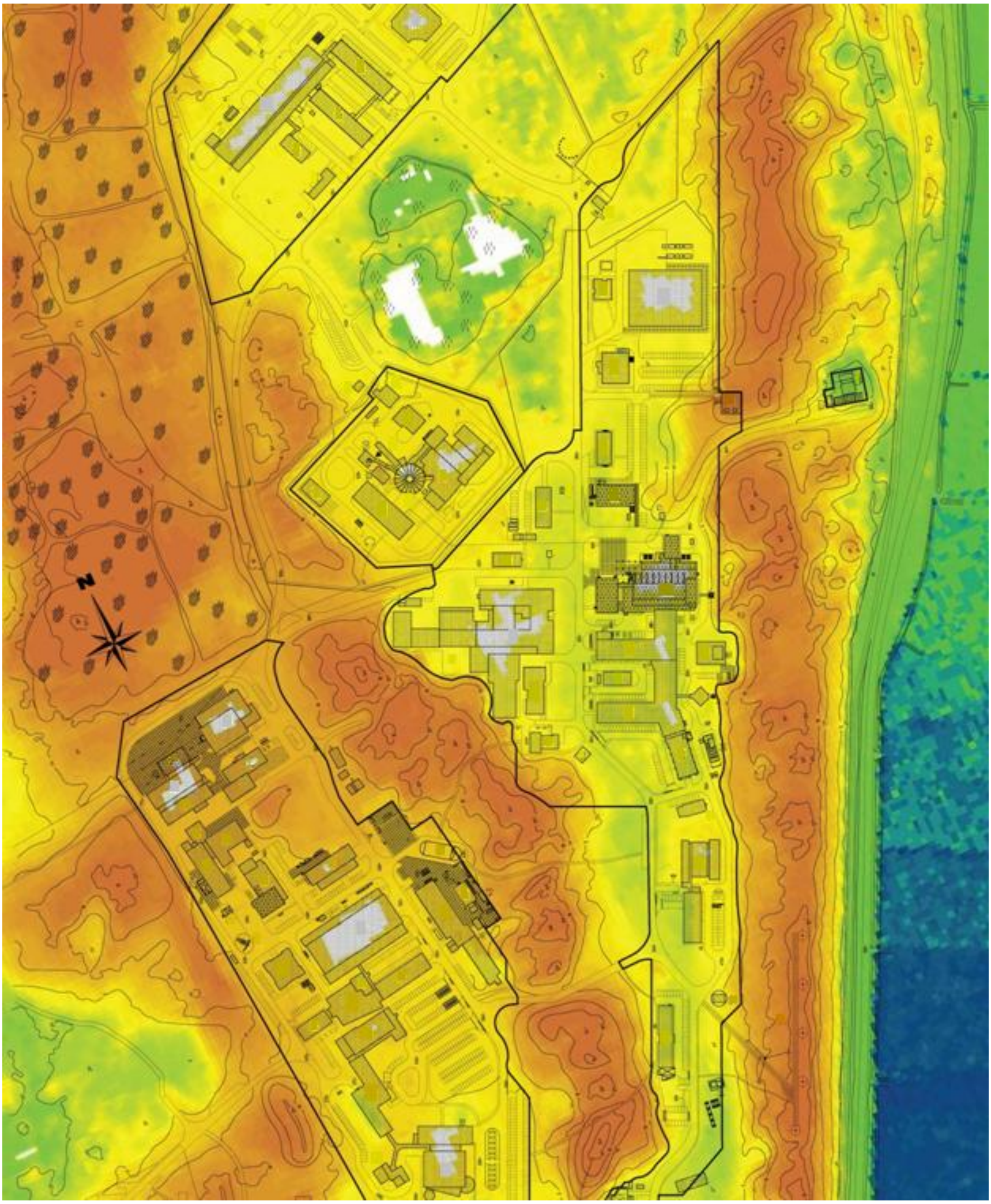

Figure 3-2: Height contours at OLP site. Picture based on the OLP site map and the elevation map of The Netherlands.

3.1.2.3 Main operating provisions to prevent flood impact to the plant

There are no operating provisions to invoke in case of expected high tides or other possible causes of flooding for any of the activities on the OLP. However, the HFR will be shut down at a mean sea water level of 3.25 m+NAP, This is enforced by “Rijkswaterstaat” as part of their flood defence measures.

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

Although the most probable scenario is that the site itself is not directly threatened by flooding, a large area of Noord Holland is. Flooding of this area will have a large impact on the operation on the OLP. One has to assume that external power is not available and that one has to rely on the emergency diesel generators for electrical power. The site will be difficult to reach for staff as well as supplies, as road infrastructure will be affected. Also communication will be a major problem. Telephone (conventional and mobile), internet and C2000 can be considered lost.

In case the site itself will not be flooded, air lifting of personnel, diesel fuel, food and other supplies is still possible. There are no specific procedures present to deal with a flooding situation. In case the site itself is (partly) flooded also air lifting will become increasingly difficult.

Existing procedures should be reviewed with regard to flooding conditions, and actions that have to be executed to improve the robustness for flooding of the OLP should be identified.

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

All facilities have their own maintenance system and operating procedures to ascertain the proper functioning of structures, systems, and components. No specific measures and procedures exist for (threatening) external flood situations.

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 there are no procedures and processes to deal with flooding situations, there are no procedures that deal with mobile equipment and supplies that are planned for use, nor are those features available or external on stock. General procedures are in place to ascertain minimum amounts of diesel fuel and fresh water supplies for cooling purposes. The on-site emergency power system is equipped with a common main tank with a capacity of 10 m^3. At a level of 4 m^3 a top-up order is given to the oil company. The required minimum level of the tank is at a volume of 0.5 m^3. This leads to a worst case net capacity of the main tank of 3.5 m^3. In accident conditions water supply to the pool for refill can be provided by the public water supply system, via the hydrophore system. This can be performed by pumps or gravity driven. For the latter a separate pipeline is installed; however the feasibility of this facility to refill the pool still has to be elaborated (Chapter 5 of this report). A minimum of 85 m^3 water is kept reserved in the fresh water basins in the drinking water pumping station.

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

No deviations with respect to the licensing base have been established.

3.2 Flood Sources

In addition to the ENSREG report contents, this section considers the flood sources.

The potential flood sources are:

- high tide, possibly combined with storm surge;
- tsunami;
- extreme rainfall;
- secondary cooling water pipe rupture.

These sources will be discussed in the next sections.

3.2.1 High tide, possibly combined with storm surge

A high tide, possibly in combination with a storm surge may lead to flooding. However, the dunes in front of the OLP form a multiple barrier of at least 250 m wide and a height between 8 and 16 m+NAP. This limits the flooding probability enormously. The probability that the nearby dike “Pettemer Zeedijk” or “Hondsbossche Zeewering” will fail is much higher than the probability that the dunes in front of the OLP will fail. This means that the flooding threat of the OLP will most probably come from the land side. In this case the impact is limited by:

1. a single row of dunes at the landside of similar height as the dunes at the seaside;
2. the vast area of land that has an elevation of 0 m+NAP, at the land side, that has to be filled with water; and

3. the westerly wind, in case of such a flooding that will result in very limited wave action at the protective dunes at the land side at the east.

Although the probability is very low, flooding caused by high tides/storm surges cannot be completely excluded.

3.2.2 Tsunami

3.2.2.1 Storegga Slide (approximately 8200 years ago)

Large parts of the now submerged North Sea continental shelf (Doggerland) were flooded by the Storegga Slide tsunami, one of the largest tsunamis known for the Holocene, which was generated on the Norwegian coastal margin by a submarine landslide. The characteristics of the wave triggered by this ancient event were simulated. A calculated initial wave height of 3 m at the source of the model results in maximum deviations of about 0.5 to 0.7 m at the tidal gauges in the German Bight. It would take approximately 8.5 hours for the first wave to reach the German coast line.

Geological models suggests 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.

3.2.2.2 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 5 to 8 hours waves with a height of 0.8 to 2 m reached the Cornwall coasts with localized 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.

3.2.2.3 Dogger Bank earthquake (1931)

This earthquake, which measured 6.1 on the Richter scale, caused a small tsunami (wave amplitude at the origin 1 m). After 1 to 2 hours waves with a height of 0.8 to 2 m reached the Yorkshire and Humberside coasts. Furthermore, a possible source of a tsunami in the future has been identified, see following section.

3.2.2.4 La Palma landslide

Research has suggested that the western flank of La Palma Island is vulnerable to collapse. Likely UK coasts affected by a tsunami arising from such a collapse are those of Cornwall north and south Devon. Wave elevations of 2 m (with estimated original amplitude of 1 to 2 m) can be expected to reach the British coast in 7 to 8 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.

3.2.2.5 Possible impact at Petten

In a study performed 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 result the risk of flooding due to a tsunami is regarded as non-existent. Even when a tsunami is combined with the most extreme recorded storm surge since 1932[14] (3.25 m+NAP at Petten, 1953) this would result in a water level of 4.65 m (1.4 + 3.25) m+NAP, which is still below design level of the dunes and the level of the HFR (4.7 m+NAP) This conclusion is supported by more recent research. For the German Bight it was concluded in 2007 that Cuxhaven will be protected from the catastrophic impacts of a hypothetical tsunami. Waves of 0.5 m are expected. The same study showed waves of approximately 2 m at IJmuiden. 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.

3.2.3 Rainfall

The statistics of the KNMI show that for the area of Petten once every thousand years 117 mm of rainfall can be expected within a period of 24 hours. For the same conditions, 133 mm of rainfall can be expected within 48 hours. In the dune area unobstructed natural drainage of rainwater takes place through a sand layer of 18 meters thick. Only in very severe winters freezing and therefore partial clogging of the sand layer is possible and precipitation will collect in the deep dune valleys.

The HFR is located at a height of 5 m+NAP. In the immediate vicinity dune valleys are lower. Although it is therefore unlikely that the rain water will collect in large amounts in the direct vicinity of the HFR, it cannot be ruled out. The hilly surroundings and paving around the buildings make it very difficult to assess were water will collect. Moreover the minimum height the water has to rise is approximately

[14] Notice that these events are uncorrelated.

25 cm, before it can enter the reactor outbuilding basement, see paragraph 3.3.1. The reactor building itself is water-tight as it is sealed to preserve under-pressure. Water entering the reactor outbuilding basement will not be noticed by the operators as no means of detection are present. The ultimate consequence is a total loss of power in the reactor building and in the control room, including the battery power for instrumentation and control. If the water enters the PPG, the primary pumps and decay heat removal pumps may become unavailable. This means loss of active cooling for the HFR.

Also for all other buildings rain entering the building cannot be excluded, as the buildings have no special provisions to keep rain water out of the building in case of very heavy rain. This is especially true for buildings with (large) paved areas around them such as HCL-RL and HCL-MPF. Special attention should be paid to the WSF that is surrounded by approximately 2500m^2 of fluid tight pavement. The drain pump has a capacity between 10 – 15 m^3 per hour. According to KNMI statistics on hourly precipitation, 7 mm per hour can be expected five times a year, an amount that fails the pump capacity. However the fall and lay-out of the pavement is designed to create a storage capacity of approximately 125m^3 before water enters the building. The return period for 125m^3 of rain is over a 100 years. Combined with the pump capacity this means that rain should not be a problem, as long as the fall of the pavement is preserved.

The consequences will however be limited to possible loss of containment.

3.3 Evaluation of safety margins

3.3.1 Estimation of safety margin against flooding

First the effects and consequences of loss of electricity supply by on site and off site flooding will be described. Next, the effect and consequences of water on site is described per building.

3.3.1.1 Flooding of the Electricity Supply

System description

The basic diagram showing the off-site 10 kV main electrical distribution system of the OLP is presented in Figure 23 (chapter 1). The preferred 10 kV supply is provided by two separate feeders (ground cables) from the Liander 50 kV public grid switchyard Schagen. In the field the 10 kV feeders are rolled out in separate tracks. The north track, marked in blue in Figure 23 (chapter 1), connects Schagen consecutively to transformer stations Secondary Pump Building, Covidien 3, Primary Pump Building and Technology building. The location of these transformers, the building numbers, the elevations, and mutual connections are summarized in Table 3-2. The south track, marked in red, connects Schagen consecutively to transformer stations AS Test Field, ECN 4, ECN 5, ECN 4, Covidien 2, ECN 6, ECN7 and Covidien. The summary is given in Table 3-3.

The circuit between the north and the south track can be closed manually: all the transformers in the Secondary Pump Building and the Primary Pump building can be connected to the Covidien transformer station. In case of high water level, this Covidien transformer is the first one that will be flooded. It can therefore be assumed that the connection is not available in the complementary safety margin assessment.

Transformer	Building	Elevation		Connected transformer(s)	Connected building(s)
		floor	bus bar		
		[m+NAP]	[m+NAP]		
Secondary Pump Building		5.5	5.75	Covidien 3	
Covidien 3		4.2	4.45	Primary Pump Building	-
Primary Pump Building		5.5	5.75	Technologiehal	
Technologiehal		4.3	4.3	-	

Table 3-2: Transformers on OLP in the north trace.

Transformer	Building	Elevation		Connected transformer(s)	Connected building(s)
		floor	bus bar		
		[m+NAP]	[m+NAP]		
AS Testveld		12.5	12.75	ECN 4	-
Testveld 1		12.5	12.75	-	-
ECN 4		5.85	6.1	ECN 5	
ECN 5		6.3	6.55	ECN 3	-
ECN 3		4.6	4.85	Covidien 2, ECN 7	MPF
Covidien 2		4.4	4.65	ECN 6	-
ECN 7		3.9	4.15	-	-
ECN 6		3.9	4.15	Covidien	HCL
Covidien		3.25	3.5	-	-

Table 3-3: Transformers on OLP in the south trace.

The elevations in the table are ground elevations plus 0.25 m. It is assumed that short circuit occurs if the water level is 0.25 m above ground level. It is further assumed that the safety functions of the transformer stations work properly. This means that the circuit breakers to other transformer stations transfer open in case of short circuit.

Next to the public grid an Emergency Power system is present in a separate building, with a critical flooding height of 4.10 m+NAP.

Effects

The only flooding source of significance that is capable of threatening the whole OLP is flooding caused by high tides and storm surges. In case of such an event, it is very likely that – given the characteristics of

OLP location (located in a dune area) - all off-site power is lost, because the Schagen switch yard or a transformer station between Schagen and the OLP is also flooded, while the OLP itself is not. A loss of off-site power (LOOP) situation is the result. Another situation is flooding of the OLP itself, by failure of the dunes in front of the OLP. In that case off-site power may still be available and the consequences are determined by the water level on site.

Consequences

A LOOP: the hinterland is flooded, but the OLP itself is not affected.

The consequences of LOOP (diesel generators available) are described in detail in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink):

- The STEK building will lose its electrical supply. As the buildings are out of operation and are used to store transport containers with amounts of low, intermediate and high level radioactive waste (LAVA, MAVA, HAVA) pending their shipment to COVRA, this has no consequences;
- For the LFR the consequence is that radiation monitoring fails. Since the LFR is not operated anymore, the ventilation system is in the shutdown mode, including closed ventilation shut-off valves. So this has no further consequences.
- The ECC's and the Telecommunication Centre in the General Laboratory building lose their function.

The remaining buildings have back-up power supply from the emergency diesels, that ensure the functioning of the required safety functions: the confinement and cooling function for the HFR and the confinement function (under-pressure) for HCL-RL, HCL-MPF, water treatment building, DWT, WSF, LFR and JGL.

B OLP is flooded

In this case six cliff edge water levels can be distinguished for the electrical system:

1. 4.10 m+NAP: the water level reaches the air inlet of the diesel generators. This results in loss of the Emergency Power supply. This means that at the next five cliff edge levels a complete loss of AC power results
2. 4.15 m+NAP: transformer ECN 6 is flooded, resulting in short circuit in this transformer. HCL-RL is without electricity.
3. 4.45 m+NAP: transformer Covidien 3 is flooded. The connection to the transformers of the Primary Pump Building and Technology building is cut off. The Reactor Building, the Reactor Outbuilding,

the Air Treatment Building, the Primary Pump Building and the JRC building housing the Emergency Coordination Centre are without electricity.

4. 4.85 m+NAP: transformer ECN 3 is flooded. The HCL-MPF and the Jaap Goedkoop Laboratory are without electricity.
5. 5.75 m+NAP: transformer Secondary Pump Building is flooded. The Secondary Pump Building is without electricity.
6. 6.10 m+NAP: transformer ECN 4 is flooded. The Water Treatment Building, Waste Treatment Building, the Pipe Cleaning Building, the Radioactive Waste Building, the Fermi Building, the General Laboratory and the Forum building are without electricity.

3.3.1.2 Flooding of the HFR, ground level 5.00 m

Effect

For the HFR, the following sequence occurs if the water level rises:

- 4.10 m+NAP: the water level reaches the air inlet of the diesel generators. This results in failure of the diesel generators. If off-site power is already lost due to problems off-site the OLP, all electricity supply to the HFR is lost, with the exception of the battery powered instrumentation and control (3 hours grace time) and the decay-heat removal pumps (approx. 2 hour grace time). This is the SBO1 situation as described in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink). In a SBO1 situation radio-isotope production facilities will lose their active cooling, however cooling from core and spent fuel will be provided by water of the reactor and storage pools;
- 4.30 m+NAP: the water level enters the control of the diesel generators. This has no effect, because the diesel generators failed at a lower water level;
- 4.45 m+NAP: transformer Covidien 3 has short circuit. This results in a complete loss of electric power supply (as the diesels are already failed at 4.10 m+NAP) in the Reactor Building, the Reactor Outbuilding, the Air Treatment Building the Primary Pump Building, with the following consequences:
    - no ventilation of the containment;
    - loss of electricity supply to the HFR. Instrumentation and control of the reactor is switched over to battery power. These batteries have a capacity of 3 hours. The decay heat pumps can by run on battery power (from a dedicated set) for one hour.
- 5.00 m+NAP: water on the roof of the duct from the Reactor Building to the Primary Pump Building (“swan lake”). The roof will collapse, resulting in water in the swan lake and the cellars of the primary pump building. The decay tanks and the hot drain tanks become submerged. The decay tanks

are always filled with water and will not float. The hot drain tanks can be empty. It is expected that these tanks will break away from the floor and will float. The contents will be released;

- 5.00 m+NAP: water in the inlet duct of the Air Treatment Building. This has no consequence, because the ventilation failed when transformer Covidien 3 already failed;
- 5.00 m+NAP: the transfer hall ("Hoge Montage hal") floods. The drums with nuclear material are submerged. This has no consequences, because the drums will not float and are water tight;
- 5.00 m+NAP: water enters the primary pump building. The pool cooling pumps and the decay heat pumps are submerged. This effects only the decay heat removal pumps, because the electricity supply to the primary pumps failed when transformer Covidien 3 failed. Forced circulation of the primary circuit is lost.
- 5.25 m+NAP: the basement of the Reactor Outbuilding floods by water entering the basement inlet outside of the building. The effect is a complete station black out, as all bus bars will become submerged, which means that the battery powered electricity supply to instrumentation and control is also unavailable. Also, the racks of the instrumentation and control itself are submerged, resulting in loss of these systems.
- 5.25 m+NAP: short circuit in the converter for the batteries. This results in failure of the emergency power to the decay heat removal pumps. However the pumps itself fail at a lower level;
- 5.40 m+NAP: the water submerges the batteries clamps of the HFR Uninterruptible Power System. This has no consequences, because the converter failed at a lower water level;
- 5.75 m+NAP: transformer Secondary Pump Building has short circuit. This results in loss of electric power supply in the Secondary Pump Building and subsequent loss of secondary cooling flow. This has no additional consequences as the HFR already is in station black-out situation and LOOP already caused loss of secondary cooling;
- 6.00 m+NAP: the transformers in the Primary Pump Building are submerged. This has no effect, because the electricity failed when transformer Covidien 3 failed;
- 6.50 m+NAP: there are several possibilities of water leakage into the containment. It is assumed that 1.4 m water level is necessary to breach containment: water in the transfer hall pushes the door to the containment open.
- 7.10 m+NAP: water reaches the drinking water pump station, with the dedicated reserve cooling water storage. There are no consequences as the HFR assigned components are passive and not influenced by flooding. Valves are located in the basement of the HFR-outbuilding.

Cliff edges

Several cliff edge water levels can be distinguished for the HFR:

- 4.10 m+NAP: failure of the diesel generators. As normal power is still available, there are no consequences and therefore no real cliff edge. If off-site power is also failed by this water level (because of flooding of the hinterland) a SBO1 situation develops (Chapter 5 of this report): battery power available for decay heat removal (approx. 2 hour) and instrumentation and control (3 hours). Main consequence is the loss of active cooling. Opening of the convection valves is necessary, after depletion of the decay heat batteries, to maintain cooling in this situation. Approximately 600 hours are available to take measures before the spent fuel15 becomes uncovered. See Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink) for more detailed scenarios and consequences. Isotope production facilities that require forced cooling could heat up and become damaged, causing a small release inside containment. At present the available thermo-hydraulic calculations of the Mykonos facility do not provide conclusive evidence to preclude that damage.
- 4.45 m+NAP: two effects: (1) failure of containment ventilation; the containment will be isolated, so that no releases will occur. (2) loss of electricity: a SBO1 situation develops as at 4.10m.;
- 5.00 m+NAP: possible release of contaminated water from the damaged hot drain tanks, that start floating;
- 5.25 m+NAP: loss of the AC part of the HFR Uninterruptible Power System for decay heat removal: SBO2 situation (Chapter 5 of this report) develops. Again approx. 600 hours is available to take remedial actions, before fuel uncovery occurs.
- 7.00 m+NAP: breach of containment:
    - water inside containment that has in principle no additional consequences as cooling is completely passive and stability of the basin is not immediately threatened;
    - loss of containment; possible small release.

Consequences

The main consequence is loss of active cooling, with a grace time of approx. 600 hours, under the condition that the convection valves are opened. After 600 hours the spent fuel will become uncovered. Next to this, minor airborne releases could be possible from the containment (from specific isotope production facilities, for which melting cannot be precluded conclusively based on existing thermal-hydraulic calculations) or releases of contaminated water from damaged waste tanks.

In case the convection valves are not opened, approx. 8 to 12 hours grace time is available before heating up of the core due to core uncovery occurs.

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[15] Worst case scenario, resulting is minimum time period available (Chapter 5 of this report).

3.3.1.3 Flooding of the HCL-RL, ground level 3.90 m

Effect

For the HCL-RL, the following sequence occurs if the water level rises:

- 3.90 m+NAP: the cellar floods. The ventilation of the “pluggennesten16” is blocked. This has no further consequences.
- 3.90 m+NAP: the storage pool floods. This has no consequences, because the drums are water tight.
- 3.90 m+NAP: the “pluggennesten” flood. This has no consequences, because the configuration and procedures prevent recriticality (Section 6.3.4 of this report).
- 3.90 m+NAP: the second storage pool with “filterbussen” floods. This has no consequences, because the pool is already filled with water.
- 4.10 m+NAP: the water level reaches the normal and emergency bus bars17 in the HCL-RL, resulting in loss of power. The consequence is that the under-pressure in the HCL-RL building cannot be maintained.
- 4.15 m+NAP: transformer ECN 6 has short circuit. This has no effect, because the electricity was cut off at a lower water level.
- 5.90 m+NAP: the water level blocks the ventilation channels in the HCL-RL. This has no effect, because the ventilation was blocked at a lower water level.

Cliff edges

One cliff edge water level can be distinguished for the HCL-RL:

- 4.10 m+NAP: loss of under-pressure, which could result in loss of containment.

Consequences

The cells are isolated, but when ventilation fails a leakage can develop at the openings of the manipulators although this leakage is counteracted by the elimination of the under-pressure regime in other parts of the building. In this situation it could be possible that radioactive substances are leaking to other places in the building. At relatively high wind speeds along the building, radioactive substances can

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[16] Storage facility consisting of vertical storage pipes in the floor.

[17] The Emergency Power supply itself also fails at this flooding level

be blown through the air outside the building, but the radiological consequences would still be relatively small. A proposition of the exact radiological consequences is not available, quantification of possible releases are still to be provided (recommendation in Chapter 5 of this report).

3.3.1.4 Flooding of HCL-MPF, ground level 3.90 m

Effects

For the HCL-MPF, the following sequence occurs if the water level rises:

- 3.90 m+NAP: the cellar floods. The liquid waste tanks will float as they are not bolted down, resulting in pipe break and ultimately release of the contents of the tanks.
- 4.85 m+NAP: transformer ECN 3 has short circuit, resulting in loss of electric power supply in the HCL-MPF as the Emergency Power supply has already failed at 4.10 m+NAP. The consequence is that under-pressure of the HCL-MPF building cannot be maintained; this affects the containment function.
- 4.90 m+NAP: water flows into the hot cells. This results in release of the contents of the cells.
- 6.90 m+NAP: the technical area on the first floor floods, submerging the bus bar. This has no effect, because the electricity was cut off at a lower water level.
- 6.90 m+NAP: the emergency power floods. This has no additional consequence as the diesel building is already submerged at this level.

Cliff edges

Three cliff edge water levels can be distinguished for the HCL-MPF:

- 3.90 m+NAP: release of the contents of the liquid waste tanks;
- 4.85 m+NAP: loss of under-pressure, which could result in loss of containment function of the HCL-MPF;
- 4.90 m+NAP: release of the contents of the hot cells.

Consequences

The cells are isolated, but when ventilation fails a leakage can develop at the openings of the manipulators although this leakage is counteracted by the elimination of the under-pressure regime in other parts of the building. In this situation it could be possible that radioactive substances are leaking to other places in the building. At relatively high wind speeds along the building, radioactive substances can be blown through the air outside the building, but the radiological consequences would still be relatively small. A proposition of the exact radiological consequences is not available. Next to this contaminated water from the waste tanks may spill.

3.3.1.5 Flooding of Water treatment building, ground level 7.85 m

Effects

- 6.10 m+NAP: transformer ECN 4 has short circuit, resulting in loss of electric power supply in Decontamination Building. The consequence is that the under-pressure in the water treatment building cannot be maintained.
- 7.85 m+NAP: the basement floods. The contents of the sump (contaminated water) could be released

Cliff edges

- 6.10 m+NAP: no under-pressure.
- 7.85 m+NAP: release of sump contents.

Consequences

Flooding water contaminated with sump water may be spilled

3.3.1.6 Flooding of Water treatment building, ground level 7.85 m

Effects

For the Water Treatment Building, the following sequence occurs if the water level rises:

- 6.10 m+NAP: transformer ECN 4 has short circuit, resulting in loss of electric power supply in Water Treatment Building. The consequence is that the under-pressure in the water treatment building cannot be maintained.
- 7.85 m+NAP: the basins in the cellar flood. The contents will be released, but this will not give nuclear contamination to the environment because the basins contain clean water.
- 7.85 m+NAP: the sink with waste tanks floods. The waste tanks will float, resulting in pipe break and ultimately release of the contents of the tanks. Minor releases are possible.
- 8.35 m+NAP: the water level reaches the bus bar in the Water Treatment Building. This has no effect, because the electricity was cut off at a lower water level.

Cliff edges

Two cliff edge water levels can be distinguished for the water treatment:

- 6.10 m+NAP: no under-pressure.
- 7.85 m+NAP: release of waste tanks contents.

Consequences

Contaminated water from the waste tanks may be spilled.

3.3.1.7 Flooding of DWT, ground level 7.85 m

Effects

For the DWT storage building, the following sequence occurs if the water level rises:

- 6.10 m+NAP: transformer ECN 4 has short circuit, resulting in loss of electric power supply in the DWT storage building. The consequence is that the under-pressure in the DWT building cannot be maintained.
- 7.85 m+NAP: the cellar flood. The drums with filter resins are submerged. Since the drums are water tight, there is no contamination. The drums could start floating and get dispersed.
- 8.35 m+NAP: the water level reaches the bus bar in the DWT storage building. This has no effect, because the electricity was cut off at a lower water level.

Cliff edges

One cliff edge water level can be distinguished for the DWT:

- 6.10 m+NAP: no under-pressure. No radioactivity will escape (Section 5.5.1.4 of this report).

Consequences

None.

3.3.1 8 Flooding of the DWT Pipe Cleaning facility, ground level 7.85 m

Effects

For the DWT pipe cleaning facility, the following sequence occurs if the water level rises:

- 6.10 m+NAP: transformer ECN 4 has short circuit, resulting in loss of electric power supply in the DWT pipe cleaning facility building. The consequence is that the cleaning of pipes stops. This has no consequences.

- 8.85 m+NAP: the sludge tank underneath the centrifuge floods. The consequence is the release of its contents.

Cliff edges

One cliff edge water level can be distinguished for the pipe cleaning:

- 8.85 m+NAP: release of the contents of the sludge tank underneath the centrifuge. Minor release is possible.

Consequences

Major consequence is a possible spill of contaminated water.

3.3.1.9 Flooding of WSF, ground level 7.85 m

Effects

For the WSF building, the following sequence occurs if the water level rises:

- 6.10 m+NAP: transformer ECN 4 has short circuit, resulting in loss of electric power supply in the Waste Storage Building. The consequence is that the under-pressure in the WSF building cannot be maintained.
- 7.85 m+NAP: the pipes with drums in the cellar flood. The drums with high radioactive material will be submerged. The contents will remain subcritical (Section 6.3.4 of this report).
- 7.85 m+NAP: the ground floor floods. The contents of the damaged, corroded or leaking drums on the ground floor is released.
- 8.35 m+NAP: the water level reaches the bus bar in the WSF building. This has no effect, because the electricity was cut off at a lower water level.

Cliff edges

Two cliff edge water levels can be distinguished for the WSF:

- 6.10 m+NAP: no under-pressure.
- 7.85 m+NAP: release of the contents of the leaking or damaged drums into the water.

Consequences

Release of from leaking / damaged drums.

3.3.1.10 Flooding of STEK, ground level 7.85 m

Effects

For the STEK building, the following sequence occurs if the water level rises:

- 6.10 m+NAP: transformer ECN 4 has short circuit, resulting in loss of electric power supply in the STEK building. The consequence is that the ventilation fails. This has no consequences.
- 6.66 m+NAP: the cellar floods. The cellar is empty, so there are no consequences
- 6.66 m+NAP: the ground level floods and nuclear storage containers become submerged. The containers are water tight and there are no consequences.
- 7.16 m+NAP: the water level reaches the bus bar in the STEK building. This has no effect, because the electricity was cut off at a lower water level.

Cliff edges

There are no cliff edge water levels for the STEK, as there are no consequences.

Consequences

None.

3.3.1.11 Flooding of LFR, ground level 6.66 m

Effects

For the LFR building, the following sequence occurs if the water level rises:

- 6.10 m+NAP: transformer ECN 4 has short circuit, resulting in loss of electric power supply in the LFR building. The consequence is that radiation monitoring fails. Since the LFR is not operated anymore, the ventilation system is in the shutdown mode, including closed ventilation shut-off valves. So this has no consequences.
- 6.66 m+NAP: the cellar with the fuel storage flood, resulting in water entering the storage tubes. The fuel storage drums are submerged. Fuel (new as well as spent) does not become critical. Containers are watertight, so no possibility for releases exists.
- 7.16 m+NAP: the water level reaches the bus bar in the LFR building. This has no effect, because the electricity was already cut off at a lower water level.

Cliff edges

No cliff edge as there are no radiological consequences.

Consequences

None.

3.3.1.12 Flooding of the Telecommunication Centre, ground level 5.85 m

Effects

For the Telecommunication Centre in the General Laboratory building, the following sequence occurs if the water level rises:

- 5.85 m+NAP: the cellar of the General Laboratory floods. The telecommunication off site and on site fails. Communications off site could have failed much earlier given the cause of the flooding.
- 6.10 m+NAP: transformer ECN 4 has short circuit, resulting in loss of electric power supply in the General Laboratory building. This has no effect, because the telecommunication failed at a lower water level. Emergency power is not available as the Emergency power system failed at a much lower water level.

Cliff edges

One cliff edge water level can be distinguished for the Telecommunication Centre:

- 5.85 m+NAP: failure of the telecommunication.

Consequences

No radiological consequences, but communication will be severely hampered.

3.3.1.13 Flooding of Emergency Coordination Center in the Forum building, ground level 5.50m

Effects

For the Emergency Coordination Centre in the Forum building, the following sequence occurs if the water level rises:

- 5.50 m+NAP: the Forum building starts to flood and may be inaccessible.
- 5.50 m+NAP: the cellar floods, resulting in electricity failure.

- 6.10 m+NAP: transformer ECN 4 has short circuit, resulting in loss of electric power supply in the Forum building. This has no effect, because the electricity was already cut off at a lower water level.

Cliff edges

One cliff edge water level can be distinguished for the Emergency Coordination Centre in the Forum building:

- 5.50 m+NAP: ECC in the Forum building becomes unusable.

Consequences

Loss of ECC in the Forum building: loss of monitoring equipment.

3.3.1.14 Flooding of Emergency Coordination Center at JRC, ground level 4.40 m

Effects

For the Emergency Coordination Centre at JRC, the following sequence occurs if the water level rises:

- 4.40 m+NAP: the JRC building housing the ECC starts to flood and may be inaccessible. Main bus bar is flooded and the ECC becomes unavailable.
- 4.45 m+NAP: transformer Covidien 3 has short circuit, resulting in loss of electric power supply in the JRC building housing the ECC.

Cliff edges

One cliff edge water level can be distinguished for the Emergency Coordination Centre at JRC:

- 4.40 m+NAP: ECC at JRC becomes unusable.

Consequences

The consequences are loss of the ECC at JRC, and loss of monitoring equipment.

3.3.1.15 Flooding of Jaap Goedkoop Laboratory, ground level 3.90 m

Effects

For the Jaap Goedkoop Laboratory, the following sequence occurs if the water level rises:

- 3.90 m+NAP: the cellar floods, submerging the power supply for the laboratory. This results in loss of electric power supply in the laboratory. This has no nuclear consequences.

- 3.90 m+NAP: the cellar floods, submerging the communication equipment. The external communication fails.
- 3.90 m+NAP: the cellar with waste tanks floods. The waste tanks will float, that could result in pipe break and ultimately release of the contents of the tanks: loss of containment.
- 4.85 m+NAP: transformer ECN 3 has short circuit, resulting in loss of electric power supply in the Jaap Goedkoop Laboratory building. The consequence is that the under-pressure in the Jaap Goedkoop Laboratory building cannot be maintained: possible loss of containment of glove boxes and other material handling and storage.
- 9.90 m+NAP: the technical area on the second floor floods, submerging the bus bar. This has no additional effect, because the electricity was already cut off at a lower water level.

Cliff edges

Two cliff edge water levels can be distinguished for the Jaap Goedkoop Laboratory:

- 3.90 m+NAP: two effects: (1) communication fails and (2) release from waste tanks.
- 4.85 m+NAP: loss of containment function; possible releases from glove boxes and from storage/handling.

Consequences

Consequences are the possible release of contaminated water and airborne releases from glove boxes as a result of loss of under-pressure / ventilation.

3.3.1.16 Summary of effects of flooding

Storm surge high tide, wide spread flooding

It is very likely that in case of an extreme storm surge the Zijper Polder will be flooded, while the OLP itself is not affected (see Section 3.2 “Flood sources”). In case of such a flooding of the Zijper Polder, it is almost certain that the OLP loses all off-site power. The emergency diesel generators of the OLP can supply power for 42 hours before refuelling is necessary (Chapter 5 of this report). In this situation the OLP becomes an island and personnel, fuel and equipment must be supplied through the air or by boat. Buildings not connected to the Emergency Power system lose their (safety) function. Communication becomes severely hampered. The active cooling of the HFR by the secondary cooling water system stops immediately and the convection valves of the reactor vessel need to be opened in time to maintain decay heat removal. In this situation approximately 600 hours are available for remedial measures before the top the (spent) fuel becomes uncovered (Chapter 5 of this report).

After 42 hours the diesel fuel supplies are exhausted and a station black out situation arises. For those facilities, other than the HFR, that are connected to the Emergency Power supply the under-pressure needed to maintain confinement is lost, and small releases of radioactive materials can occur (Chapter 5 of this report).

When the OLP itself is flooded the above described situation with regard to confinement of the facilities after 42 hours develops immediately. In addition the waste tanks in HCL-MPF, JGL, HFR and DWT could float, resulting in pipe break and ultimately release of the contents of the tanks inside the buildings. Seeping/spreading of this water to the environment will cause small external releases. Also communication and supply of equipment, water, personnel and fuel will become increasingly difficult.

With rising water levels the stability of the buildings could decrease and ultimately they could collapse. The water levels needed for such a scenario are unknown and difficult to assess, as not only water height but also wave action and design of the building will play a role. Next to this, it is questionable if a collapse will increase the already described radiological consequences very much, as most scenarios describe maximum releases.

Heavy rain and pipe break

Consequences of both heavy rain or break of the secondary cooling system pipe, could be the same as for high tide and storm surges, but will be restricted to one building and the maximum water level will be limited and not jeopardise the stability of the buildings.

3.3.2 Measures which can be envisaged to increase robustness of the plant against flooding

- All existing procedures with respect to the impact of flooding will be reviewed. Procedures with respect to the identified bottlenecks will be adapted.
- The possibilities to replenish diesel fuel and potable water supply during flooding conditions will be investigated.
- The possibilities to improve availability of replacement staff in case of flooding conditions will be investigated.
- Securing the power supply of the Emergency Communication Centre ("INO room") under extreme conditions would strengthen the Emergency Response Organisation (ERO).
- Improve the flooding resistance of the Telecommunication Communication Centre.
- The use of autonomous wireless battery-based techniques on site and satellite-based communication systems for off-site and emergency voice and data communication would strengthen the ERO.

- Bolting down of waste tanks in the HCL-MPF, JGL and HFR would improve the robustness of the waste tanks in case of flooding and or seismic events.
- Installation of external connections for auxiliary diesel generators for power supply to vital components would increase the margin in case of Station Black-out.
- (Additional) leakage detection systems should be installed in the HFR Reactor Outbuilding, HFR Primary Pump Building (PPG) and WSF.
- A leakage detection system should be installed in the Primary Pump Building.
- The possibility to seal the penetrations of the cabling duct between the Primary Pump Building and the Reactor Outbuilding to reduce leakage to the Reactor outbuilding will be evaluated.

4 Extreme weather conditions

This chapter describes the assessment of the design basis of the OLP with respect to extreme weather conditions. The weather conditions and effects taken into account are:

- high and low water temperature of the Noordhollands Kanaal;
- formation of ice in the Noordhollands Kanaal;
- extremely high and low air temperature;
- extremely high wind, wind gusts and whirlwinds;
- wind missiles and hail;
- salt deposition;
- heavy rainfall;
- heavy snowfall;
- lightning strike;
- credible combinations of the weather conditions mentioned above.

The assessment has been carried out on the OLP facilities as described in Chapter 1, as well as the Emergency Coordination Centre (ECC) and the emergency power supply.

Each facility may consist of several buildings. Table 4-1 of section 4.2.1 describes which buildings belong to each facility and specifies which buildings are not in scope of this assessment. The ECC is not formally known as a facility, but is listed here because of its importance in case of an emergency. The ECC can be located in the upstairs conference room ("Grote Vergaderzaal") of the Forum building and at JRC. Buildings not in scope are those that fulfil no safety function. These buildings are indicated with 'N' in the column 'Safety function' of Table 4-1. For a brief overview of safety functions and SSCs of each facility, see Appendix A.6.

4.1 Design basis

4.1.1 Reassessment of weather conditions used as design basis

This chapter describes the weather conditions that constitute the design basis of the buildings in scope. This is done by evaluation of the original building strength calculations. In a number of cases where building data could not be retrieved unambiguously, the prevailing weather conditions are estimated by engineering judgment. In case of engineering judgment, a building with known properties is considered to represent other buildings built in the same period. In Table 4-1 these cases are indicated.

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

This section describes the weather conditions and the level of resistance of the SSCs of the OLP against them. The selection of buildings in scope has been identified in the previous chapter. In Table 4-1, the General Laboratory building of ECN is included as part of the ECC, because in its basement ICT servers are located. These servers play an important role in telecommunication, during normal operation and emergency situations.

Water temperature of the Noordhollands Kanaal

<u>HFR</u>

Water temperature of the Noordhollands Kanaal is of concern for cooling of the high flux reactor (HFR), which is part of the HFR facility. Water for the secondary cooling system of the reactor is taken from this canal through a concrete pipeline of 120 cm diameter and 1.5 km in length. A minimum or maximum allowable secondary intake water temperature is not defined as design basis. The secondary outlet water temperature may not exceed 40 °C (environmental limit), which would correspond with a water intake temperature of approximately 25 °C. Low water temperatures (below 0 °C) result in ice formation on the Noordhollands Kanaal, which is handled separately.

<u>Other facilities</u>

For none of the remaining facilities water temperature of the Noordhollands Kanaal is of concern.

Formation of ice on the Noordhollands Kanaal

Ice formation is not defined as design basis. In general, formation of ice is a common and relatively slow process that can be anticipated adequately. In case of formation of ice on the Noordhollands Kanaal, the cooling water intake may become unavailable. Since there is no alternative system to supply cooling water to the reactor, the reactor will be shut down according to procedure if this occurs.

If the cooling water intake becomes unavailable at low water temperatures due to formation of ice in the canal, the reactor is shut down.

Air temperature

For none of the facilities a maximum outside air temperature is defined as design basis.

HFR

For the Reactor building, a minimum wall temperature of –10 °C is defined as an operating limit under specific circumstances. If the temperature of the intake air of the heaters of the ventilation system in the Reactor Building becomes lower than 1 °C and if the water outlet temperature of the heater responsible for warming up this intake air is less than 2.5 °C then ventilation is switched off and reactor power automatically decreases to 1 MW in order to prevent freezing of the heaters. If this situation persists, the reactor must eventually be shut down completely.

The transition plate of the containment shell [18] of the reactor building is designed to have sufficient strength down to temperatures of −20 °C, as this plate cannot be replaced in case of brittle rupture.

The coolant of the diesel generators in their separate building is cooled by ambient air. To prevent freezing of the coolant, additives are used to increase the frost resistance down to approximately −35 °C. The fuel inventory for the diesel generators is capable of withstanding temperatures down to −20 °C. The fuel is stored outside the diesel building in an underground tank.

The batteries in the Primary Pump Building that power the decay heat pumps in case of an emergency, are located in rooms with conditioned atmosphere. If room heating fails, it will take at least 24 hours before room temperatures reach below 0 °C. This is long enough, considering the battery standby time of 2 hours.

Other facilities

For other facilities besides the HFR, outside air temperature has not been retrieved as design basis.

[18] Transition plate: this plate is located where the cylindrical shell emerges from the concrete foundation.

Facility code.	Facility name	Building name	Safety function (#)	Allowable Wind speed [m/s]	Allowable roof load [kg/m^2]	Height of roof edge [m]
HCL	Hot Cell Laboratory	LSO	Cn, R	40	100	0.4 *
WSF	Waste Storage Facility	Radioactive waste building	Cn	40 **	100 **	0.1 *
JGL	Jaap Goedkoop Laboratory	Jaap Goedkoop Laboratory	Cn	42	56	0.2
STEK	STEK hall	STEK hall	Cn	40	100	0.3 *
HFR	High Flux Reactor	Reactor building	Cn, Co, R	61	n.a.	n.a.
		Reactor outbuilding	Co, R	40	100	0.4 **
		Primary pump building	Cn, Co, R	40 **	100 **	0.2
		Storage tanks	Cn, Co, R	40 **	100 **	0.2
		Storage building	N	40 **	100 **	0.2
		Secondary pump building	U	40 **	100	0.4 **
		Chlorine bleach building	N	40 **	100	0.4 **
		Air treatment building	Cn	40	100	0.4 **
		Security lodge	N	n.a.	n.a.	n.a.
		NDE building	N	n.a.	n.a.	n.a.
LFR	Low Flux Reactor	Fermi building	Cn, Co, R	40	100	0.3 *
DWT	Decontamination and Waste Treatment	Decontamination building	Cn	40	100	0.3 *
		Water treatment building	Cn	40 **	100 **	0.9 *
		Waste treatment and recycling building	Cn	40 ** 39	100 ** 50	0.1 *
ECC	Emergency Coordination Centre	Forum	SAM	40	50	0.2 *
		Joint Research Centre (JRC)	SAM	40 **	100 **	0.1
		General laboratory	SAM	40	50	0.5 *
EPS	Emergency Power Supply	Emergency power building***	Co	40	100	0.4 **

(#) Cn = confinement, Co = cooling, R = reactivity control, U = ultimate heat sink, SAM = Severe Accident Management, N = none, * Maximum height of roof edge, based on detailed measurement of roof height performed by ECN. Roof may consist of several sections and / or be a gable roof, ** Value adopted from the LSO building based on engineering judgment. *** This station also supplies electricity to the HCL, DWT, STEK and LFR.

Table 4-1: Overview of potential cliff edges

Wind

Table 4-1 indicates the wind speed resistance for all buildings in scope. These calculated wind speeds are derived from thrust values stated in building strength calculations using the Bernoulli formula $F = ½ρV^2c$ assuming $ρ = 1.25$ kg/m^3 and form factor $c = 1$. Note that these calculated wind speeds are merely indicative and intended to provide a comparative overview. For the exact design basis figures and units, see the indicated references.

The wind speeds are conservatively considered as potential cliff edges in the consequence analysis of Section 4.2.1. *It must be noted that exceeding a potential cliff edge not necessarily leads to consequences; only the possibility exists.*

Appendix A.7 gives an impression of the wind speeds and the corresponding Beaufort number.

HFR

The Reactor Building is able to withstand wind speeds of up to 61 m/s. The stack of the Air Treatment Building with a height of 40 m above ground level is designed to withstand wind speeds of at least 47 m/s. The design of this building itself is based on a thrust value due to wind of 100 kg/m^2, which corresponds with an approximate wind speed of 40 m/s.

HCL

The HCL facility (LSO complex) was designed to withstand a thrust value of 100 kg/m^2 due to wind which corresponds with an approximate wind speed of 40 m/s. Wind load is addressed in the most recent 10-EVA, where it was indicated that the stability of the HCL needs to be reassessed.

Other facilities

For other facilities the maximum allowable calculated wind speeds are presented in Table 4-1.

Wind missiles and hail

Wind missiles are projectiles propelled by extreme winds. Hail is defined as precipitation in the form of spherical or irregular pellets of ice larger than 5 millimetres in diameter. The possible effect due to wind missiles is considered to cover that of hail. Therefore, hail is not analysed separately.

HFR

According to the HFR Safety Report, wind missiles may occur during storms that by itself have a devastating effect. Also, a loosened wind turbine wing could cause damage. For the first wind mill on the OLP, the HAT, the problem of loosening turbine wings was investigated, and it was found that no measures were necessary for the HFR, but instead for the Forum with its canteen facilities.

Other facilities

For none of the other facilities, wind missiles and hail are considered as design basis. Wind missiles may cause windows to break and other structures to be damaged with loss of containment as the ultimate consequence.

Salt deposition

Deposition of salt on structures is a normal phenomenon in coastal areas. A side effect that can be caused by dry wind is the accelerated deposition of salt (carried by air from the North Sea). In general, if salt deposits on electrical components, this may lead to loss of offsite power. Since external power is supplied to the OLP by ground cables no vital electrical components are exposed directly to the outside air. However, accelerated corrosion of (non-safety related) equipment inside the GBD building has been reported due to salt that allegedly entered the building via air ducts. To avoid malfunction of this equipment due to salt-induced corrosion regular maintenance is carried out.

In conclusion, salt deposition is not considered a concern for the facilities and is not defined as design basis.

Rainfall

Extreme rainfall may induce an additional load on building roofs [19]. If building roofs have raised edges, water may accumulate on top of these buildings if all drainage pipes are blocked. This water causes

[19] Extreme rainfall may also lead to flooding. This topic is covered in Chapter 3 of the final report.

additional load on the civil structure. As part of this assessment, it is assumed that extreme rainfall will last for 48 hours in combination with blocked roof drainage. Roof edge heights are given in Table 4-1.

For none of the facilities a maximum amount of rainfall is defined as design basis. The maximum allowable roof loads presented in Table 4-1 are conservatively considered as potential cliff edges in the consequence analysis of section 4.2.1.

It must be noted that exceeding a potential cliff edge not necessarily leads to consequences, only the possibility exists.

HFR

For the Reactor building (which has a spherical dome), water accumulation due to heavy rainfall can be ruled out. For the other buildings belonging to the HFR facility, allowable roof loads are presented in Table 4-1.

Other facilities

For buildings of the remaining facilities, the maximum load due to water accumulation resulting from rainfall is determined by the height of the roof edges. Table 4-1 presents the allowable roof loads for all remaining buildings in scope. Allowable roof loads are often specified as “variable” loads. In some cases, allowable load due to snowfall is specified separately. Allowable loads due to water are not specified in building calculations. Allowable roof loads of the various buildings are further discussed in the following subsection “Snowfall”.

Snowfall

Heavy snowfall may lead to accumulation of snow on roofs, inducing a load on the civil structure. Appendix A.8 gives an overview of the building standards adopted with regard to snow load on the roofs. Many buildings are designed according to TGB 1955 that prescribes a minimum resistance against snow load of 50 kg/m^2 for roofs with an angle of 30° or less. Because it is irrelevant whether a roof load is caused by water or snow, the maximum allowable roof load presented in Table 4-1 is considered as potential cliff edge in later consequence analysis.

HFR

The reactor building has a spherical roof shape, which prevents 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. For other HFR buildings, the maximum allowable roof load is presented in Table 4-1.

<u>Other facilities</u>

The maximum allowable roof load belonging to other facilities varies, see Table 4-1.

Lightning

Lightning strike may have an impact on electrical systems such as the I&C of the HFR. Another phenomenon that may induce effects comparable to lighting is EMP due to geomagnetic storms [20]. By engineering judgment, the effects on I&C that can be expected from EMP due to sun burst are considered smaller or of the same magnitude compared to those caused by lightning strike.

The buildings of the OLP are equipped with lightning protection according to standard NEN 1014. This standard is valid for all building installations that have been constructed before February 1, 2009 and has four security classes: LP1 to LP4. A higher LP (lightning protection) number indicates that more advanced protection equipment is applied against lighting strike. The classification is based on the consequences that can be expected due to lightning strike. In Appendix A.8 the LP class of each building is presented.

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

In many cases, extreme weather conditions are not included in the original design basis. It is beyond the scope of this report to postulate and validate missing specifications, but it is recommended to do so for buildings that fulfil a safety function in a subsequent study, based on the credible consequences of exceeding a potential cliff edge.

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

This section describes the nature of most extreme weather conditions and the expected frequency of occurrence.

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[20] Geomagnetic storms (or 'sun bursts') are created when the Earth's magnetic field captures ionized particles carried by the solar wind due to coronal mass ejections or coronal holes at the sun.

Water temperature of the Noordhollands Kanaal

Since water intake temperature is not defined as design basis, no frequency of occurrence of unacceptable values is produced. The only limiting environmental water temperature is that of the secondary outlet water, which may not exceed 40 °C (environmental limit). To avoid potential exceedance due to high intake water temperature, the reactor power is decreased accordingly.

However, to give an impression, water temperature in the Noordhollands Kanaal at the location of Petten will be determined by the temperature of water coming from the hinterland, and cooler sea water.

The maximum water temperature in the hinterland (River Rhine at Lobith) is calculated to be approximately 28.3 °C once every 1,000 years (10^{-3} per year), whereas a water temperature of 25 °C at that location may occur every two years. More towards the coast, water temperatures decrease. Measurements performed at Zaandam between 1988 and 1991 show water temperatures ranging from 4.4 °C to 22.5 °C [21]. At Den Helder, the water temperature is rarely more than 20 °C on average.

At the location of the HFR cooling water intake, a water temperature of 24 °C has been recorded in June 2004. At that time, the maximum allowable outlet water temperature was 34 °C instead of 40 °C currently and reactor power was temporarily reduced in order to prevent exceeding of the allowable outlet water temperature. After this event, the allowable outlet water temperature was increased from 34 °C to 40 °C. In the history of the HFR, this is the only case in which reactor power was reduced in relation with outlet water temperature. In the current situation, an inlet water temperature of 24 °C would probably not lead to reduction of reactor power.

Considering a maximum allowable water intake temperature of approximately 25 °C and full reactor power operation, it is credible that reactor power is reduced occasionally in order to maintain water outlet temperature below the operating criterion of 40 °C.

Formation of ice on the Noordhollands Kanaal

Formation of ice on the Noordhollands Kanaal is a common process that can be expected every year. Severe ice formation occurs less frequently. If this would lead to unavailability of the cooling water intake, the reactor is shut down according to procedure.

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[21] Data adopted from www.waterbase.nl.

Air temperature

Research has shown that maximum expected air temperature in the Netherlands might increase to approximately 40 °C in the next 100 years. The highest measured air temperature since 1901 is 38.6 °C (Warnsveld, 1947, source: KNMI). The lowest air temperature with a return period of 100 years is approximately –20 °C. However, lower temperatures have been recorded; –27.4 °C (Winterswijk, 1942.)

As shown, the presented historical extreme temperatures occurred at inland locations. For Petten, located near the sea, these extremes are dampened by the North Sea water. Between 1906 and 2011 the maximum air temperature measured in Den Helder was 33.9 °C. The minimum air temperature measured was –18.8 °C (source: KNMI). Considering these figures, it is credible, yet very rare that air temperature will go down to –20 °C.

Wind

For wind types, the following subdivision is used:

- Extreme wind speed (10 minutes average);
- Wind gusts (usually less than 20 seconds);
- Whirlwinds (unpredictable).

Extreme wind speeds

Research shows that the maximum wind speed (hourly average) that can be expected once every 10,000 years is approximately 38 m/s [22]. Instead of the hourly average wind speed, the 10 minutes average wind speed is sometimes used as the design basis of structures. The 10 minutes average is calculated as 130% of the hourly average (source: KNMI) which amounts to approximately 50 m/s. For the buildings at OLP, in many cases a thrust pressure is given to quantify resistance against loads due to wind.

Wind gusts

A wind gust (“windstoot”, “windvlaag”) is a sudden, brief increase in speed of the wind. The duration of a gust is usually less than 20 seconds. KNMI has determined that the maximum wind gust is roughly 1.5

[22] Instead of the hourly average wind speed, the 10 minutes average wind speed is sometimes used as the design basis of structures. The 10 minutes average is calculated as 130% of the hourly average (source: KNMI) which amounts to approximately 50 m/s (180 km/h). For the buildings at OLP, in many cases a thrust pressure is given to quantify resistance against loads due to wind.

times the maximum hourly average wind speed. At a wind speed of 38 m/s, this results in a maximum wind gust of 57 m/s once every 10,000 years on average. The highest wind gust measured near Petten between 1961 and 2007 is approximately 44 m/s.

Whirlwind

A whirlwind ("windhoos") has 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. In general, little statistical data are available on whirlwinds. The highest wind speed that was observed by a monitoring station due to a whirlwind in the Netherlands is 56 m/s. On average, each year about two whirlwinds cause some damage to the infrastructure somewhere in the Netherlands, over an area of one square km. It is estimated that for a random location in the Netherlands, the risk of damage by a whirlwind is 10^{-5} per year.

For further evaluation of the structural building integrity, an enveloping wind speed of 57 m/s is adopted, which covers that of a whirlwind.

Wind missiles and hail

Extreme winds such as wind gusts and whirlwinds may facilitate wind missiles. Therefore, the average frequency of occurrence of wind missiles is linked to the return period of these extreme winds. Hail is a relatively common phenomenon. In the Netherlands, hail with a pellet diameter of 2 cm or more occurs on average five times a year (source: KNMI).

Salt deposition

Dry storms may facilitate salt deposition, but a frequency of occurrence of salt deposition cannot be produced. In general, it is a relatively slow process that can be anticipated adequately.

Rainfall

The statistics of the KNMI show that for the area of Petten once every thousand years (10^{-3} per year) 117 mm of rainfall can be expected within a period of 24 hours. For the same conditions, 133 mm of rainfall can be expected within 48 hours [23]. If after 48 hours rain continues to fall and drainage remains blocked,

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[23] If after 48 hours rain continues to fall and drainage remains blocked, the water level on top of buildings may rise up to a level of 176 mm within 9 days, the longest period of extreme rain analyzed by KNMI.

the water level on top of buildings may rise up to a level of 176 mm within 9 days, the longest period of extreme rain analysed by KNMI.

The largest amount of rainfall within 48 hours measured in the vicinity of Petten was approximately 123 mm. The measurement covers the period from 1951 to 2011 (source: KNMI).

Snowfall

On average, approximately 25 cm of snowfall occurs once every 50 years at the location of Petten. However, on March 3, 2005, 48 cm of snow was measured near Petten. The largest snow depth measured since 1951 in the Netherlands is 59 cm (Zweelo, Drenthe) on March 5, 2005 (source: KNMI). This amount of snow would cause a load [24] of approximately 60 kg/m^2. It must be noted that, if snowfall occurs over 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. Therefore, 0.8 m of fresh snow is assumed conservatively. As a result, the maximum expected load on roofs due to snow accumulation is approximately 80 kg/m^2.

Severe snow storms can be expected in the Dutch climate but do not occur frequently. No statistical data on snow storms are available from the KNMI. However, the last severe snow storm was recorded in 1979. Other previous storms occurred during the winters of 1958 and 1963. It is estimated that for a period of 5 years, a snow storm has a probability of occurrence of between 25 to 50 %.

Lightning

The climatology of lightning is limited. The average impact frequency in the Netherlands is 2 to 3 hits per square km per year. For the location of Petten, 0.3 to 0.8 lightning impacts per square km per year can be expected. Consequences of a lighting strike may vary, depending on various factors such as intensity, exact location of impact and the effectiveness of the lightning protection system.

It must be noted that the lightning frequency mentioned in this paragraph is merely indicative. No relation exists between this frequency and the adopted level of lightning protection for the various buildings (Appendix A.8) which is based on estimated consequences.

[24] The density of fresh snow is assumed to be 100 kg/m^3.

4.1.1.4 Consideration of potential combination of weather conditions

The following combinations of extreme weather conditions are considered the most realistic:

1. High air temperature + high water temperature;
2. Low air temperature + low water temperature + snow;
3. Snow + extreme wind;
4. Extreme wind + extreme rainfall + lightning.

Ad 1. Combined occurrence of high air and high water temperature is considered a temporary phenomenon because air temperature will decrease at night. At high water temperature, the reactor power will be decreased in order to meet the environmental criterion (40 °C outlet water temperature). Ultimately, the reactor might completely be shut down.

Ad 2. Combined occurrence of low air temperature, low water temperature and snow is in itself not considered a problem. However, a possible effect at low temperatures glazed frost which may affect logistics. In case of glazed frost and icy roads, a winter maintenance procedure is initiated. This procedure also deals with consequences of snowfall. Due to the absence of pylons on the OLP [25] there is no risk of accumulation of ice on power lines. At low air temperatures, low air humidity could present electrical problems for the ICT servers in the General Laboratory building. To prevent this, the servers are located in a controlled atmosphere.

Ad 3. Snow and wind combined may lead to clogging of air intake vents. This is may be an issue for The LSO and GBD buildings, the Air Treatment Building and the Jaap Goedkoop Laboratory [26]. Air intakes may also become clogged due to formation of ice as a result of low temperatures and high humidity. The consequences are similar to clogging due to snow. In case ice starts to form on the air intakes of the Air Treatment Building, operators remove it periodically. In addition, the emergency diesel generators in the Emergency power building depend on outside atmosphere for the intake of combustion air for their cooling. Air intakes may become clogged during a severe snowstorm, but since the structure is not leak tight sufficient air will be drawn into the building by the running diesel generators. Ice formation on the

[25] The 10 kV external power is supplied to OLP through ground cables. For more detail on external power supply, refer to Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink).

[26] Air intakes may also become clogged due to formation of ice as a result of low temperatures and high humidity. The consequences are similar to clogging due to snow.

exhaust air grids of the Emergency power building is of no concern to safety since hot air from the diesel generators is exhausted through these vents, which will melt any snow or ice present.

Ad 4. High winds, combined with extreme rainfall and lightning can be expected during a thunderstorm. Because the loads caused by these weather conditions are different, they will not reinforce each other's effect on the plant.

In general, it is advised to establish a measure to safely shut down nuclear facilities in case (a combination of) extreme weather conditions are expected, as well as a clear definition of extreme weather conditions.

4.1.1.5 Conclusion on the adequacy of protection against extreme weather conditions

The adequacy of protection for the separate and combined phenomena is discussed in the preceding sections. Potential combination of extreme weather conditions may induce effects that otherwise would not occur. However, these effects are mitigated either by design or by procedures. Adequacy of protection of the facilities against separate extreme weather conditions will be indicated by analysis of safety margins in the following section.

4.2 Evaluation of safety margins

4.2.1 Estimation of safety margin against extreme weather conditions

This paragraph contains an overview of the safety margins with regard to the various extreme weather conditions identified previously.

A safety margin is present if a positive difference exists between the allowable load (or: potential cliff edge, by design and/or by building standard, as listed in Appendix A.8) and the maximum load on buildings that can be expected as a result of the extreme weather conditions. Loads due to snowfall, rainfall and wind are derived from the weather conditions listed in the previous section. The remaining conditions (water temperature, air temperature, formation of ice and lightning strike) cannot be translated into cliff edges and are only qualitatively discussed (i.e. not quantified). In case a cliff edge can be exceeded, the building in question is conservatively assumed to fail [27] and potential consequences are identified. Table 4-2 presents the margin for resistance against severe snow, rain and wind.

For buildings for which no information was found, the data of the HCL is adopted. This is indicated with * in table 2.2. The HCL building is chosen as a reference because it was constructed as one of the first structures on the OLP and represents the building standards from the early period.

The roof load margin is determined by the presence of any roof edges:

- 133 kg/m^2 due to rainfall and accumulation of water (if roof edges are 0.13 m or more),
- 80 kg/m^2 due to snow.

All buildings, except for the Reactor building, have roof edges of various heights. Due to the spherical shape of the Reactor building, large snow accumulation is not expected. Furthermore, this building is designed to withstand an external pressure of 0.1 bar (approximately 1000 kg/m^2). No additional margin is specified.

The maximum wind speed is considered equal for all OLP buildings. The wind load margin is based on the maximum wind gust of 57 m/s, as defined in Section 4.1.1.3.

[27] Because the exact consequences of exceeding a cliff edge cannot be determined within the scope of this report, this approach is adopted.

Facility code.	Facility name	Building name	Rain load margin?	Snow load margin?	Wind speed margin?
HCL	Hot Cell Laboratory	LSO	N	Y	N
WSF	Waste Storage Facility	Radioactive waste building	- *	Y	N
JGL	Jaap Goedkoop Laboratory	Jaap Goedkoop Laboratory	N	N	N
STEK	STEK hall	STEK hall	N	Y	N
HFR	High Flux Reactor	Reactor building	n.a.	n.a.	Y
		Reactor outbuilding	N	Y	N
		Primary pump building	N *	Y *	N
		Storage tanks	N *	Y *	N
		Storage building	N *	Y *	N
		Secondary pump building	N *	Y	N
		Chlorine bleach building	N *	Y	N
		Air treatment building	N	Y	N
		Security lodge	n.a.	n.a.	n.a.
		NDE building	n.a.	n.a.	n.a.
LFR	Low Flux Reactor	Fermi building	N	Y	N
DWT	Decontamination and Waste Treatment	Decontamination building	N	Y	N
		Water treatment building	N *	Y *	N
		Waste treatment and recycling building	- * N	Y * N	N N
ECC	Emergency Coordination Centre	Forum	N	N	N
		Joint Research Centre (JRC)	- *	Y *	N
		General laboratory	N	Y	N
EPS	Emergency Power Supply	Emergency power building	N	Y	N

* Margin adopted from LSO building.

Table 4-2: Margins against extreme weather conditions.

Water temperature of the Noordhollands Kanaal

The maximum allowable secondary water outlet temperature is 40 °C. The corresponding water inlet temperature at full reactor power is approximately 25 °C. The actual water temperature in the Noordhollands Kanaal determines the *operational* margin. If the inlet water temperature is below 25 °C, the margin is positive. If inlet water temperature exceeds 25 °C, the reactor power will be reduced in order to maintain a water outlet temperature of ≤ 40 °C so the margin will not decrease below 1. No cliff edge exists.

Formation of ice on the Noordhollands Kanaal

No quantifiable margin for ice formation can be given. Formation of ice is not considered a problem because it does not affect availability of the pool cooling water. In case ice does block or damage the cooling water intake, cooling is possible using the pools, as described in relation with Loss of Ultimate Heat Sink (LUHS). In the most extreme case (long duration of unavailability of water intake) sufficient means of cooling are available to cope with this situation (see Chapter 5) so no cliff edge effects are expected to occur as a result of the available means of cooling.

Air temperature

For outside air temperature, sufficient margin exists with respect to extremely low temperatures (–20 °C). This has never occurred in the past century (where the lowest temperature was measured at –18.8 °C.) No cliff edge exists.

Wind

Considering that the HFR Reactor Building is able to withstand loads due to wind speeds up to 61 m/s, this building is not expected to fail due to either maximum credible wind gusts or whirlwinds.

The safety margin for wind resistance of the remaining OLP buildings is summed up in Table 4-2. If a positive margin is exists, this is indicated with “Y”, other cases are indicated with “N”. If no margin exists (i.e. the maximum weather condition equals the cliff edge value), this is indicated with “-“.

As stated previously, the maximum credible wind speed is taken as 57 m/s. All buildings, with the exception of the Reactor Building (design basis 61 m/s) that could be evaluated based on calculations, appear to be less resistant.

If a cliff edge is exceeded, it is conservatively assumed that a building fails. An indication of the consequence of failure of each individual building are described in Table 4-3. These consequences should be analysed into more detail.

Wind missiles and hail

No quantifiable margin for wind missiles and hail can be given. Possible consequences should be analysed into more detail.

Salt deposition

No quantifiable margin for salt deposition can be given.

Rainfall

Considering the KNMI statistics, 133 mm of rainfall can be expected within 48 hours. Depending on the height of a possible roof edge, this determines the induced load with a maximum of approximately 133 kg/m^2. Since almost all buildings have roof edges higher than 133 mm, 133 kg/m^2 is assumed for all buildings (with greater roof height) except for the Reactor Building because of its spherical roof.

Considering the allowable roof loads presented in Table 4-1, a maximum load of 133 kg/m^2 exceeds the cliff edge for most buildings. At this level, as a consequence, failure of the building is conservatively assumed, yet very unlikely given:

- the statistics on expected rainfall as presented in Section 4.1.1.3,
- the presence of multiple drain pipes.

This is supported by the fact that heavy rainfall has never led to structural damage to OLP buildings. The maximum amount of rain ever measured (123 mm in 48 hours, in September 1976) did not lead to any problems. In addition, 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 are cleared if necessary.

The failure mechanism due to excessive roof load differs from failure due to excessive wind load. For example, excessive wind load may induce more global effects (spreading of radioactive substances) compared to collapse of roofs due to excessive load. However, the consequence of building failure in terms of loss of safety functions due to excessive roof load is assumed comparable to failure due to excessive wind load. An indication of the consequences of failure of each individual building are described in Table 4-3.

The consequences should be analysed into more detail.

Snowfall

For most buildings, a load of 80 kg/m^2 due to snow will not result in exceeding the maximum permissible roof load. For these buildings, no cliff edge exists. The DWT waste recycling building may suffer some damage. Cliff edge type effects that possibly result are as yet uncertain.

In addition, to avoid hazardous situations, 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. It should be noted that buildings with the least resistance to snow should be monitored first.

Lightning

No quantifiable margin for lightning can be given.

Combination of weather conditions

No quantifiable margin for combinations of weather conditions can be given. However, none of the analysed combinations of extreme weather conditions is considered to lead to a loss of safety functions.

Facility code.	Facility name	Building name	Consequence
HCL	Hot Cell Laboratory	LSO	For HCL-RL: failure of ventilation, possibly followed by leaking of radioactive substances to other places in the building. At relatively high wind speeds along the building, radioactive substances can be blown through the air outside the building, but the radiological consequences would still be relatively small. For the HCL-MPF, in addition, water from the waste tanks may spill.
WSF	Waste Storage Facility	Radioactive waste building	Release of from leaking / damaged drums.
JGL	Jaap Goedkoop Laboratory	Jaap Goedkoop Laboratory	Possible release of contaminated water and airborne releases from glove boxes as a result of loss of under-pressure / ventilation.
STEK	STEK hall	STEK hall	No radiological consequences.
HFR	High Flux Reactor	Reactor building	None. Cliff edge is not exceeded.
		Reactor outbuilding	Total loss of power in the reactor building and in the control room, including the battery power for instrumentation and control.
		Primary pump building	Primary pumps and decay heat removal pumps may become unavailable. This means loss of safety functions for the HFR.
		Storage tanks	Partial loss of HFR cooling water. The reactor core may be damaged earlier. In the part 'loss of safety functions' this scenario is described in more detail.
		Storage building	None.
		Secondary pump building	Loss of secondary cooling flow.
		Chlorine bleach building	No loss of safety functions.
		Air treatment building	Loss of ventilation of Reactor building.
		Security lodge	No loss of safety functions. However, access to the HFR site may be hampered.
		NDE building	No loss of safety functions.
LFR	Low Flux Reactor	Fermi building	No radiological consequences.
DWT	Decontamination and Waste Treatment	Decontamination building	Possible spill of contaminated water and objects.
		Water treatment building	Contaminated water from the waste tanks may be spilled.
		Waste treatment and recycling building	None. Possible spill of contaminated water.
ECC	Emergency Coordination Centre	Forum	Loss of ECC in the Forum building, loss of monitoring equipment.
		Joint Research Centre (JRC)	Loss of ECC at JRC, loss of monitoring equipment.
		General laboratory	No radiological consequences, but communication will be severely disrupted.
EPS	Emergency Power Supply	Emergency power building	Failure of the diesel generators. If off-site power is already lost due to problems off-site the OLP, all electricity supply to the HFR is lost, with the exception of the battery powered instrumentation and control (3 hours grace time) and the decay-heat removal pumps (approx. 2 hour grace time).

Table 4-3 Indication of consequences due to exceeding of potential cliff edges.

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

The following recommendations were identified to increase robustness of the OLP facilities against extreme weather conditions:

- The uncertainty of the extreme weather margins can reduced by including certain missing weather conditions in the original design basis. Building specifications with respect to extreme weather conditions that not yet have been retrieved should be identified and subsequently margins and potential cliff edges should be re-evaluated.
- A procedure to safely shut down nuclear facilities in case of defined extreme weather conditions will be established.

5 Loss of electrical power and loss of ultimate heat sink

Impact of loss of electrical power supply and/or of loss of the ultimate heat sink should be dealt with for all nuclear facilities that exist at the NRG premises in Petten, for an increasing level of severity of these events and/or their combinations. For loss of electrical power supply this means that a distinction is made between:

- Loss of off-site power (LOOP);
- Meaning that there is no supply from external grid(s);
- Loss of off-site power and emergency AC power supply (SBO1);
- Meaning that in addition to the loss of external grid(s) connection also the ordinary emergency power system providing AC power is not available;
- Loss of all electrical power supply (SBO2);
- Meaning that in addition to SBO1 also the "permanent installed diverse back-up AC power sources" are not available (that is, the AC power part fed by batteries fails too).

For loss of the ultimate heat sink this means loss of cooling by the water of the Noordhollands kanaal. An alternate ultimate heat sink, as it is indicated by ENSREG, is not included in the designs of the separate facilities at the Petten site. Therefore this situation will not be dealt with. For the assessment, with regard to the facilities under consideration a distinction can be made between facilities for which all events apply, facilities for which only loss of electrical power supply applies and facilities that will not suffer by these events. Table 5-1 lists all facilities and the applicable events.

Based on Table 5-1 the following can be listed:

- The only facility that contains systems that have to be cooled and for which emergency power supply is required for this purpose, is the HFR; therefore all events and the indicated combinations apply for the HFR;
- For HCL, MPF, WSF, JGL and DWT which contain radioactive materials, no cooling is required. Because impact on the environment due to the events under consideration is the basic issue, only confinement has to be dealt with for these facilities. This is established by the (electrical powered) ventilation system that provides under-pressure. Loss of electrical power therefore poses a threat for confinement.

- LFR and STEK facilities are out of operation. The buildings (mainly STEK) contain amounts of low, intermediate and high level radioactive waste (LAVA, MAVA, HAVA) stored in containers pending their shipment to COVRA. Under-pressure is not applied; as due for their transportation these radioactive materials are properly stored. Therefore the events under consideration do not impact these facilities.
- For NRG-RE only the under-pressure and isolation of the fume hoods shall be maintained to assure confinement.

From this listing it can be concluded that for the safety margin assessment (stress test) for the facilities that are present at the NRG premises in Petten, for the issue loss of electrical power supply and loss of ultimate heat sink, this test can be limited to:

1. The assessment of the impact of LOOP-SBO, LUHS and their combination for the HFR reactor and the HFR spent fuel pool.
2. The assessment of the impact of LOOP-SBO on confinement of the facilities HCL, MPF, WSF, JGL and DWT (including glove boxes) and NRG-RE for its fume hoods.

In the following chapter these issues will be dealt with in conformity with the format proposed by ENSREG.

facility	LOOP	SBO1	SBO2	LUHS	combination
HFR	x	x	x	x	x
HCL	u	u	u		
MPF	u	u	u		
LFR					
WSF	u	u			
JGL	u	u			
STEK					
DWT	u	u			
GBD	u	u[28]			
x: event is applicable for the entire facility **u: event affects only under-pressure, so confinement of the building**					

Table 5-1: Facilities and their applicable events.

5.1 High Flux Reactor

5.1.1 Loss of electrical power

All off-site electric power supply to the site is lost. The off-site 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.

[28] Under-pressure fume hoods (Onderdruk zuurkasten)

- For loss of electrical power supply a distinction is made between;
- Loss of off-site power (LOOP);
- Meaning that there is no supply from external grid(s);
- Loss of off-site power and emergency AC power supply (SBO1);
- Meaning that in addition to the loss of external grid(s) connection also the ordinary emergency power system providing AC power is not available;
- Loss of all electrical power supply (SBO2);
- Meaning that in addition to SBO1 also the AC power backed-up by batteries is not available.

5.1.1.1 Loss of off-site power

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

Design provisions

Two separate feeder cables provide 10 kV supply from the public 50 kV grid to feed two circuits that are separated by several manual operated normally open (NO) circuit breakers.

In case of a long term loss of one feeder cable an alternative circuit can be established by connecting both circuits. The procedure to perform this has to be executed by Liander personnel (responsibility of public grid operator) that are on call. It can be done within a time frame of 3-4 hours. In this way long term emergency diesel generator operation of the OLP-EPS can be avoided.

Internal back up provisions

The HFR design includes an on-site emergency power system that is supplied by three diesel generator sets (A, B and C) that act as a double back up (3x100%) in case of failure. This means that by failure of A, C will take over its load. In case both A and C fail, diesel generator B, which basically is feeding the other facilities, will shed its load and will be connected to bus bar A. In this way emergency power for the HFR is restored.

In addition to this, the Uninterrupted Power System (UPS), consisting of two separate systems, provides power mainly to the following systems:

- Decay heat removal pump 1 and 2 (P-04-PMP, P-05-PMP)
- The reactor protection system, the control room annunciator system, the instrumentation in the HFR control room, the data-acquisition systems DAS-DACOS and the 110 V DC reactor control rectifiers DC-A and DC-B, in short the control systems.

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

Autonomy of power sources depends on capacities of fuel tanks for diesel generators, on battery capacities and on their loads.

From Chapter 1.3.5 the following can be summarized:

For the fuel of the Emergency Power System are available:

- one common main tank with a capacity of 10 m^3
- per diesel generator one day tank of 0,6 m^3 capacity.

Minimum amounts available are:

- 3,5 m^3 in the main tank;
- 0,48 m^3 per day tank;

Taking into account maximum real loads per diesel generator, the stand-by time per diesel generator is:

- Diesel generator A: 37 h;
- Diesel generator B: 33 h;
- Diesel generator C: 50 h;

By ENSREG timeframes of 24 and 72 hours are indicated for external delivery of portable light equipment and heavy materials respectively. For the HFR (generators A and C) the short stand by time is met; even there is a margin here of 13 h. The stand-by time of generator B is sufficient for a safe closing down of all operations in the connected NRG facilities and to evacuate their premises, also meeting the first ENSREG timeframe.

For the period of 72 hours it can be concluded from the above that for the HFR this will pose a problem, because after 37 hours generator A runs out of fuel. Diesel generator C will take over the load of diesel generator A and can run another 6 hours for the combined loads. Normally, then diesel generator B should replace diesel generator C, but it already ran out of fuel after 33 hours after start of the event.

In case of LOOP, when diesel generator C is shut down while diesel generator A is running, fuel consumption is less; this results in an additional stand-by time of diesel generator A of 5 hours. Based on the severity of the initiator of LOOP, operators may decide to run only diesel generator A to save fuel. In this situation both diesel generators B and C will be stopped as soon as possible. In case of an immediate stop of B and C, the stand-by time of diesel generator A, consuming the available main stock and its own day tank stock, will be extended to approx. 89 hours. This means a margin of 17 hours to the ENSREG requirement which was intended for a nuclear power plant in the first place. This can be additionally extended with 20 hours by a succession of another 10 hours operation of the restarted diesel generator C followed by 10 hours operation of, the restarted but load-shedded, diesel generator B.

Shut down of diesel generators at start of the event, what is required for this situation, will be difficult to judge and is currently not covered by procedures.

Another possibility to increase the stand-by time of diesel generators A and B, while C is shut down, is to increase the diesel fuel stock that should be available at a minimum. Increasing that amount to 8.1 m^3 will enlarge the stand-by time for diesel generator A up to over 72 hours. This means that the main tank should be almost charged to its full capacity.

For a minimum main stock of 8.6 m^3 and taking into account the 0.48 m^3 per day tank per diesel generator, all three engines will meet the 72 hours requirement.

Additionally, at the DWT-WSF facility diesel fuel is stored in a tank. It is normally in use for cleaning purposes and for the fork-lift trucks and has a minimum stock of 0.5 m^3 and a maximum of 3.2 m^3. The tank can be picked-up by a fork-lift truck and transported to the EPS building in case of an emergency in order to top-up the EPS main fuel tank.

5.1.1.2 Loss of off-site power and loss of the ordinary back-up AC power source (SBO1)

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

For this situation the NV system, backed-up by batteries, supplies AC power to the residual and decay heat removal pumps and other safety related equipment. The VZO system supplies uninterrupted power amongst others to the reactor protection system and the 110V DC rectifier system for control purposes. So for this situation emergency cooling and control systems stay in operation for the duration of the NV and VZO stand-by times.

Further, no usable permanently installed AC power sources are present at the OLP. The same applies to mobile units. At the OLP a fixed ECN 175 kVA emergency diesel generator is present that supplies power to the ECN/NRG data-centre and the fuel-cell floor in case their preferred power supply from the ECN-5 substation fails. It can however not be used to "feed up" from 0.4 kV to the 10 kV transformer in the ECN-5 substation because of electrical safety considerations.

5.1.1.2.2 Battery capacity, duration and possibilities to recharge batteries

From Chapter 1.3.5 the following can be summarized

NV system stand-by times:

- Battery NV-1: 34 minutes;
- Battery NV-2: 77 minutes.

Compared to the design stand-by time of 2x30 minutes there is a margin of 51 minutes here.

VZO system stand-by times:

- Battery VZO-1: 90 minutes;
- Battery VZO-2: 90 minutes.

Compared to the design stand-by time of 2x30 minutes there is a margin of 2 hours here.

As indicated in Chapter 1.3.5 the VZO-system supplies power to the 110 V DC rectifier system that supports the HFR control system for the 0.4 kV switchboards. This system consists of two rectifier-battery combinations feeding on a common distribution board. Each one has a stand-by time of approx. 5 hours. This means that for at least ten hours after the VZO batteries are exhausted 0,4 kV control actions on the 0,4 kV switchboards can still be executed. This enables the operator to preset components (mainly valves) in a safe mode which will be resumed in case electrical power is restored.

When the batteries described above are exhausted there are at the HFR no means to recharge them.

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 (SBO2)

In this situation only the batteries of the VZO system supply uninterrupted DC power to amongst others the reactor protection system and the 110V DC rectifier system for control purposes.

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

As indicated in 5.1.1.2.2 the VZO system will operate for approx. 3 hours.

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

At the HFR or the OLP there are neither actions foreseen nor procedures in place to get AC power from transportable off-site sources.

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

Not applicable; see argumentation under section 5.1.1.3.2.

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)

In the situation that loss of off-site power occurs, there will be no secondary cooling because the secondary cooling water system will fail as it is not powered by the emergency power system. This means that this situation is identical to the situation of loss of ultimate heat sink (LUHS), because all remaining systems that are important for cooling in this situation are still fed by the emergency power system. Therefore, for this situation reference is made to section 5.1.2.

In case LOOP is supplemented to SBO1 or SBO2, due to failure or non-availability of the emergency power support systems (AC and DC), these situations are identical to the combined LUHS-SBO1 and LUHS-SBO2 situations. These are dealt with in section 5.1.2

However, with regard to the availability of systems that rely on electrical power supply by the emergency power systems it can be concluded that the following periods of time are important:

<u>In case of LOOP:</u>

- 42 hours of selective, sequential, operation of the diesel generators before refuelling is necessary. This number is based on 37 hours of normal operation of generator A, extended by 5 hours if generator C is shut down at the onset of LOOP.

<u>In case of SBO1:</u>

- 34 min of operation of the decay heat removal pump 1 and the facilities cooling pumps;
- 77 min of operation of the decay heat removal pump 2 in sequence with decay heat removal pump 1;
- 3 hours for instrumentation and control system availability (including the reactor protection system).

In case of SBO2

- 3 hours for instrumentation and control system availability (including the reactor protection system).

5.1.1.4 Conclusion on the adequacy of protection against loss of electrical power

Because loss of electrical power in its basic event immediately results in loss of ultimate heat sink, the adequacy of measures relies on the impact of that event (LUHS), therefore reference has to be made to the elaboration of that event in 5.1.2.

Nevertheless, there are no additional measures envisaged to restore core cooling by restoration of or to prolong electrical power supply.

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

Potential cliff-edge effects

In case of loss of electrical power supply, the following potential cliff-edges have been identified:

For LOOP:

Running out of diesel fuel supply:

- Per diesel generator a limited volume of fuel stock is available. Electrical power supply can be provided by approx. 42 hours. After that period the situation will turn over into SBO1.

For SBO1/SBO2:

Exhaustion of batteries:

- Primary (circulation) flow will stop after approx. 2 (34+77min) hours (NV-systems). Switch over to cooling by natural convection of reactor pool water is envisaged;
- Loss of instrumentation and control systems (including the reactor protection system) after 3 hours (VZO system);
- No back up is provided.

For induced LUHS:

- See 5.1.2

Measures that can be envisaged to improve the robustness of the plant

Possible measures to improve the robustness of the plant in case of loss of electrical power supply

For LOOP:

- Improve operation of the secondary cooling water system;
- Supply emergency power to the low-capacity pump of this system; in this way LOOP will not automatically cause LUHS;
- Enlarge diesel fuel stocks;
- To meet the 72 hours operation requirement enlarge the minimum required volume of the main fuel stock up to 8.1 m^3 for operation of diesel generator A and B for that time frame, or to 8.6 m^3 for 72 hours of operation for all three engines;
- Provide sequence procedures;
- To meet the 72 hours operation requirement, provide procedures to shut down diesel generator C or diesel generator B and C to enlarge the stand-by time of diesel generator A in case it is running.

For SBO:

- Enlarge operation time of the control systems. Either enlarge battery capacity of the VZO system or provide alternative electrical (AC) power back-up for this system.

For induced LUHS:

- See 5.1.2

5.1.2 Loss of the ultimate heat sink

ENSREG definition of Loss of Ultimate heat Sink: "The connection with the primary ultimate heat sink for all safety and non-safety functions is lost. The site is isolated from delivery of heavy material for 72 hours by road, rail or waterways. Portable light equipment can arrive to the site from other locations after the first 24 hours."

ENSREG defines the ultimate heat sink (UHS) as "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, supplemented by an alternate UHS, for example a lake, a water table or the atmosphere".

Characteristic for those media is that they are almost infinite and are in their bulk hardly affected by the heat discharge (this means slight local temperature increase).

In this definition, for the HFR the ultimate heat sink is established by the Noordhollands kanaal for the supply of water and the North Sea for the discharge.

An alternate ultimate heat sink, as it is indicated by ENSREG, does not exist at the Petten site. However alternate cooling is provided by the (reactor) pools, this means that for their water supply the reactor pools can be considered as back up.

Introduction

Cooling systems

Heat produced by the reactor, by facilities, by spent fuel and by other radioactive materials that are stored in the pools, is removed by the reactor cooling system and finally transferred to the environment. The cooling system includes:

- *The primary cooling water system;* This system provides cooling of the reactor and transfers the heat to the secondary cooling water system;
- *The pool cooling water system;* This system provides cooling of the pools viz. reactor pool and pool 1 and pool 2 and transfers the heat released by the reactor vessel by convection, the heat produced by stored spent fuel and other radioactive materials as well as heat produced by facilities, to the secondary cooling water system;
- *The secondary cooling water system;* This system transfers heat from primary and pool cooling water system to the environment viz the North Sea as it takes water in from the Noordhollands kanaal;
- *The residual and decay heat removal system;* This system provides heat removal of the reactor core in case of shut down of the reactor and transfers it to the secondary cooling water system;
- *The alternate cooling provisions:*
    - *The convection valves;* In case of loss of primary system cooling due to loss of primary flow or loss of secondary system cooling, opening of these valves provides reactor core cooling by the pool water by natural convection;
    - *The pool water injection system;* This system injects by gravity water from the pool into the reactor vessel to provide reactor core cooling capacity in case of (LOCA) accidents;
    - *Public water supply system (PWN);* At accident management conditions water supply to the pool for refill can be provided by the public water supply system, via the hydrofoor system. This can be performed by pumps or partly gravity driven. For the latter a separate pipeline is installed;

- The facilities cooling systems. Divers facilities that are in place to perform irradiation tests are cooled by dedicated cooling systems, like:
    - *The primary facilities cooling system;* This system provides cooling of collimators of the horizontal beam tubes. Water is supplied by and discharged to the primary cooling water system;
    - *The pool facilities cooling system (BEKWS) ,the high pressure pool facilities cooling system (HD-BEKWS) and the extended pool facilities cooling system (U-BKEWS);* All systems cool enclosed facilities that are placed inside the reactor vessel and in the pool near the reactor vessel. Suction and discharge lines start and end respectively in the pools;
    - *The hydrofoor facilities cooling system;* This system provides the cooling of irradiation facilities outside the pools. Cooling water is discharged to the secondary water cooling system.

Electrical power supply

Electrical power is supplied by the normal power system to all cooling systems. To assure operation during loss of off-site power the following systems are fed by the emergency power system:

- The pool cooling water system;
- The residual and decay heat removal system; This system is additional backed by batteries (2 pumps, design discharge capacity for 30 min per pump);
- The hydrofoor system;
- One pump of the pool facilities cooling system supplying also water to the extended pool facilities cooling system in that situation;
- Two pumps of the high pressure pool facilities cooling system;
- Actuation of the pool water injection system.

Methodology

It is anticipated that, because the pools are partly connected, there will be some mixing of water of the three pools as well as exchange of heat between the pools during this event. Because dealing with both phenomena requires very detailed analyses a simplified, best estimate situation of complete mixing will be considered.

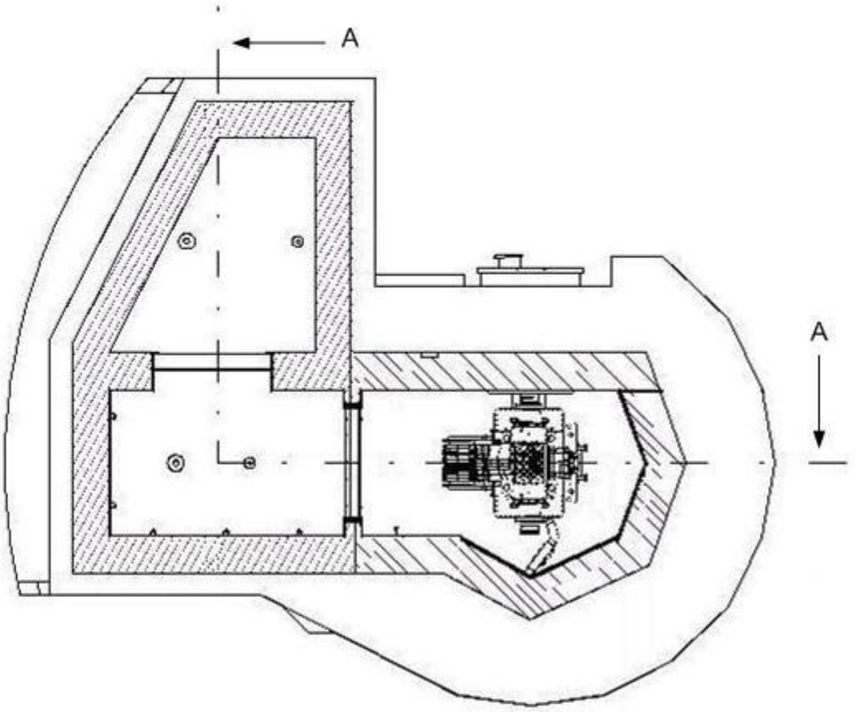
However, stratification cannot be excluded. This might result in the heating up of a smaller amount of water than in the complete mixing situation; so boiling-off will start earlier. As most conservative situation for this, this phenomenon is taken into account by consideration of complete isolation of the pools until boil-off starts.

a) Complete mixing:

Water of the reactor pool and both pools will mix completely, so they are considered to be one volume during the heating up period and the subsequent boil off. In this situation heat is provided by the combined heat sources viz. core and spent fuel. When the water level reaches the pool doors, see Figure 5-1, complete pool isolation is assumed. When the water level reaches the pool doors, see Figure 5-1, complete pool isolation is assumed.

b) Complete isolation:

This means that there will be no mixing of water of the reactor pool and both storage pools during the heating up period. Heat is provided by the separate sources viz. core and spent fuel to the related pools viz. reactor pool and storage pools respectively. When boiling off of the other pools also starts, the situation will become identical to the complete mixing situation from that point on. Therefor this situation will only be considered for the first part of the event, viz. until start of boil-off.



Cross section AA:

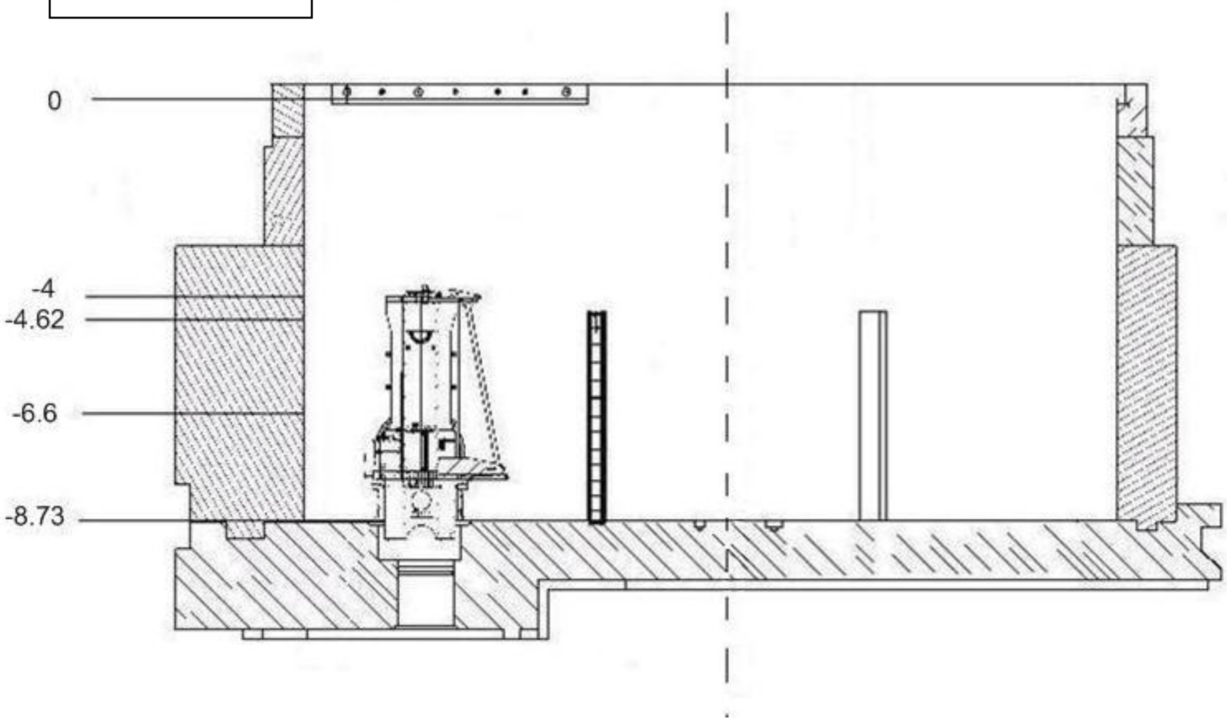


Figure 5-1: Outline of the pool configuration.

Further, ENSREG demands a separate elaboration of reactor cooling and spent fuel pool cooling. Because alternate cooling of the HFR is for its main part provided through (when replenished) and by the reactor pool and both storage pools, which are partly in common, see Figure 5-1, the following enveloping sequences are selected:

1. Cooling at complete mixing;
2. Cooling at complete isolation for the heating up period;
3. Sole spent fuel cooling at complete mixing;
4. Sole spent fuel cooling at complete isolation for the heating up period.

The selected sequences are defined as follows:

1. <u>Cooling at complete mixing</u>

   For this, best estimate, sequence it is assumed that at start of the event, present radioactive materials, producing heat, include:

   a) In the reactor: a full core at maximum power;
   b) In the storage pools: maximum allowed load of spent fuel and radioactive materials originating from experiments and the production of isotopes.

   The sequence itself starts at LUHS, initiating reactor shut down. Then depending of the sub-sequence (primary cooling system (temporary) available or not available) the first part of the decay heat of the core will be dissipated in this system. In the same time spent fuel and the other radioactive materials start heating up the pool water. Once further cooling through the primary system is no longer possible, the switch to cooling by pool water is made. Then the core will be cooled by natural convection of the pool water. From that time on, the sequence is characterized by heating up and later on by evaporation of the total pool water by heat produced by core and stored radioactive materials.

   Ultimately, water will be evaporated until uncovery of core or spent fuel occurs. This will apply when replenishment of water the pools fails.

2. <u>Cooling at complete isolation for the heating up period</u>

   For this sequence, it is also assumed that at start of the event there is a full core operating at maximum power and that the storage pools all loaded up to the allowed maximum. The sequence itself starts at LUHS, initiating reactor shut down. Then, depending of the sub-sequence (primary system (temporary) available or not available) the first part of the decay heat of the core will dissipate in the primary system. This will be followed by cooling (by natural convection) by water of the reactor pool. Because complete isolation applies, it is assumed that heating up of the separate pools is provided by the separate heat sources they contain. This means that decay heat from the core (whether or not after heat up of the primary system) will be dissipated only in the reactor pool. Heat produced by the radioactive materials, stored in the pools (main part is spent fuel) will be dissipated only in the storage pools. This will result in different timeframes for heating up and start of boiling off of the reactor pool versus the storage pools. At the moment that one of the pools starts boiling off, for the assessment, it is assumed that complete mixing starts.

Because complete isolation applies, it is anticipated that for the long run, when evaporation is ongoing, this situation and the situation of complete mixing will become identical. This is due to the fact that for both situations the same amounts of heat (GJ) and water (m^3) apply for heating up and evaporation. So for the remaining part of this situation reference is made to the complete mixing situation.

In this way, it is (conservatively) estimated what the shortest period of time will be that boil-off will start at which pool.

3. Sole spent fuel cooling at complete mixing

For this sequence it is assumed that for the maximum heat load of the storage pools the core is transferred to the pool and that for the remaining part the pool is loaded with spent fuel and other radioactive materials that originate from experiments and production of isotopes, to the maximum extend allowed.

The sequence itself starts with LUHS. Due to complete mixing the total amount of water of the three pools that is available will be heated up first. Then boil-off of the pool water starts until the water level reaches the separation doors. Only the remaining part of the storage pool will be used for continued boil-off until uncovery of the spent fuel occurs, in case replenishment of pool water occurs.

For the assessment it is assumed that, according to todays’ practice, storage of the core in the storage pool will be arranged 3 hours after reactor shut down.

4. Sole spent fuel cooling at complete isolation for the heating up period

Compared to the mixing situation at first there will be a smaller amount of water that will be heated up to saturation. Then boil-off start and so does mixing. From this moment it is assumed that the assessments become identical while the second part of the pool water starts heating up until saturation followed by start of the boil-off the entire pools. Therefore this situation will not be dealt with further.

In the following part of this chapter, both situations of cooling of the reactor will be dealt with. Because only the first parts of these sequences differ (anticipating a different period of time before evaporation starts) the complete mixing situation will be dealt with in detail; for the situation of complete isolation only the first part will be presented. The sole spent fuel cooling is presented in chapter 5.2.

Supporting calculations are listed in Appendix A.9.

5.1.2.1 Design provisions to prevent the loss of the primary ultimate heat sinkThe (primary) ultimate heat sink is the water provided by the Noordhollands kanaal that, after heat up, is discharged to the North Sea.

The interface between reactor systems and this ultimate heat sink is established by the secondary water cooling system. The secondary side of this system is an open connection channel from Noordhollands kanaal to the North Sea. To prevent blocking of the inlet and outlet of this system, the following features are included:

- A course inlet screen;
- A first strainer equipped with scraper chains (rotating scraper);
- A second (fine mesh) strainer also equipped with systems to remove among others small parts of seaweed;
- A chloride bleaching injection system, to prevent precipitation of mussels;
- A de-aeration point at the upper part of the inlet;
- A vacuum system to prevent accumulation of air in the upper part of the outlet.

To assure the operation of the secondary cooling water system this system includes:

- Three main cooling pumps with capacity of 1100 m^3/h to 1450 m^3/h depending on the number of pumps in operation. These pumps can be operated in parallel or provide back-up for each other;
- One small pump with capacity of 800 m^3/h, to be operated in combination with one main pump or all to provide maximum flow;
- To prevent corrosion and/or growth of algae, one main pump will be kept in operation during all plant conditions;
- Three heat exchangers in parallel to cool the primary cooling water system;
- One heat exchanger to cool the pool cooling water system.

5.1.2.2 Loss of the primary ultimate heat sink

Loss of the ultimate heat sink for the HFR means that there will be no supply of water from the Noordhollands kanaal or there will be no discharge to the North Sea.

Alternatives for these inlet and outlet are not included in the design.

5.1.2.2.1 Availability of an alternate heat sink

In terms of the ENSREG definition, the HFR is not equipped with an alternate ultimate heat sink. In spite of this, alternative cooling of the reactor can be provided in three ways:

1) By heating up the water of the primary cooling water system. For this situation circulation of the primary water through the core by the pumps of the residual and decay heat removal system is necessary. The temperature of the primary water will rise, heat release to the environment by convection, will occur (the latter phenomenon is not included in the assessment).
2) By application of pool water to cool the reactor. For this situation the convection valves (one above and one below the reactor core) are included in the design. Opening of these valves provides supply of pool water to the reactor. Because these valves are locked they have to be unblocked and opened local by hand; cooling will be performed by natural circulation of the pool water through the core. Ultimate, pool water will start to evaporate.
3) By combination of both preceding ways. For the situation that heating up of the primary cooling water system during the course of the event is not sufficient to store all produced heat switch over to cooling the reactor by the pool water can be decided.

Dedicated procedures to perform these actions in this situation are not available. However there exist procedures to drain and fill the pools and operate the convection valves.

In addition to this, heat producing facilities are installed in and outside the reactor vessel. These are cooled by the dedicated cooling systems like BEKWS, U-BEKWS and HD-BEKWS. Those cooling systems are partly driven by primary water flow or by own system pumps.

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

As indicated before two ultimate situations have to be assessed for the LUHS event, viz.:

a) The LUHS time constraints for complete mixing of the pools
b) The LUHS time constraints for complete isolation of the pools for the heating up period

A. <u>The LUHS time constraints for complete mixing of the pools</u>

In case of loss of the primary ultimate heat sink, which initiates reactor shut down, the combination as indicated in 5.1.2.2.1 will be applied. The sequence to consider includes:

1 Heating up of the primary system. Temperature rise from 50 to 100 ˚C of 100 m^3 of primary water; by decay heat of the core;

2 Insert pool water into the reactor. After step 1decay heat from the core will be dissipated in the pool water (320 m^3) until this starts evaporating;

3 Heating up of pool water. Heat from spent fuel and other radioactive materials that are stored in the pools already heats pool water during the period that decay heat of the reactor is dissipated in the primary system (step 1, until step 2 starts);

4 Evaporation of pool water down to 4m. Available is 160 m^3 .This is derived from the operation constraint that the water level of the pools may not decrease more than 4 Mathis 4 m refers to the minimum water layer needed for protection against radiation from stored fuel and targets in the pools. The moment that saturation of the water of the storage pool is reached, evaporation starts. From this moment on, heat dissipated in the storage pool contributes to the evaporation of the indicated 160 m^3 water;

5 Evaporation until separation of the pools. It is noted that lower levels of the pool water are acceptable; ultimately when no replenishment can be performed. Then, approx. 25 m^3 of "common" water above the pool doors will be evaporated. Then separation of pools by the doors occurs;

6 Evaporation of separate pools. Step 5 will be followed by evaporation of 35 m^3 of reactor pool water and 28 m^3 of storage pool water before core uncovery and spent fuel uncovery respectively occur;

7 Cooling of the facilities. During this sequence cooling of the facilities will be performed in parallel to the before mentioned steps: heat produced by the facilities is assumed to be 1 MW at a maximum. This amount is added to the reactor power at start of the assessment; decline of the decay heat of the fuel that is present in the facilities will be identical to the decay of the core.

Appendix A.9, section 9.1.2 presents the decay heat curve for the average core that is installed in the HFR. From this curve the heat produced by the core is derived. Absorption by the several water supplies is assessed. The results are listed in Table 5-2.

B. <u>The LUHS time constraints for complete isolation of the pools for the heating up period</u>

The sequence to deal with for this situation is, while the reactor is shut down due to LUHS:

1 Heating up the primary system (step 1 of A.).Temperature rise from 50 to 100 ˚C of primary water, by decay heat of the core.

2 Insert pool water into the reactor (step 2 of A.). Heating up 130 m^3 of the reactor pool water[29] from 37 to 100 ˚C; by decay heat of the core, at complete isolation conditions until evaporation starts.

3 Start of heating up of pool water. Temperature rise of 190 m^3 (106+84) of storage pool water from 37 to 100 ˚C; by decay heat of the stored spent fuel until evaporation. Heat from spent fuel and other radioactive materials that are stored in the pools already heats up pool water in parallel with step 1 and 2.

4 Start of evaporation. When boil-off starts it is assumed that this situation will become identical to the situation of complete mixing; therefor, for the remainder, reference is made to the assessment of that situation.

The results from calculations of Appendix A.9 are listed in Table 5-2

[29] It is assumed that small parts of the total of 155 m^3 of the pool inventory will not be affected by this process

Step	Sequence	Mixing situation			Isolation situation		
		Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)	Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)
1	T rise of primary system 50 to 100 ˚C	100	21	12	100	21	12
2	Heating up reactor pool water from 37 ˚C to 100 ˚C	320[30]	84	73	130	34	45
3	Heating of remaining pools, starts in parallel with 1&2	320	84	73	190	50	137
4	Evaporation of 4 m of water from the pools	160	346	464			
5	Evaporation to separated pools	25	54	536			
6	Evaporation in separate pool, until first fuel uncovery	29	62	709			
A	*Extra time if water is injected*	*105*	*227*	*330 extra*			

Table 5-2: results of heat dissipation by alternate reactor cooling including primary system heating

From this table the following can be concluded:

- Boil off will start after approx. 45 hours for the most conservative case and after 73 hours for the estimate based on the assumption of perfect mixing;
- The -4 m level will be reached after 464 hours;

[30] For the mixing situation the pool water volumes are combined to only one volume of 320 m^3 heated up by the combined sources (so step 2 and 3 are merged to one step)

- Fuel uncovery occurs after 709 hours in the spent fuel pool; more than one day later followed by core uncovery. This is due to the fact that at that time the heat output of both sources are almost equal, while the remaining amount of water in the reactor pool is larger;
- 330 hours of additional time are gained by supplying of extra water that already is stored on-site.

For the ENSREG 72 hours mark it is noted that for the most conservative case there will be a boil-off of approx. one day (72-45 hours) following the heat up of the primary system and the reactor pool water. The amount of boiled-off water for this period is approx. 10 m^3 this will result in a decrease of the water level of the pools of approx. 0.25 m.

For the best estimate situation boil-off starts after those 72 hours.

5.1.2.3 Loss of the primary ultimate heat sink and the alternate heat sink

Because in terms of the ENSREG the HFR is not equipped with an alternate ultimate heat sink this combination of events is not applicable for the HFR

5.1.2.3.1 External actions foreseen to prevent fuel degradation

From Table 5-2 it can be concluded that given the sequence constraints for a time period of 464 hours core cooling is assured; even it can be prolonged up to approx.700 hours until uncovery of the spent fuel occurs.

For this assessment basically it is assumed that sufficient pool water is available. In case the pool water content is less or exceeds the indicated limit (-4 m of top of water level), replenishment of water can be performed. Available stocks are:

- Deminwater storage: 15 + 30 m^3 water can be supplied by the demin water system (pump) to the pool cooling water system, so to the pool;
- Cooling water storage: 2 x 30m^3 can be supplied by the fill and drain system (pump) to the pool cooling water system, so to the pool;
- PWN supply: from the public water supply system (PWN) water can be drained to the pools; supply will be performed via the hydrofoor system.

Dedicated procedures to perform these actions in this situation are not available. However there exist procedures to drain and fill the pools and to operate the convection valves. In addition to the actions listed before, the following alternatives can be considered in case these preceding actions are not possible to perform or expire e.g. due to lack or exhaustion of supplies.

However these alternatives are preliminary, therefore there exist no additional means and procedures to implement these alternatives, as there are:

- Supply by the fire brigade; In line with those actions listed before water can be replenished to the pool cooling water system by the fire brigade by connections to this system via Stortz valves in cell-8 (PPG). Water can be supplied by the fire fighting system;
- Discharge to DWT; To avoid evaporation of the pool water there is the possibility to discharge pool water that is heated up to the Decontamination and Waste Treatment Facility. In this situation sufficient replacement from the public water supply system via hydrofoor (and pool fill system) shall be assured;
- Restoration of secondary cooling; Another possibility is to restore the secondary cooling by deployment of the fire brigade and application of the ducts of the secondary cooling water system. This may be successful because all remaining cooling systems are still in operation. This action includes supply of external water to the secondary cooling system at the first de-aeration point. Suction of water can be performed from either the North Sea or the HFR external pond (in case only the inlet of the secondary cooling water system is not available alternative suction from the Noordhollands kanaal might be possible too);
- Alternative secondary cooling[31]; Alternative secondary cooling means alternative cooling of the pool cooling water system (BKWS) by supply of hydrofoor water to the BKWS heat exchanger and discharge to the outlet of the secondary cooling water system. This will be established by fire hoses (and –connections), while control of radioactive material in the discharge line is included. Water for this alternative cooling will be supplied by the public water supply system;

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

From Table 5-2 it can be concluded that, given the sequence constraints, time before replenishment of pool water shall start within 464 hours. Then supply of a maximum of 105 m^3 can prolong the cooling period for another 330 hours in case this water is evaporated, finally followed by an infinite supply of the public water supply system. Application of the latter requires local manual operation of system valves inside the containment when large amounts of water already are evaporated. This introduces the risk of

[31] It is understood that this way of pool cooling will be applied during the long outage of 2012.

inaccessibility of the containment. Remote control of supply of the public water supply system should be fostered.

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

It can be concluded that the plant state LUHS is controlled by the systems that are available on-site supported by the external PWN.

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-edges

Cliff-edges are characterized by the interruption of cooling of the reactor. For the LUHS event for the HFR the following potential cliff-edges can be identified:

- Failure to dissipate decay heat to the whole content of the primary system;
- Failure to open the convection valves;
- Failure to replenish pool water;
- Failure to perform the indicated actions.

Potential action to prevent cliff-edge effects

Dealing with these potential cliff-edges is envisaged as follows:

<u>Failure to dissipate decay heat to the primary system</u>

The content of the primary system can be applied to dissipate decay heat from the core. In case this content is not available due to loss of primary flow, switch over to cooling by the pool water has to be performed within several hours. As a result, this is not a cliff-edge.

<u>Failure to open the convection valves</u>

Operation of the convection valves is local, by hand; including unblocking of these valves. Arguments that these actions should not be successful performed are:

- inaccessibility of the control device and/or;
- mechanical failure of the system.

Because opening of these valves is required before heating up and evaporation of pool water starts, inaccessibility is very unlikely as well as mechanical failure. Nevertheless this might then become a cliff-edge. The consequences of this cliff edge are described in section 5.3.

Failure to replenish pool water

After 464 hours water of the pools shall be replenished. A few ways to perform this are at hand. In case these ways fail or are not available, evaporation of the pool water will continue. This will basically cause an increase of the radiation level in the containment and in the control room. Habitability of both rooms will be endangered for the long run. This situation is elaborated in chapter 5.3.

Alternative actions like those based on deployment of the fire brigade or discharge of hot water to avoid evaporation, are not yet prepared and available for the time being. However it is anticipated that there is a lot of time to start these actions.

Failure to perform the indicated actions

Dedicated means and procedures to perform the indicated actions are not at hand and so trained in a way that smooth implementation is assured. Correction of this issue should be envisaged.

Measures to increase the robustness for LUHS

To deal with the potential cliff-edges in a more extensive way, the following measures can be (considered to) dealt with:

- Develop a set of procedures or enhance the existing set and implement a training program. Issues to be addressed in case of LUHS, are:
    - Heating up of the primary system;
    - Related switch over to core cooling by pool water;
    - Replenishment of pool water;
- Add or install means to enhance and ease the implementation of the indicated measures, e.g.:
    - To remotely control PWN water flow to the pool;
    - To remotely unblock and operate the convection valves;
- Elaborate the indicated alternative actions based on deployment of the fire brigade. Implement resulting measures, features and procedures, to enlarge possibilities for back up of the existing provisions, e.g.
    - Injection of (sea)water in the secondary cooling water system via the first de-aerating point;

- Injection of water to the pool cooling water system via the Stortz valves;

- Alternative secondary cooling of the pool cooling water system;
- Elaborate the indicated feed and bleed of water to the pools by PWN (feed) and DWT (bleed) to avoid evaporation (a first assessment, see Appendix A.9, section 9.2.5), showed that the PWN flow necessary to perform the action successfully, shall start at 6 m^3/h and will reduce during the course of the event to a fixed flow of approx. 3 m^3/h for the long run; currently the discharge capacity of DWT is 2 m^3/h and storage capacity 100 m^3, so this does not match at this moment).
- Elaborate the possibility of dissipation of the produced heat to the environment by circulation of the water of the primary and pool cooling systems and cooling (ventilation) of the rooms that contain these (non-isolated) systems. It is indicated that this way of "cooling" might be sufficient to prevent boiling of primary water.

5.1.3 Loss of the primary ultimate heat sink, combined with station black out

In line with the definition of station black out, for the HFR, see 5.1.1, the combination of loss of the primary ultimate heat sink and station black out can also be distinguished in two parts, viz.:

<u>LUHS-SBO1</u>

This means:

- The loss of the ultimate heat sink, viz. the Noordhollands kanaal for supply and the North Sea for discharge, combined with
- The unavailability of the emergency diesel generators viz. loss of the ordinary AC power back-up

<u>LUHS-SBO2</u>

This means:

- The loss of the ultimate heat sink, viz. the Noordhollands kanaal for supply and the North Sea for discharge, combined with
- The unavailability of the emergency diesel generators viz. loss of the ordinary AC power back-up and
- The unavailability of the AC power back-up by batteries that feed the pumps of the residual and decay heat removal system

Compared to the LUHS situation this means that:

- For the SBO1 situation, heat up of the primary system is limited to approx. 2 hours, which reflects the capacity of the batteries feeding both pumps that shall be operated in sequence; for the SBO2 situation forced primary water circulation is not available for this purpose anymore;
- Cooling of the reactor can still be performed with pool water by local manual opening of the convection valves;
- Additional supply to the pool from on-site available stocks is not possible due to lack of electrical power to the fill and drain pumps of the primary and pool cooling water systems;
- Supply to the pools by the public water supply system cannot be performed because transfer by the hydrofoor system (no E-power) is not possible;
- Cooling of the molybdenum facilities like Prometeo, Incomodo, Tycomodo and Mykonos will stop. Of all these facilities only the integrity of the Mykonos facility is endangered as the present fuel will become damaged and release of radioactive materials will occur. It is indicated that all the remaining facilities will suffer some temperature increase but will never exceed design limits.

5.1.3.1 Time of autonomy of the site before loss of normal reactor core cooling condition

For the LUHS-SBO events the complete mixing and complete isolation will be dealt with identical to the LUHS events

A. <u>The LUHS-SBO1 time constrains for complete mixing of the pools</u>

For the LUHS-SBO1 situation, initiating reactor shut down, the sequence to consider is:

1 Heating up of the primary system; Temperature rise of 100 m^3 of primary water during approx. 2 hours (battery capacities) starting at 50 ˚C; by decay heat of the core;

2 Insert pool water into the reactor; After those two hours decay heat from the core will be dissipated in the pool water (320 m^3) until this starts evaporating (heat up to 100 ˚C)

3 Heating up of storage pool water; Heat from spent fuel and other radioactive materials that are stored in the pools already heats pool water (320 m^3) during the same two hours of step 1, starting at 37 ˚C;

4 Evaporation of reactor pool water; Available is 160 m^3 due to the constraint of maximum decrease of the water level of the pools of 4 m. Step 3 will be performed in parallel with step 1, 2 and part of step 4. The moment that saturation of the water of the storage pool is reached, evaporation starts. From this moment on, heat dissipated in the storage pool contributes to the evaporation of the indicated 160 m^3 water.

5 Evaporation until separation of pools; Evaporation of approx. 25 m^3 of pool water will continue until top of the separation doors is reached; than complete isolation of the pools is assumed and separate heat dissipation to the pools by the related heat sources occurs

6 Evaporation until fuel uncover; Hereto, the lower level option is applicable resulting in core or spent fuel uncovery when approx. 35 m^3 or 28 m^3 of reactor pool water or of storage pool water respectively is evaporated. See chapter 5.3

7 Cooling of the facilities; During this sequence cooling of the facilities will not be available; heat to absorb is added to the decay heat of the reactor

A.9 section 9.1.2 presents the decay heat curve for the average core that is installed in the HFR. From this curve the heat produced by the core is derived. Absorption by the several water supplies is assessed. The results for this sequence are listed in Table 5-3 for the LUHS-SBO1, and more details can be found in Appendix A.9 section 9.2.2.

B. **<u>The LUHS-SBO1 time constrains for complete isolation of the pools for the heating up period</u>**

For LUHS-SBO1, initiating reactor shut down, the sequence to consider is:

1 Heating up the primary system; During two hours decay heat from the core will be dissipated in the primary system;

2 Insert pool water into the reactor; Heating up 130 m^3 of the reactor pool water from 37 to 100 ˚C, by decay heat of the core, at isolation conditions, starting after 2 hours of heating up the primary system (remaining heat absorption by the primary system is neglected)

3 Start of heating up of storage pool water; Temperature rise of 190 m^3 of storage pool water from 37 to 100 ˚C; by decay heat of the stored spent fuel Heat from spent fuel and other radioactive materials that are stored in the pools already heats up pool water in parallel with step 1 and 2.

4 Start of evaporation; When boil-off starts it is assumed that this situation will become identical to the situation of complete mixing; therefor, for the remainder, reference is made to the assessment of that situation

Appendix A.9 section 9.1.2 presents the decay heat curve for the average core that is installed in the HFR. From this curve the heat produced by the core is derived. Absorption by the several water supplies is assessed. The results for this sequence are listed in Table 5-3 for the LUHS-SBO1, and more details can be found in Appendix A.9 section 9.2.2.

Step	Sequence	Mixing situation			Isolation situation		
		Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)	Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)
1	2 hours of core heat to primary system	100	5.5	2	100	5.5	2
2	Heating up reactor pool water from 37 ˚C to 100 ˚C	320	84	60	130	34	31
3	Heating of remaining pools, starts in parallel with 1&2	320	84	60	190	50	137
4	Evaporation of 4 m of water from the pools	160	346	444			
5	Evaporation to separated pools	25	54	516			
6	Evaporation in separate pool, until first fuel uncovery	29	62	689			
A	*Extra time if water is injected*	*105*	*227*	*330*			

Table 5-3 Results of heat dissipation by alternate reactor cooling including two hours of primary system heat-up

From Table 5-3 it can be concluded that in case power is available during the first 2 hours of the events, so limited heat can be dissipated in the primary system:

- Boil-off will start after approx. 31 hours for the most conservative case and after 60 hours for the best estimate.
- The -4 m level will be reached after 444 hours. This is nearly one day shorter than the former situation (12 hours heat up of the primary system)

- Fuel uncovery occurs after 689 hours in the spent fuel pool; more than one day later followed by core uncover. This is due to the fact that at that time the heat output of both sources are almost equal, while the remaining amount of water in the reactor pool is larger.
- Compared to the former situation (12 hours heat up of the primary system) it can be indicated that the long run events occur about one day earlier in this situation.
- Again 330 hours of additional time is gained by supply of extra water that is already stored on-site.

For the ENSREG 72 hours mark it is noted that there will be a boil-off for about (72-31=) 41 hours in the most conservative estimate, and for (72-60=) 12 hours under the well-mixed assumption. The amount of water boiled off at 72 hours is 17 m^3 for the conservative estimate, resulting in a water level drop of about 0.4 m.

C. **<u>The LUHS-SBO2 time constrains for complete mixing of the pools</u>**

For the LUHS-SBO2 situation, initiating reactor shut down, the sequence is:

1 Heating up of pool water; Pool water will be inserted into the reactor from start of the sequence. Decay heat of the reactor plus heat from the spent fuel etc. will be dissipated in the pool water (320 m^3 from 37 to 100 ˚C) until this starts evaporating.

2 Evaporation of pool water; Available is 160 m^3 derived from the indicated operation constraint.

3 Uncovery of fuel elements; Again uncovery of fuel elements will proceed identical to the LUHS situation.

4 Cooling of the facilities; During this sequence cooling of the facilities will not be available; heat to absorb is added to the decay heat of the reactor.

Appendix A.9 section 9.1.2 presents the decay heat curve for the average core that is installed in the HFR. From this curve the heat produced by the core is derived. Absorption by the several water supplies is assessed. The results are listed in Table 5-4 for the LUHS-SBO2 situation; more details can be found in Appendix A.9 section 9.2.3.

D. **<u>The LUHS-SBO2 time constrains for complete isolation of the pools</u>**

1 Insert pool water; Opening of the convection valves to start circulation of pool water through the core. This results in heat up of 130 m^3 of the reactor pool water from 37 to 100 ˚C, by the decay heat of the core

2 Heating up of storage pool water; Temperature rise of 190 m^3 of storage pool water from 37 to 100 ˚C; by decay heat of the stored spent fuel.

3 Start of evaporation; When boil-off starts it is assumed that this situation will become identical to the situation of complete mixing; therefor, for the remainder, reference is made to the assessment of that situation.

From calculations in Appendix A.9 section 9.2.3, the results are listed in Table 5-4.

Step	Sequence	Mixing situation			Isolation situation		
		Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)	Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)
1	Heating up reactor pool water from 37 ˚C to 100 ˚C	320	84	56	130	34	24
2	Heating of remaining pools, starts in parallel with 1	320	84	56	190	50	137
3	Evaporation of 4 m of water from the pools	160	346	437			
4	Evaporation to separated pools	25	54	508			
5	Evaporation in separate pool, until first fuel uncovery	29	62	681			
A	*Extra time if water is injected*	*105*	*227*	*330*			

Table 5-4 Results of heat dissipation by alternate reactor cooling by only pool water

From Table 5-4 it can be concluded that in case no power is available, so no heat is dissipated in the primary system:

- Boil-off will start after approx. 24 hours for the most conservative case and after 56 hours for the well-mixed estimate.

- The -4 m level will be reached after 437 hours for both situations; 2 hours of extra cooling by the primary system (sequence B) provides eventually a delay of events by about 47 hours.
- Fuel uncovery occurs after 681 hours in the spent fuel pool; approx. one day later followed by core uncovery. This is due to the fact that at that time the heat output of both sources are almost equal, while the remaining amount of water in the reactor pool is larger. Compared to the situation that electrical power is available, it can be indicated that for this long run this rough estimate shows a difference of over one day.
- Again 330 hours additional time is gained by supply of extra water that already is stored on-site.

However cooling of the facilities will stop. This will result in an increase of the temperature of the facilities. For all facilities except for MYKONOS the temperature rise will not exceed design limits, so no damage will occur. For the MYKONOS production facility it is concluded that this increase of temperature will result in target damage and releases to the containment via the pool.

For the ENSREG 72 hours mark it is noted that there will be a boil-off for two days (72-24 hours) in the most conservative estimate, and for 16 hours (72-56 hours) under the well-mixed assumption. The amount of water boiled off at 72 hours is 20 m^3 in the conservative estimate, resulting in a water level drop of about 0.5 m.

5.1.3.2 External actions foreseen to prevent fuel degradation

For this assessment basically it is assumed that sufficient pool water is available. In case the pool water content is less or exceeds the indicated limit (-4 m of top of water level), replenishment of water shall be performed. From the possibilities to replenish pool water by stored water and/or by the PWN no one remains because all are (electrical) pump driven and pumps fail due to lack of electrical power.

From the alternatives indicated in 5.1.2.3.1 only remains:

- Supply by the fire brigade; Water can be replenished to the pool cooling water system by the fire brigade by connections to this system via Stortz valves in cell-8 (PPG). Water can be supplied by the fire fighting system.

The other alternatives suffer lack of electrical power supply; therefore these alternatives are not an option.

In addition to this it shall be elaborated to which extent gravity driven supply of PWN water can be applied to replenish the pools or to re-establish the water level of the pools to the level of the water stocks of PWN (“niveau reinwaterkelders”), meeting all constraints for direct refill of the pools by PWN.

For this situation it also has to be taken into account that PWN situation can be two-fold

1. SBO is not applicable for this system as it is external, having its own emergency power provision
2. SBO applies also for this system, this means that only 85 m^3 is available as this is the amount that is assured for cooling down of the reactor

It is emphasized that neither features or means nor procedures are at hand to perform such additional actions successfully.

5.1.3.3 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-edges

Cliff-edges are characterized by the interruption of cooling of the reactor. For LUHS event for the HFR the following potential cliff-edges can be identified:

- Exhaustion of batteries before switch over to cooling by pool water is allowed;
- Failure to open the convection valves;
- Failure to replenish pool water by PWN;
- Failure to perform the indicated actions.

Potential action to prevent cliff-edge effects

Dealing with these potential cliff-edges is envisaged as follows:

<u>Exhaustion of batteries before switch over to cooling by pool water is allowed</u>

The capacities of the dedicated batteries are sufficient to run both indicated pumps each for at least 30 min in sequence. This is in line with procedure H-04 that states that opening of the convection valves shall be performed after 5 min. As a result, this is not a cliff-edge.

<u>Failure to open the convection valves</u>

Operation of the convection valves is local, by hand; including unblocking of these valves. Arguments that these actions should not be successful performed are:

- inaccessibility of the control device and/or;
- mechanical failure of the system.

Although very unlikely, especially because actions are required before evaporation of pool water starts, this might then become a cliff-edge.

Failure to replenish pool water

After 437 hours water of the pools shall be replenished to meet the basic sequence constraints. Only supply by the fire brigade is available. In case this option fails or is not available, evaporation of the pool water will continue. This will basically cause an increase of the radiation level in the containment and in the control room. Habitability of both rooms will be endangered for the long run. This event is elaborated in chapter 5.3.

It is emphasized that this supply by the fire brigade is not prepared and available for the time being. However it is anticipated that there is lot of time to start those actions.

Failure to perform the indicated actions

Means and procedures to perform the indicated actions are not at hand and so trained in a way that smooth implementation is assured. Correction of this issue is envisaged.

Measures to increase the robustness for LUHS-SBO

To deal with the potential cliff-edges in a more extensive way the following measures can be (considered to) dealt with:

- Develop a set of procedures or enhance the existing set and implement a training program. Issues to be addressed in case of LUHS-SBO are:
    - Supply of pool water for core cooling;
    - Replenishment of pool water by the fire brigade;
- Add or install means to enhance and ease the implementation of the indicated measures, e.g.:
    - To remotely unblock and operate the convection valves;
- Elaborate the indicated alternative actions based on deployment of the fire brigade.
    - Implement resulting measures, features and procedures, to enlarge possibilities for back up of the existing provisions, e.g.;
    - Injection of water to the pool cooling water system via the Stortz valves;
- Elaborate to which extent gravity driven supply of PWN water can be applied to replenish the pools or to re-establish the water level of the pools to the level of the water stocks of PWN ("niveau reinwaterkelders"), meeting all constraints for direct refill of the pools by PWN.

5.2 Spent fuel storage pools

General

In the HFR radioactive heat producing materials are stored in the pools 1 and 2. According to the Operating Limits and Conditions in the HFR may be present, at a maximum:

- 756 fuel (spent) elements; including 33 “fresh” fuel elements of the core (stored 3 hours after shut down);
- For production of isotopes:
    - 25 kg U235 in total 100 kg uranium;
    - 5 kg remaining fissile materials;
- For transport of uranium filters of the production of isotopes
    - 25 kg enriched uranium.

For this assessment it is assumed that the pool is full loaded with spent fuel that is stored:

- “fresh” core: 33 fuel elements that passed 6 cycles in the core and 3 hours decay time at start of the event;
- Batch wise, 5 fuel elements that passed 6 cycles in the core;
- Every month, during 11 months per year, one batch.

This means that for heat production decay heat of 5 fuel elements is taken into account, plus decay heat from the next 5 fuel elements that already stayed one month in the pool etc.

This results in a total decay heat production of the spent fuel starting at approx. 80 kW decreasing to 60 kW within one month.

In addition to this, heat produced by the remaining radioactive materials that are stored in the pool is assumed to be constant and should be added while it declines according to the same decay heat curve as is applied for the spent fuel.

The HFR Dokpak from 1992 uses 100 kW for the heat production from stored spent fuel. The current amount of stored spent fuel is smaller than it was in 1992. Therefore, for this assessment, heat production is assumed to be 100 kW and it is conservatively assumed that for the duration of the event this heat production will be constant. The total heat production then is this 100 kW plus the decay heat of the core relaxed by 3 hours.

Where relevant, equivalent information is provided for the spent fuel storage pools as explained in Section 5.1 for the HFR.

Cooling systems

For heat removal from the pool the following two systems are included in the design:

- *The pool cooling water system;* This system provides cooling of the pools viz. reactor pool, pool 1 and pool 2 and transfers the heat released by the reactor vessel by convection, the heat produced by stored spent fuel and other radioactive materials as well as heat produced by facilities, to the secondary cooling water system
- *The secondary cooling water system;* This system transfers heat from primary and pool cooling water system to the environment viz the North Sea

Electrical power supply

Electrical power is supplied by the normal power system to both cooling systems. Only the pool water cooling system is fed by the emergency power system.

5.2.1 Loss of electrical power

Electrical power supply by regular systems and emergency power systems is presented in 5.1.

With regard to heat removal from the pools, loss of electrical power, for all situations listed in 5.1 (viz. LOOP, SBO) means that heat removal by the secondary system is not available. This is identical to the LUHS situation, in which heat produced by the radioactive materials that are stored in the pools, is dissipated in the pools. This issue is dealt with in 5.2.2.

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

Measures envisaged to increase the robustness of the plant in case of loss of electrical power are listed in 5.1.

In addition to these measures that deal with generic improvement of the electrical power supply, assurance of heat removal by the secondary system could be improved by connecting (part of) the secondary cooling water system to the emergency power system. This prevents LOOP to turn immediately into and be identical to LUHS with regard to cooling.

5.2.2 Loss of the ultimate heat sink

Loss of the ultimate heat sink for the HFR means that there will be no supply of water from the Noordhollands kanaal or there will be no discharge to the North Sea.

Alternatives for these inlet and outlet are not included in the design

5.2.2.1 Availability of an alternate heat sink

In terms of the ENSREG definition, the HFR is not equipped with an alternate ultimate heat sink.

Instead, heat produced by the radioactive materials stored in the pools, will be dissipated in these pools. This means that pools will heat up and finally pool water will start evaporating.

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

For the assessment of sole spent fuel cooling, the situations of complete mixing and complete isolation of the water of the pools only differ in the sequential use of water during the heating up phase. For the complete mixing situation the available water volume of the pools is taken at one time. For the complete isolation situation this phase is split into first heating up the water of the storage pool followed by heating up of the water of the other pools. Therefore, for the assessment, this heating up is indicated first, followed by the heating up of the remaining amount of water. Summing the results reflects the merging of these steps to the heating up of the entire, mixing, pool water. For the LUHS situation the sequence to consider therefore includes:

1 Heating up of storage pool water; Heating up 106 m^3 of the water of pool 1 37 to 100 ˚C. It is assumed that during heating up there will be no mixing of water of the reactor pool and the pools 1 and 2.

2 Heating up of remaining pool water; After saturation of the storage pool water the 214 m^3 of remaining pool water rises from 37 to 100 ˚C, basically without mixing of water. Note: summing up the results of step 1 and 2 equals the situation of mixing and heating up 320 m^3 of all pools.

3 Evaporation of pool water; Available is 160 m^3. This is derived from the operation constraint that the water level of the pools may not decrease more than 4 m. This 4 m refers to the minimum water layer needed for protection against radiation from stored fuel and targets in the pools and the head of the reactor vessel. It is noted that lower levels of the pool water are acceptable; ultimately when no replenishment can be performed approx. 214 m^3 of water will be evaporated before uncovery of spent fuel elements will occur. For this situation the amount of this 214 m^3

stems from 185 m^3 common volume of all pools plus 29 m^3 remaining spent fuel pool water when pools are separated due to low water level. This is elaborated in chapter 5.3.

Appendix A.9 section 9.2.4 includes the estimate of the heat dissipated in the pools by the stored radioactive materials. This is based on the decay heat production curve, presented in Appendix A.9 section 9.1.2 and the (heat) load derived from the maximum loading of spent fuel elements and other radioactive materials, as this is allowed by the OLCs, in the sequence defined above. The results are listed in Table 5-5.

From Table 5-5 it can be concluded that while only heat is produced in the storage pool:

- The spent fuel pool itself would be saturated after 13 hours, if no mixing with the other pools occurs. Saturation of the entire pool would occur after 59 hours.
- The 4 meter level occurs after 445 hours, and separation into separate pools after 517 hours. These are similar times as for the scenarios with the core still in the reactor.
- Since all heat production takes place in the spent fuel pool, the water level after separation will fall faster in this pool than in scenarios where the core is still in the reactor. After 603 hours, the top of the spent fuel will be reached. This is a difference of ample 3 days less than the sequence when complete station black out is assumed and shut down of the reactor the moment the event starts.

For the ENSREG 72 hours mark it is noted that for the most conservative case there will be a boil-off for 60 hours in the most conservative estimate, and for 13 hours under the well-mixed assumption. The amount of water boiled off at 72 hours is 33 m^3 in the conservative estimate, resulting in a water level drop of about 0.8 m.

Step	Sequence	Mixing situation[32]			Isolated situation		
		Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)	Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)
1	T rise of fuel pool from 37 to 100 ˚C	320	84	59	106	27.7	13
2	T rise of remaining pools from 37 to 100 ˚C	320	84	59	214	55.8	59
3	Evaporation of 4 m of water from the pools	160	346	445			
4	Evaporation to separated pools	25	54	517			
5	Evaporation in separate pool, until first fuel uncovery	29	62	603			
A	*Extra time if water is injected*	*105*	*227*	*330 extra*			

Table 5-5: Results of heat dissipation by alternate spent fuel cooling for LUHS

From Table 5-5 it can be concluded that while only heat is produced in the storage pool:

- The spent fuel pool itself would be saturated after 13 hours, if no mixing with the other pools occurs. Saturation of the entire pool would occur after 59 hours;
- The 4 meter level occurs after 441 hours, and separation into separate pools after 500 hours. These are similar times as for the scenarios with the core still in the reactor;
- Since all heat production takes place in the spent fuel pool, the water level after separation will fall faster in this pool than in scenarios where the core is still in the reactor. After 608 hours, the top of the spent fuel will be reached. This is a difference of approx. 4 days less than the sequence when complete station black out is assumed and shut down of the reactor the moment the event starts.

For the ENSREG 72 hours mark it is noted that for the most conservative case there will be a boil-off for 60 hours in the most conservative estimate, and for 13 hours under the well-mixed assumption. The

[32] For the mixing situation step 1 and 2 are merged

amount of water boiled off at 72 hours is 33 m^3 in the conservative estimate, resulting in a water level drop of about 0.8 m.

5.2.2.3 External actions foreseen to prevent fuel degradation

From Table 5-5 it can be concluded that, given the constraints for the sequence, for a time period of 608 hours spent fuel cooling is assured. It has to be emphasized that this only refers to cooling of spent fuel in the pool that includes regular storage of a “fresh” unloaded core

Identical to the reactor cooling, pool water can be replenished in case it starts evaporating. The same stocks are available, viz.:

- Deminwater storage (45 m3);
- Cooling water storage (60 m3);
- PWN supply (infinite).

Hereto, it shall be assumed that in case the pool water content is less or exceeds the indicated limit (-4 m of top of water level), replenishment of water can be performed.

Dedicated procedures to perform these actions in this situation are not available. However there exist procedures to drain and fill the pools and to operate the convection valves.

Actions by the fire brigade can be considered to:

- replenish water to the pool cooling water system by connections to the pool cooling water system via Stortz valves . Water can be supplied by the fire fighting system
- restore secondary cooling by injection of (sea) water into this system
- apply alternative secondary cooling (see 5.1.2.3.1)

Also supply by PWN and drain of hot water to DWT to avoid evaporation might be possible. Again it is emphasized that neither features or means nor procedures are at hand to perform these additional actions successfully.

5.2.2.4 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 various examples of time delay from reactor shutdown to loss of normal reactor core cooling condition (e.g., start of water loss from the primary circuit)

From Table 5-5 it can be concluded that, given the constraints for this sequence, time before replenishment of pool water shall start is 445hours. Then supply of a maximum of 105 m^3 can prolong the cooling period for 330 hours in case this water is evaporated, finally followed by an infinite supply of the public water supply system. It is indicated that to ensure the latter supply under “steam conditions”, remotely control for the supply should be fostered.

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

It can be concluded that, for spent fuel pool cooling, the plant state LUHS is controlled by the systems that are available on-site supported by the external PWN.

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

Potential cliff-edges

Cliff-edges are characterized by the interruption of cooling of the pools. For LUHS event for the HFR the following potential cliff-edges can be identified:

- Failure to replenish pool water;
- Failure to perform the indicated supporting/follow up actions.

Potential action to prevent cliff-edge effects

Dealing with these potential cliff-edges is envisaged as follows:

<u>Failure to replenish pool water</u>

After 441 hours water of the pools shall be replenished. A few ways to perform this are at hand. In case these ways fail or are not available, evaporation of the pool water will continue. This will basically cause an increase of the radiation level in the containment and in the control room. Habitability of both rooms will be endangered for the long run. This event is dealt with in chapter 5.3. Alternative actions based on deployment of the fire brigade are not prepared and available for the time being. However it is anticipated that there is a lot of time to start these actions.

<u>Failure to perform the indicated actions</u>

Means and procedures to perform the indicated actions are not at hand and so trained in a way that smooth implementation is assured. Correction of this issue is envisaged.

Measures to increase the robustness for LUHS

To deal with the potential cliff-edges in a more extensive way the following measures can be (considered to) dealt with:

- Develop a set of procedures or enhance the existing set and implement a training program. Issues to be addressed in case of LUHS are:
    - Replenishment of pool water;
- Add or install means to enhance and ease the implementation of the indicated measures, e.g.:
    - To remotely control PWN water flow to the pool;
- Elaborate the indicated alternative actions based on deployment of the fire brigade. Implement resulting measures, features and procedures, to enlarge possibilities for back up of the existing provisions, e.g.
    - Injection of (sea)water in the secondary cooling water system via the first de-aerating point;
    - Injection of water to the pool cooling water system via the Stortz valves;
    - Alternative secondary cooling of the pool cooling water system;
- Elaborate the indicated feed and bleed of water to the pools by PWN (feed) and DWT (bleed) to avoid evaporation
- Elaborate the possibility of dissipation of the produced heat to the environment by circulation of the water of the cooling systems and cooling (ventilation) of the rooms that contain these (non-isolated) systems. It is indicated that this way of “cooling” might be sufficient to prevent boiling of primary water.

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)

In chapter 5.1.1 a distinction is made between LOOP, SBO1 and SBO2. With regard to heat removal of the spent fuel pool of the HFR this distinction is not applicable. Even more, LOOP and in its succession SBO1 and SBO2 already cause loss of ultimate heat sink. This makes the situations of spent fuel pool cooling for the combinations of events viz. LUHS-SBO1 and LUHS SBO2 identical to the LUHS situation, except for the possible actions to replenish pool water; the latter will be indicated. Heat removal, for all situations, is only performed by first heating up pool water followed by evaporation of this water.

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

Identical to the LUHS situation, see 5.2.2.2.

5.2.3.2 External actions foreseen to prevent fuel degradation

Again it can be concluded that for a time period of 603 hours spent fuel cooling is assured. It has to be emphasized that this only refers to spent fuel cooling; the combination of reactor core cooling and spent fuel cooling is dealt with separately in chapter 5.1.

Identical to the reactor cooling pool water can be replenished in case it starts evaporating. Hereto, it shall be assumed that in case the pool water content is less or exceeds the indicated limit (-4 m of top of water level), replenishment of water shall be performed.

Only the actions by the fire brigade to replenish water to the pool cooling water system by connections to the pool cooling water system via Stortz valves can be considered. Water can be supplied by the fire fighting system.

Further more elaborate gravity driven supply of PWN water to the pools.

Again it is emphasized that neither features or means nor procedures are at hand to perform these additional actions successfully.

5.2.3.3 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 various examples of time delay from reactor shutdown to loss of normal reactor core cooling condition (e.g., start of water loss from the primary circuit)

From Table 5-5 it can be concluded that time before replenishment of pool water shall start, is 441 hours. Then supply by the fire brigade shall start.

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

It can be concluded that, for spent fuel pool cooling, the plant state LUHS-SBO is controlled by the supply of water by the fire brigade.

5.2.3.5 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-edges, the ways to deal with these and measures that can be envisaged to increase the robustness of the plant for the LUHS-SBO situation, are identical to those that are listed for the LUHS situation.

5.3 Supplementary consequences

ENSREG indicates that an assessment should be made for the situation that cliff-edges will occur and will result in loss of all means to mitigate the plant conditions dealt with in this report, so fuel damage will result. The most severe events, in this context, are the events that

- no forced primary cooling water flow occurs and no pool water will be inserted into the reactor;
- no pool water will be replenished

No forced primary flow and no water insertion in the reactor

This is the situation that primary water flow stops and that opening of the convection valves fails, so no heat will be dissipated into the total content of the primary system and no pool water will be inserted into the reactor vessel. So cooling of the reactor by pool water is not available. However a small amount of primary cooling water will heat up and a small natural recirculation flow through the APE lines will develop. Then evaporation will start and the natural recirculation stops. Steam will be released through the expansion vessel. Approx. 10 % of the primary water content will be used during this heating up phase and approx. 5 % for evaporation before the core uncovery and core heat up starts.

The sequence to consider is:

1 Heating up of primary cooling water; To simplify the assessment it is assumed that first 10 m^3 of water will be heating up

2 Evaporation of primary cooling water; From this 10 m^3 of water approx. 5 m^3 will evaporate

3 Heating up of core; The core temperature will increase rapidly

Based on the heat production of the core as in Appendix A.9 section 9.1.2, the results are listed in Table 5-6, calculating heating up of 10 m^3 and evaporating 5 m^3 of the primary cooling system water.

Step	Sequence	Volume involved (m^3)	Duration (hrs)
1	Heating up of primary water	10	0,5
2	Evaporation of primary water	5	7

Table 5-6 results of heat dissipation in the primary system without primary flow

From the results listed in Table 5-6 it can be concluded that after approx. 7 hours core temperature starts rising, provided 10% of the primary system inventory is applied, of which half of it will evaporate.

This is in the same order of magnitude of Risk Scoping Study results, chapter 6.2, which are based on RELAP assessments, that show that for an almost similar sequence core damage will arrive after approx. 12 hours. Those RELAP calculations take also dissipation of heat from the reactor into the pool into account.

Because it is indicated that for the LUHS situation with regard to heat production and heat removal there are slightly differences compared to the RELAP sequence, for the event under consideration it is concluded that core heat-up will start approx. 7 to 12 hours after initiating of the event. There might be some preference for 12 hours as those RELAP calculations are supposed to be more reliable then this rough assessment.

No water replenishment

This is the situation that refilling of the pool is not possible and that pool water start evaporating and continues to evaporate until uncovery of the core or of the spent fuel occurs and temperature of the fuel elements will increase rapidly.

Before this situation will be reached, the habitability of the containment and the control room will by endangered by the increase of radiation as the water level of the pools will decrease.

Before defining the sequence to be considered the following, illustrated in 0, has to be taken into account:

- Today exchange of irradiation targets is performed on floating platforms in the pools, covered by a fixed water layer to provide radiation shielding. This means that when water pool level decreases, the platforms will move downwards and that a fixed water layer will remain above the platforms to assure shielding. In this way lowering the water level by approx. 4 m is acceptable, because the platforms still will be shielded sufficient.

- Uncovery of the core or of spent fuel is indicated to occur when the water level is 2.17 m below the top of the pool doors. This means that the water level then is decreased approx. 6.79 m.
- Mixed cooling, this means mixing of pool water during the event, only will be possible in case the water level is above the top of the pool doors. For the assessment the -4.62 m level is marked to change from mixed cooling to isolated cooling. The latter means that decay heat from the reactor will be dissipated in the remaining reactor pool water, the heat from the spent fuel in the water of pool 1(storage pool).

The following sequences have to be considered:

A. Core and spent fuel cooling:

This refers to the situations presented in chapter 5.1, including reactor shut down at start of the event (core inside reactor vessel) and maximum loaded storage pool (spent fuel and other radioactive materials limited by OLC conditions). Because the situations of (heat) solation and mixing of pools are identical for the long run, these situations are dealt with at one time. The event of total station black out is considered to be the enveloping one.

B. Sole spent fuel cooling:

This refers to the situations presented in chapter 5.2 including an unloaded core (3 hours delay time) that is stored in the maximum loaded storage pool. Because for this configuration LUHS is identical to LUHS-SBO, total station black out is also considered to be the enveloping event.

Based on this, the following sequences have to be considered to assess the time frame of severity of the event.

1 Heating up plus evaporation of pool water; Decay heat from the reactor and heat from the spent fuel etc. or from the sole spent fuel will be dissipated in the pool water (320 m^3) until this starts evaporating. For this step it is assumed that the water level will decrease 4 m, referring to today's situation of floating platforms.. This equals evaporation of approx. 160 m^3 of pool water.

2 Continued evaporation; Lowering the water level down to the separating doors, additional 25 m^3 for evaporation is used

3 Separate evaporation

For A: core and spent fuel cooling, for core uncovery, approx. 35 m^3 reactor pool water has to be evaporated until top of fuel is reached and temperature of the core starts rising. For reaching top of spent fuel element 28 m^3 of pool water has to be evaporated

For B: sole spent fuel cooling, only the water of the storage pool, 28 m^3, has to be evaporated.

The results for these situations are copied from Table 5-4 and Table 5-5 and merged in Table 5-7.

From the results listed in Table 5-7 it can be concluded that in case there will be no replenishment of pool water at all the following situations will occur:

1. After approx. 440 hours the water level of the pools will be decreased 4 m; this is the operation limit of today for the accessibility of the containment (radiation protection).
2. After 500 hours the top of the doors will be reached, this means that from this moment evaporation of the pools is driven by different heat sources viz. the decay heat of the reactor dissipating its remaining heat in the reactor pool and the spent fuel etc. dissipating their heat into pool 1.
3. Uncovery of spent fuel elements will occur after 608 hours in case of unloaded core and 2 to 3 days later when the reactor is shut down at start of the event (full core in reactor vessel).
4. In case at start of the event the reactor is in operation and is shut down (loaded core), while the storage pools are loaded up to their maximum, uncovery of the spent fuel elements will last another 3 days compared to the unloaded core situation. Uncovery of the core then will last additional more than one day longer.

Step	Sequence	Mixing situation						Isolation situation					
		Core and spent fuel cooling			Sole spent fuel cooling			Core and spent fuel cooling			Sole spent fuel cooling		
		Volume involved (m^3)	Energy involved (GJ)	End time (hours since shutdown)	Volume involved (m^3)	Energy involved (GJ)	End time (hours since shutdown)	Volume involved (m^3)	Energy involved (GJ)	End time (hours since shutdown)	Volume involved (m^3)	Energy involved (GJ)	End time (hours since shutdown)
1 a	Heating up reactor pool water from 37 ˚C to 100 ˚C	320	84	56	320	84	59	130	34	24	106	27.7	13
1 b	Heating of remaining pools, starts in parallel with 1a	320	84	56	320	84	59	190	50	137	214	55.8	59
2	Evaporation of 4 m of water from the pools	160	346	437	160	346	445						
3	Evaporation to separated pools	25	54	508	25	54	517						
4	Evaporation in separate pool, until first fuel uncovery	29	62	681	29	62	603						

Table 5-7: Results of heat dissipation in the pools without any replenishment

5.4 Other NRG facilities at the OLP

Where relevant, equivalent information is provided for the other nuclear facilities at OLP premises.

In this section the other nuclear facilities at the OLP will be described in relation to LOOP-SBO. These facilities are less complicated as the HFR; only the confinement functionality has to be investigated. Cooling does not apply to these facilities, while criticality is dealt with in Section 6.3.4 of this report.

5.4.1 The nuclear facilities

5.4.1.1 Hot Cell Laboratories

The hot cell laboratories (HCL) consist of two buildings: the Research Laboratory (RL) and the Molybdenum Production Facility (MPF). These two buildings are connected by two corridors: one for employees for daily use and one is serving as a possible escape route during accidents. An airlock at the corridors separates the ventilation (under-pressure) systems of the RL and MPF building parts. Therefore and because of the separate electrical power systems the MPF will be described in section 5.5.1.2.

The normal electrical power for the HCL-RL is delivered from substation ECN 6. The main distribution board is placed on the ground floor together with the distribution boards K3 (NS) and K3E (NS) of the emergency power supply. The cables connected to these distribution boards are run in cable trays that are situated in the first basement. On the second floor a main distribution sub-board is placed together with distribution sub-board K3E1 (NS) for the emergency power supply. These two distribution boxes are not separated from each other and placed in the same room. Further a number of emergency power sub-boards are located in separate rooms throughout the ground- and first floor.

5.4.1.1.1 Loss of electrical power

In case of LOOP the systems of the HCL-RL that are equipped with emergency power are connected to emergency power bus bar A. These facilities remain in function when emergency power generators A and C fail and emergency power generator B feeds bus bar A after load-shedding. The emergency power supply connection is tested once every six months, except when the emergency power supply has been used to deliver power to the various HCL systems for a period during these six months and was functioning well.

The HCL-RL is equipped with a water pool of 58 m^3 that is used for storage of processed U-targets with negligible heat production. The pool water is circulated by pumps through a purification system that has no cooling function whatsoever. Loss of electrical power to this system will have no radiological consequences.

The following main components of the HCL-RL are supplied with electrical energy by emergency bus bar A:

- Manipulators;
- Emergency lighting on the ground- and first floors;
- Pumping equipment and the ventilators of the ventilation system;
- Emergency compressor;
- Radioactive release monitoring equipment.

Because the ventilation system, which is necessary to maintain confinement, remains in operation during LOOP, the confinement function is maintained. So, LOOP will not result in releases of radioactive materials.

In case of LOOP-SBO1 the consequences of failure of power supply for the HCL-RL are failure of the intake- and exhaust ventilators resulting in the elimination of the under-pressure regime. The cells are isolated, but a leakage can develop at the openings of the manipulators. In this situation it could be possible that radioactive substances are leaking to other places in the building. At relatively high wind speeds along the building, radioactive substances can be blown through the air outside the building, but the radiological consequences would still be relatively small. A proposition of the exact radiological consequences is not available, quantification of possible releases are still to be provided.

In case of LOOP-SBO2 the batteries of the fire detection system have a stand-by time of 72 hours. The alarm- and radiation monitoring system and the buildings' intercom system are equipped with a no-break power supply to deliver uninterrupted power during the initial period of the emergency power system. The no-break batteries have a discharge time of approx. 30 minutes. From that time on the eventual radioactive releases will not be recorded anymore. The battery stand-by time of 1 hour of both the centralized and the decentralized emergency lighting system is sufficient for evacuation of the building in an orderly fashion. With regard to possible releases, the SBO2 situation is identical to the SBO1 situation, so quantification is to be provided.

5.4.1.2 Molybdenum Production Facility

The Molybdenum Production Facility (MPF) is part of the Hot Cell Laboratory (HCL), but has a separate, independent power supply. In the MPF molybdenum is separated from uranium originating from the HFR. This process takes place in concrete cells using manipulators placed outside the concrete cells.

The normal electrical power for the MPF is delivered from substation ECN 3. The main distribution board is placed on the first floor together with the main distribution board K7A (NS) of the emergency power supply. Further a number of emergency power sub-boards are located in separate rooms throughout the MPF building.

5.4.1.2.1 Loss of electrical power

In case of LOOP the emergency power facilities of the MPF are fed from bus bar B during a loss of electrical power. These are however disconnected at the moment generator B feeds bus bar A in case the generators A and C should fail and load-shedding occurs.

The emergency power supply is tested once every six months, except when the emergency power supply has been used to deliver power to the various systems for a period during these six months and it was functioning well.

The following main components of the MPF are supplied with electrical energy by emergency bus bar B:

- Pumping equipment and intake- and exhaust ventilators;
- Emergency lighting;
- Control panels of cells and pumps of cells 01/11 and 02/12;
- Warning- and monitoring systems and the central intercom;
- Fire alarm system and CO2 fire fighting system;
- MPF emergency compressor

The Control panels of cells and pumps of cells 01/11 and 02/12, the warning- and monitoring systems and the central intercom and the fire alarm system and CO_2 fire fighting system are equipped with a no-break power supply to deliver an uninterrupted power supply during the initial period of the emergency power system. The discharge time of the batteries of the fire fighting department is 72 hours. The other no-break batteries have a discharge time of around 30 minutes. The batteries of the decentralized emergency lighting system have a discharge time of at least 1 hour.

Because the ventilation system, which is necessary to maintain confinement, remains in operation during LOOP, the confinement function is maintained. So, LOOP will not result in releases of radioactive materials.

In case of LOOP-SBO1 the consequences of failure of power supply for the MPF are a failure of the intake- and exhaust ventilators resulting in the elimination of the under-pressure regime. The cells 01/11 are isolated, but a leakage can develop at the openings of the manipulators although this leakage is counteracted by the elimination of the under-pressure regime in other parts of the building. In this situation it is possible that radioactive substances are leaking to other places in the building. At relatively high wind speeds along the building radioactive substances can be blown through the air outside the building, but the radiological consequences would still be still relatively small. A proposition of the exact radiological consequences is not available, quantification of possible releases are still to be provided.

In the production cells H_2 is released during the molybdenum separation process. This H_2 is led to a copper oxide furnace where copper oxide is reduced to copper and water is produced in an exothermic reaction ($CuO + H_2$ -- 380˚C--→ $Cu + H_2O$ +129 kJ). The heat produced in this process is negligible and does not require forced cooling. The reduction of the copper oxide takes place in a completely closed system. So, failure of the ventilation system does not cause an H2 problem here. The driving forces of the reduction process are not dependent of an emergency power source.

In case of LOOP-SBO2 the batteries of the fire detection system have a stand-by time of 72 hours. The batteries of the alarm- and the radiation monitoring system and the buildings intercom system have a stand-by time of ca. 30 minutes. From that time on the eventual radioactive releases will not be recorded anymore. The battery stand-by time of 1 hour of the emergency lighting system is sufficient for evacuation of the building in an orderly fashion.

The battery stand-by time of 1 hour of the emergency lighting system is under most circumstances sufficient for evacuation of the building in an orderly fashion. The molybdenum-dissolution step might take longer than this 1-hour period, if it happens to be started shortly before the occurrence of SBO.

With regard to possible releases, the SBO2 situation is identical to the SBO1 situation, so quantification is to be provided.

5.4.1.3 Waste Storage Facility

Inside the Waste Storage Facility (WSF) radioactive waste and nuclear material is stored in areas under the ground floor. The ventilation system creates an under-pressure in the areas under the ground floor where the nuclear fuel and the radioactive waste is stored and the filtering system filters out the radioactive particles in the air. The amount of radioactive particles is monitored by the monitoring system. The normal electrical power for the WSF is delivered from substation ECN 4.

5.4.1.3.1 Loss of electrical power

In case of LOOP the ventilation system, the filtering system and the radiation monitors of the WSF are connected with emergency bus bar A. In case of LOOP-SBO1 when having loss of electrical power, the ventilation system stops and the under-pressure created by the system no longer exists. Also the filtering system stops filtering. In case of LOOP-SBO the WSF has an UPS for the radiation monitoring system which feeds system for a period of one hour. The radioactive material is enclosed in barrels. Therefore without the above systems properly working still no critical situation will develop. Only at the moment the barrels are damaged, radioactive particles can be set free without being monitored and filtered by the monitoring system and the filtering system. At this moment, some barrels are already damaged..

5.4.1.4 Decontamination and Waste Treatment Facility

In the Decontamination and Waste Treatment Facility (DWT) the processing and storage of radioactive waste and materials takes place in rooms under the ground floor, which are liquid tight, The normal electrical power for DWT is delivered from substation ECN 4.

5.4.1.4.1 Loss of electrical power

In case of LOOP the ventilation system, the filtering system, the monitoring system and the emergency lighting in the DWT is connected with the emergency bus bar A. In case of LOOP-SBO1 as the electrical power is lost, no radioactivity will escape; the barrels with the nuclear materials are leak tight and stored under concrete. Only at the moment the barrels are damaged, radioactive particles can be set free without being monitored and filtered by the monitoring system and the filtering system. The LOOP-SBO2 situation does not apply to the DWT since there are no battery back-up systems to speak of.

In a solitary LOOP-SBO situation there will be no damage to those barrels, therefore no releases will occur.

5.4.1.5 Jaap Goedkoop Laboratory

In the Jaap Goedkoop Laboratory (JGL) research is performed to materials used by nuclear fusion, research to the shortening of the lifetime of nuclear waste, production of the medical isotope ^{177}Lu and the development of nuclear medicines. The normal electrical power for the JGL is delivered from substation ECN 7.

5.4.1.5.1 Loss of electrical power

In case of LOOP the following systems are supplied with electrical energy by emergency bus bar B:

- Air conditioning systems for building and off-gas systems for glove-boxes;
- Wall sockets in laboratories for active experiments;
- Warning and monitoring systems and central intercom;

- Fire warning system;
- Emergency lighting system;

In case of LOOP-SBO1 failure of electrical power supply does not imply the spreading of radioactivity. Only at the moment that ^{177}Lu production takes place in the glove-boxes and LOOP-SBO1 occurs, the under-pressure disappears however confinement, by isolation by design, of the box is still maintained so no releases occur. Only in case the isolation of the glove-boxes fails, it is possible that ^{177}Lu will be released in the building and, with relatively high winds, can be set free in the environment outside the building. It is indicated that the combination of LOOP-SBO and failure of glove-box isolation is very unlikely.

The failure of electronics of active experiments and CO gas monitoring is not a problem, since the radioactivity stays inside the containment. In case of LOOP-SBO2 the batteries of the fire detection system have a stand-by time of 72 hours. The battery stand-by time of 1 hour of the emergency lighting system is sufficient for evacuation of the building in an orderly fashion.

In conclusion, in solitary LOOP-SBO situations, there will be no releases of radioactive materials

5.4.1.6 NRG-RE

The NRG-RE building is used for research and advice about handling radioactive waste and radiation. In order to do this the building is equipped with laboratories for analyses. NRG-RE also calibrates measurement systems and provides dosimetry. The normal electrical power for NRG-RE is delivered from substation ECN 3.

5.4.1.6.1 Loss of electrical power

In case of LOOP the following systems of NRG-RE are supplied with electrical energy by emergency bus bar B:

- Air conditioning systems
- No-break batteries for information systems
- LAN data
- CRM engineering workstation

In case of LOOP-SBO the no-break sets of the above information systems have a limited stand-by time of ca. 15 minutes which only causes a possible loss of data. In the fume-hoods the under-pressure will be lost but no radioactivity will escape at the moment the emergency power supply fails to deliver electrical energy It is common practice to stop processes and to isolate these hoods in case of emergencies. Formal procedures are not at hand. Development of such procedures is proposed as measure.

5.4.1.7 Low Flux Reactor

The Low Flux Reactor (LFR) is out of service. The normal electrical energy for monitoring and ventilation is supplied by substation ECN 4. The LFR is not equipped with emergency power. In case of LOOP the monitoring system will become unavailable. This is however not a problem since the LFR is not operated anymore. Since this is the case the ventilation system is in the shutdown mode. This means that the ventilation shut-off valves are closed and as a result the system is isolated so no (eventual) radioactivity will escape. LOOP will not affect the system isolation.

5.4.1.8 STEK hall

The STEK facility itself is decommissioned, but the remaining hall is used for storage of radioactive waste that is enclosed in barrels. This is mainly LAVA, MAVA and HAVA pending for their shipment to COVRA. The electrical power for monitoring and ventilation is supplied by substation ECN 4. The STEK hall is not equipped with emergency power. In case of LOOP the monitoring system will become unavailable. For the situation under consideration this is not a problem since the waste is stored in closed barrels so no radioactivity will escape. However it should be elaborated whether uninterrupted monitoring should be performed in case of events that risk damage of the barrels.

In addition, the ventilation shut-off valves close at LOOP and the system will be isolated.

5.4.2 Remaining NRG facilities

Apart from the above nuclear facilities at the OLP some other non-nuclear NRG facilities can be distinguished that can be important in case of a severe incident. The eventual effects of LOOP-SBO are discussed below.

5.4.2.1 NRG Security Lodge

5.4.2.1.1 Loss of electrical power

The normal electrical power for the NRG main security lodge is delivered from substation ECN 3. In case of LOOP the following systems are connected to emergency bus bar B:

- drive motor of the main gate and signalling of the speed bump;
- access control system and camera main gate;

- no-break set with battery back-up for information systems

The no-break set of the information systems in the security lodge has a stand-by time ca.15 minutes to overcome voltage dips:

- In case of Loop-SBO the consequences of loss of electrical power for security lodge are:
- Impossible to open a main gate by remote control and signalling of approaching traffic is not functioning anymore. This does not pose a big problem, since only one of the two main gates is electrically driven and can be opened manually during an emergency situation;
- Non-availability of access control system and camera security. In that case visitor access can be manually controlled. Loss of camera information can be solved by extra surveillance;
- Non-availability LAN and local computers after no-break back up time is finished;
- The sign table and the monitors of the central reporting room malfunction when the UPS is exhausted. No incoming emergency calls can be answered anymore;
- The lighting of the central reporting room is not functioning anymore.

5.4.2.2 NRG Fire Department

The normal electrical power for the NRG fire house is delivered from substation ECN 3. The command centre of the fire brigade (CMP) is located in the HFR security lodge (BAC) that has an electrical power connection to emergency bus bar A. In case of a fire alarm at the OLP the fire brigade is alerted from the BAC.

Note; For security reasons the impact of LOOP-SBO on the BAC itself is not included in this report.

5.4.2.2.1 Loss of electrical power

In case of LOOP the following systems of the fire department building 2 are connected to emergency bus bar B:

- lighting central reporting room and paging system of the building
- LAN and local computers

In case of LOOP-SBO the consequences of loss of electrical power and as a result the loss of lighting can be compensated by the fire brigade emergency lighting with battery back-up. The loss of the paging system has as a back-up the digital C2000 communication network. The local computers contain the digital version of the fire brigade response plans. During LOOP-SBO a paper copy has to be used.

Apart from the provisions above the fire brigade is equipped with a small petrol fired mobile generator of some kW's. Additional an iron stock of diesel fuel is kept for the fire engine trucks.

5.4.2.3 ECN 3 Compressor

In the ECN 3 substation a part of the building is squared off as a compressor station that supplies tool- and instrument air for the OLP including the HFR. In case of LOOP the following systems are supplied with electrical energy by emergency bus bar B:

- Electrical driven compressor
- Control panel for a compressor driven by a gas motor
- Control panel and battery charger for a diesel driven compressor

This means that in case of LOOP the tool- and instrument air supply for the OLP is secured because of the emergency power to the above equipment. In case of LOOP-SBO1 the tool- and instrument air supply for the OLP is not secured because of lack of electrical power to both the control panels of the gas motor- and diesel motor driven compressors. The compressor system is equipped with a buffer capacity of two buried tanks with a 10 m^3 volume that are kept at a pressure of 6.5 bar. The OLP tool air PE-piping network (∅ 100 mm) with a total length of approx. 2 km is a further buffer of approx. 15 m^3.

HFR:

Th spembley suits and fresh air masks in the HFR use the tool- and instrument air supply. These will therefore not be usable in case of LOOP-SBO. It is not likely that during LOOP-SBO1 work will be executed requiring this type of equipment.

Further the three instrument air compressors in the reactor hall, the primary pump building and the ventilation building will not start because there is no emergency power available. As a result, after the buffer stand-by times of the matching compressors, the pneumatic instrumentation transmitters signals will fail to zero and can no longer be read out in the control room. The pneumatic actuators of the primary decay tank bleed valve to the off-gas tank and the ventilation- and off- gas systems will fail to safe and close the matching pneumatic operated process valves. In Table 5-8 the consequences of loss of HFR instrument air are listed. The buffer times in the table are worst case because no credit is taken for the buffer capacity of the ECN buffer tanks at the compressor building and the OLP PE-piping network.

Other NRG-facilities:

At LOOP-SBO in the HCL/MPF the emergency compressors will cease to operate. The tool air systems here have a buffer vessel capacity of some minutes that are sufficient for safe closure of all ventilation isolation valves. Further a number of small application tool air user can be identified viz. DWT, NRG-RE and Fermi where tool air is used for non-safety-relevant pneumatic control systems.

Parameter or actuator	Consequence in case of LOOP-SBO1	Buffer time (hours)
Primary flow	None; flow already zero by LOOP	**0.5**
Primary pressure	Limited; sufficient redundant electronics	**0.5**
Level basin demin tanks	Limited; redundant local instruments	**0.5**
Experiments instrumentation	none, no use of instrument air for experiments	**na**
Level basins	Limited; sufficient redundant electronics	**0.5**
Flow reactor basin	None; flow already zero by LOOP	**0.5**
Primary outlet temperature	Limited; sufficient redundant electronics	**0.5**
Gas activity monitoring	Loss of scavenging air of monitors off-gas and slow-down room, readings will increase with 38 %	**2**
Containment pressure	Loss of under-pressure control function	**2**
Actuator decay tank bleed	Isolation of decay tank from off-gas system	**na**
Actuators ventilation system	Containment isolation	**na**
Actuators off-gas system	**Isolation off-gas system**	**na**

Table 5-8: Consequences of loss of HFR instrument air

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

It is envisaged that basically loss of electrical power endangers the confinement function of the facilities under consideration. Small releases inside the buildings, causing small releases to the environment may occur. Extent and content of those releases have to be determined.

Mainly for SBO situations long term monitoring and recording of radioactive releases is not available as back up batteries have a stand-by time of approx. 30 min. As duration of releases in case of SBO is not clear, increase of battery capacity to enlarge monitoring and recording time may be considered.

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 Research Location Petten (Onderzoeks Locatie Petten, OLP) accommodates a number of institutes and companies. The main organizations are:

- Energy research Centre of the Netherlands (ECN),
- Nuclear Research and consultancy Group (NRG), operating the nuclear facilities,
- Institute for Energy and Transport (IET) of the Joint Research Centre (JRC) of the European Commission, and
- Covidien (formerly Mallinckrodt Medical) that owns a number of buildings (e.g. a building housing two cyclotrons) and utilises some ECN/NRG facilities.

Besides these four organizations there are various facilitating companies present on the OLP, amongst others the OLP security, hired by NRG.

All nuclear facilities operated by NRG are owned by ECN except for the High Flux Reactor (HFR), which is owned by the European Commission, and the recently built Jaap Goedkoop Laboratory (JGL) owned by NRG itself. As licensee of these facilities NRG is responsible for their safe operation. NRG operations are compliant to the NRG management system, which amongst others covers the procedures for maintenance, notification of incidents and malfunctions. NRG has a so-called complex license in compliance with the Dutch Nuclear Energy Act (Kernenergiewet, KeW) to operate the nuclear facilities of ECN/NRG, and a separate license to operate the HFR (by decree of January 7th, 2005).

The emergency planning and organisation have been synchronised with the regional emergency plan and the national crisis organisation as defined in the National Plan for Nuclear Emergency Planning and Response (Nationaal Plan Kernongevallenbestrijding, NPK. They have been submitted to the regulatory body prior to implementation.

The licensee is responsible for on-site emergency responses as well as for providing plant status information to the authorities to support 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

High Flux Reactor (HFR)

The HFR is permanently staffed (5 up to 7 operators in 5 daily shifts), which include staffing of the control room (2 to 3 operators). A guard building provides controlled access to the HFR area. This building is permanently staffed by a group of at least 3 security guards (hired by NRG), operating in 3 shifts. Materials enter or leave the HFR area by a gate controlled from the guard building. The guard building also houses the Central Alarm Station (CAS[33]), see below.

Hot Cell Laboratories (HCL)

The HCL comprises the Research Laboratory (HCL-RL) including the Actinide Laboratory (HCL-AL), and the Molybdenum Production Facility (HCL-MPF). The HCL guard is either staffed (during working hours) or on call (pager). The HCL-RL itself is only staffed during working hours, the HCL-MPF only during Molybdenum production (twice a week for 8 hours).

Other nuclear facilities

The WSF, DWT, JGL are staffed during working hours only, the LFR and STEK hall are not staffed but supervised by the HCL guard. Outside working hours these facilities are supervised by the corporate fire brigade supported by the CAS. Processes in these facilities are monitored by the corporate monitoring network (Bedrijfbeheersysteem), which will alert the CAS in case of fire or system failures.

Supporting facilities

The corporate fire station is permanently staffed (6 firemen in 3 daily shifts). Also the Central Alarm Station (CAS), located in the HFR guard building, is permanently staffed (1 - 2 persons). All other supporting facilities are staffed during working hours only and supervised by one of the permanently staffed facilities outside working hours and the corporate monitoring network supervised by the CAS.

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[33] Also known as CMP (Centrale Meldpost)

6.1.1.2 Plans for strengthening the site organisation for accident management

Besides regular updates, as part of the operational plans of NRG and the OLP emergency response organisation ERO[34] organisation several actions are foreseen in order to improve ERO, including a clearer definition and more accurate distribution of alarm roles, and enhancement of the nuclear facilities' specific emergency procedures.

6.1.1.3 Measures taken to enable optimum intervention by personnel

General set-up of the emergency response organisation

In order to have an appropriate response in case of an accident at the Research Location Petten (OLP), the organisation of emergency response is managed at four consecutive levels, see Figure 6.1

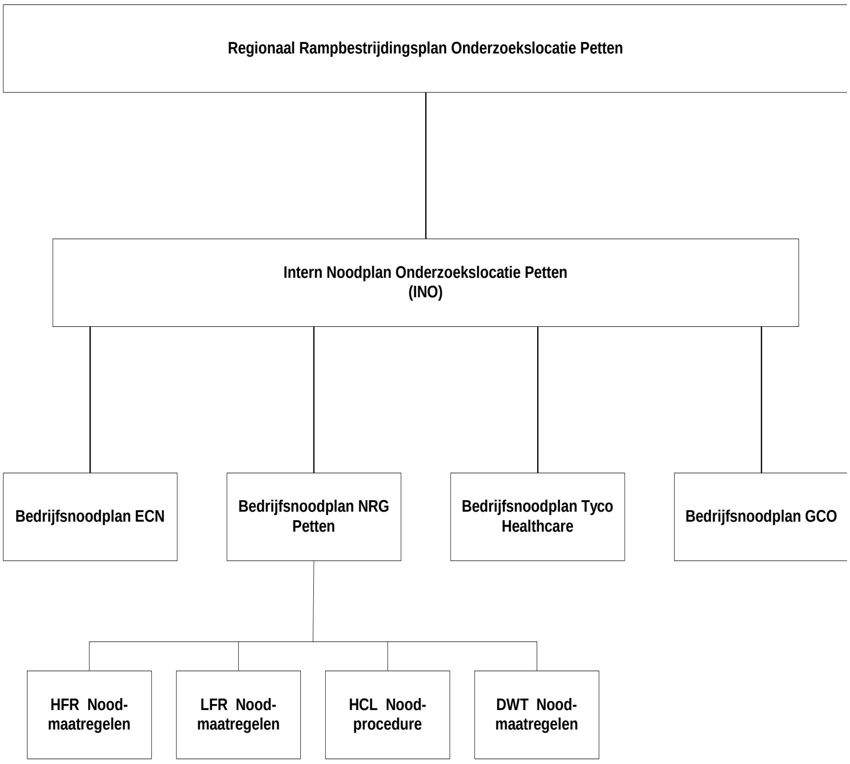

Figure 6.1: Hierarchy of emergency response plans for the OLP.

[34] Also known as INO (Interne Noodorganisatie)

- The facility-specific emergency measures and procedures (Noodmaatregelen & -procedures), under the umbrella of the corporate emergency plans cover all minor facility-specific incidents.
- The corporate emergency plans cover all minor incidents; Bedrijfsnoodplan NRG Petten (BNP) covers those at the NRG operated facilities.
- The general site-wide emergency plan for the OLP (Intern Noodplan Onderzoekslocatie Petten)[35] governs the emergency response actions on the OLP and the communication with the off-site response organisations in case potential consequences of accidents might not be limited to the NRG facilities.
- The regional disaster response plan for the OLP (Rampenbestrijdingsplan Onderzoeks- en bedrijvenlocatie Petten, RBP) describes the emergency response by the off-site/on-site organisations in case consequences of accidents may affect areas around the OLP. This regional response is backed-up by the national NPK organisation.

In case the RBP is activated, the coordination of the emergency response is taken over by external regional organisations. If the event is scaled-up to a nuclear incident, the national nuclear contingency plan NPK will be activated and the responsible organisations on national level will be called in.

Emergency Response Organisation

The emergency response organisation (ERO) of the Research Location Petten (OLP) 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. ERO 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 during emergencies;
- aftercare of an accident.

In case of any incident on the OLP, the central alarm station (CAS) will be alerted. Three potential triggers may activate the corporate emergency plan (BNP):

[35] Den Hartogh. G.P.J. *Intern Noodplan Onderzoek Locatie Petten*. ECN/NRG/Mallinckrodt Medical B.V./GCO report FDSEC/5243 version 3, November 2007

- a (potential) unsafe situation is identified
- an accident or emergency situation is identified
- alarm systems connected to the corporate monitoring network (Bedrijfbeheersysteem)

When the response to an incident can be handled by the BNP, i.e. when the potential consequences will be limited to the facility, the CAS will alert the corporate fire brigade which will support the facility staff in repressing the incident supported. If required, NRG will inform both the Dutch Nuclear Regulator (KFD) and the municipality Zijpe.

When an event is beyond the capabilities of the facility staff, or might lead to consequences beyond the NRG's facilities, the Commander of the fire brigade will activate the OLP emergency response organization (ERO). Besides, ERO can be activated directly by the nuclear facility's staff by reporting 'code ERO' to the CAS. Subsequently, CAS will page the members of the ERO crisis team on duty. This ERO crisis team will meet in one of two existing ERO crisis rooms at the OLP prepared for such occasions, both containing the necessary communication means and background documentation needed for the ERO crisis team to perform its task.

The ERO guideline describes the necessary measures to enable communication and cooperation between staff of the various facilities on the OLP. The ERO crisis team is chaired by the Head ERO on duty. Once ERO is activated, its directions have priority over the general guidelines given by the BNP and its supporting procedures, which of course will remain the basis of response actions and measures.

ERO is a scalable emergency response organisation and covers all types of emergencies: personal injuries, conventional or nuclear incidents etc. This means that the number of emergency staff alerted is determined by the scale of the incident. Head ERO will always be in charge.

6.1.1.4 Use of off-site technical support for accident management

Communication

In case the OLP disaster response plan (RBP) is activated, emergency communication starts with a call from the ERO crisis team to the notification point of the local authorities in Zijpe and to the notification point of the national authorities at the Ministry of Infrastructure and the Environment in The Hague. Subsequently the Mayor of Zijpe will formally take the lead and the regional notification point Meldkamer Noord-Holland Noord (MKNHN) will take over the communication between the regional emergency response organisations, the municipalities and the NPK organisation.

Fire brigades

In case on an emergency, the corporate fire brigade provides a fast emergency response team of six firemen and one fire engine, which is permanently stand-by. Additional support can be called in from neighbouring fire stations in Callantsoog and Schagerbrug. If needed, support of more distant fire stations in Groet and in Schoorl (response times 16-18 min.) can be called in. In case extended response is required also support will be requested from fire stations in Den Helder, Warmenhuizen, Niedorp, Julianadorp, Middenmeer, Wieringerwerf or Alkmaar (response times 30+ min.). The RBP describes several routes for up-wind approach in case toxic gases or other potential hazards are involved.

Decontamination units

In case decontamination of emergency workers is required beyond the capabilities available at the Hot Cell Laboratories (HCL) and High Flux Reactor (HFR), mobile decontamination units from the Safety Region Noord-Holland Noord can be called in and set-up on the leakage proof outdoor storage area of the Decontamination & Waste Treatment (DWT) facility, providing controlled decontamination conditions. The regional response organisation can provide a mobile decontamination unit transported by container (so-called haakarmbak). Additional decontamination units can be provided by neighbouring Safety Regions like Kennemerland. In addition, commercially available mobile decontamination units of the firm Reijm can be obtained.

Police forces, hospitals, and army

Presently there are arrangements with the hospital in Alkmaar (MCA) for treatment of (potentially contaminated) victims. See also paragraph 3.3 of the national nuclear contingency plan NPK.

Other technical support

Furthermore, the regional response organisation can provide two types of emergency response vehicles, both provided with the same basis equipment, such as foils, pumps, hydraulic equipment needed for rescue operations. One of them is additional equipped with a 50 kN winch, a 20 kVa diesel generator, additional rescue equipment as well as with equipment needed for emergency response in case of accidents with toxic and radioactive materials. When required the NPK organisation can provide technical equipment such as two mobile monitoring facilities (vans) present at the RIVM (response time ca. 2 hours after alert). With these vans, radiation monitoring, detection of toxic gasses and radionuclide specific measuring of smear tests can be performed.

Technical equipment, such as barges with hoisting equipment, can be obtained from contractors in Den Helder, Langedijk, Niedorp or Alkmaar. Additional technical support such as manpower and equipment can be provided by the regional emergency response organisation.

6.1.1.5 Procedures, training and exercises

Processes and situation in the NRG facilities are continuously monitored by local on-line monitoring systems connected to the corporate monitoring network (Bedrijfbeheersysteem), watched over by the Central Alarm Station (CAS) responsible for alarming in case of detection of fire, radiation, etc.

All procedures governing incidents and accidents are described in Section 6.1.1.3, based on NRG's management system. For each facility either general or specific procedures and guidelines exist describing actions during emergency situations. All procedures are in compliance with the legal framework for radiation protection, see Appendix 1.1.1A.13.

In case of radiological and nuclear accidents, the radiation protection organisation at NRG plays a major role. The responsibilities, competency and tasks of this organisation are described in the NRG Licence for the nuclear facilities (KeW License, Part 3 of the Safety report. In accordance with the KeW License, local radiation protection supervisors are part of the staff of each of the nuclear facilities operated by NRG. They are supported by assigned and qualified radiation health physics experts. Local radiation monitoring and registration systems (external radiation as well as airborne radioactive material) provide data to the central radiation monitoring (CRM) network enabling the general radiation protection supervisor to monitor any radiological event situation inside as well outside the nuclear facilities. Also the OLP emergency response organisation (ERO) has access to this system.

Employees of OLP organisations involved in handling of radioactive material or radiation sources are trained to handle these (radiation protection skill levels 2 - 5). Trained radiation protection experts are assigned to each nuclear facility and are alerted in case of incidents (radiation protection skill levels 2 or 3). The fire station is permanently staffed by professional firemen (see Section 6.1.1.3). The corporate fire brigade is assisted by volunteer firemen, trained in first aid and fire fighting as well as corporate certified first responders (BHVs) spread over the different buildings of the OLP.

As part of its Operational Plan, in consultation with its stakeholders the fire brigade annually prepares an exercise plan. Besides general exercises of the fire brigade in a training centre, this plan covers exercises of specific ERO scenarios or the corporate emergency plan BNP concerning various facilities on the OLP, in which the fire brigade and its supporting volunteers partake.

Once a year, in consultation with the OLP, the Safety Region Noord-Holland Noord (VNHN) organises exercises for ERO and RBP in which the Head ERO, the managers of company specific emergency plans

(BNP) of the four companies as well as the commander of the fire department take part. The evaluation of these exercises is used as a means to improve the ERO organisation.

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)

Formally, only the two fire engines stationed at the OLP fire station can be deployed by the SAM organisation. No systematic inventory was ever made concerning other vehicles present on the Research Location Petten (OLP) that might be useful for SAM. Nevertheless, a number of vehicles being used for every day work on the OLP can certainly be brought to bear in case of severe accidents.

A four-wheel drive medium sized truck (Unimog) is normally used for transport and towing of heavy objects on the OLP. In case of an event damaging a building on the site, the latter capability can be used for removing parts of the building to provide access for rescuing staff members or precious needed equipment. This truck can also be used to transport materials and equipment for mitigation of accident consequences elsewhere on the OLP. In addition, there are several forklift trucks stationed at the HFR site as well as at the DWT and HCL facilities. Though these have clearly less power and capacity as the Unimog, they can be used for the same purposes. Beside these, there are many other vehicles potentially available on the OLP, e.g. the truck of the company Ott, a contractor at the OLP.

As further mobile equipment shielded containers present on the OLP should be mentioned. These might be used directly for shielding purposes as part of accident mitigation actions, but can also be used to store radioactive (contaminated) materials.

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

Water and diesel supplies related to emergency systems and stationary or mobile emergency equipment are present at several locations on the OLP. A description can be found in Chapter 1.

6.1.2.3 Management of radioactive releases, provisions to limit them

The strategies used to manage and limit radioactive releases after a severe accident at a nuclear facility on the Research Location Petten (OLP) are described in general procedures (e.g. VGWM 2.8 and 2.9) as part of the corporate emergency plan (BNP), and in detailed procedures given in the facility-specific emergency plans. Specific work instructions are available for handling of situations in the HFR, the fire department and for handling of radiation incidents and radioactive contaminations.

As any contaminated water produced during normal operation of the nuclear facilities on the site, also contaminated water resulting from accident mitigation can be stored in the pools in the Water Treatment building of the Decontamination & Waste Treatment (DWT) facility, for subsequent treatment.

6.1.2.4 Communication and information systems (internal and external)

Several communication and information systems exist for communication in case of emergencies, both for on-site and external communication, summarised in Table 6.1 and Table 6.2.

from	by	to
employees, team managers, responsible unit managers, ERO crisis team, other OLP companies	**public wired and wireless telephone lines (KPN and GSM, respectively)**	**CAS**
CAS	regional text oriented mobile communication system (Semafoon P2000)	commander fire brigade[36]
CAS	regional mobile wireless communication (Portofoon C2000[37])	commander fire brigade
CAS	local mobile wireless communication (Mobilofoon)	security service
CAS	direct telephone line	control room HFR
CAS	direct telephone line	commander fire brigade
CAS	intercom	fire station, control room HFR
CAS	pager message system[38]	ERO team, BHV team
CAS	public wired and wireless telephone lines (KPN and GSM, respectively)	ERO team, employees, team managers, responsible unit managers

Table 6.1:Internal communication and information systems

Besides the these internal communication systems, the CAS can be alerted by alarm systems either triggered automatically or manually. The alarm systems include loss of under-pressure alarms, fire alarms

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[36] currently only used for external communication

[37] Note that the portofoon (C2000) infrastructure is not available in the shielded dome of the HFR

[38] Semafoon, Semadigit, or Semscript system

and gas detection systems. The CAS room also has access to a number of systems that are part of the OLP security system. Furthermore, several other information systems exist at the OLP:

- each building has a signal horn to accomplish emergency evacuation of the building
- outside on the OLP emergency alert signal horns are present
- the radiological monitoring network (CRM) allows access to on line data from almost all stationary radiation monitoring sensors installed in the facilities and on-site.

from	by	to[39]
CAS	direct connected emergency telephone lines (Nationaal noodnet)	MKNHN, Mayor of Zijpe, ministry of EL&I, NCC, RIVM
CAS	public wired and wireless telephone lines (KPN and GSM, respectively)	responsible unit managers MKNHN, Ministry of EL&I, KFD, other OLP companies
CAS	regional text oriented mobile communication system (Semafoon P2000)	responsible unit managers MKNHN
CAS, fire brigade	regional mobile wireless communication (Portofoon C2000)	external response teams ROT, NHN
CAS, fire brigade	local mobile wireless communication (Mobilofoon)	external response teams
ERO crisis team	fax, e-mail	external response teams and organisations etc.

Table 6.2: External communication and information systems

Besides the above mentioned external communication systems, the internal computer network is connected to the external network and can potentially be used for communication as well as to collect and exchange information.

To inform the public, the off-site emergency organisation has the following external communication systems at its disposal:

- access to radio and cable TV (public communication in Zijpe)
- loudspeaker vans (public communication in Zijpe)

39 MKNHN: regional notification point Meldkamer Noord-Holland Noord; NCC: National Coordination Centre; RIVM: Netherlands National Institute for Public Health and the Environment; KFD: Inspection of the regulatory body; ROT: regional emergency response team (Regionaal Operationeel Team) ; NHN: Safety region Noord-Holland Noord

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

Extensive destruction of the infrastructure around the plant, including interruption of external communication, constitute as such no impediment to deploy the accident management measures on the OLP as described in the corporate emergency plans and facility-specific emergency plans.

The OLP is accessible from the Westerduinweg (N502) by two entrances on the east side. The N502 is connected with the inter-provincial road N9 by two roads, the Zeeweg in the north and the extension of the N502 in the south. The OLP can also be reached by a dirt road / bicycle path from the west, from the beach. Access to the OLP site by external emergency forces may potentially be hindered when the trafficability of its access roads is compromised. A severe flooding due to a dike or dune breach would severely affect a large area for a longer period of time (Chapter 3): such effect will not be localized – as for events like explosions/blasts, fires and an airplane crash (Chapter 7) – or probably minor as for an earthquake (Chapter 2) – or very temporarily – as for extreme weather (Chapter 4).

Due to the elevation of the OLP, the site itself would probably not be flooded or only for a short while; effectively the OLP would become part of an elongated dune island between Petten and Callantsoog. Though a flooding of the hinterland will not directly affect the nuclear facilities on the OLP, it is likely that it will affect relevant parts of external infrastructures, the supply of resources and the accessibility of the OLP for employees, emergency forces, suppliers etc. At the same time the external emergency organisation would have to face major challenges in managing large-scale evacuations and flood prevention in the region surrounding the OLP.

Based on identified weaknesses in dikes and dunes, after a dike breach most probably the main part of the city of Den Helder would be flooded. In such case it would be unlikely to expecting support equipment and supplies from Den Helder. As alternative to Den Helder such resources might be supplied from Langedijk, Niedorp or Alkmaar, though trafficability of the roads connecting these locations to the OLP is questionable. However, flooding is an event which risk would certainly be forecasted well in advance, enabling the OLP to take precautionary measures and shut down vulnerable nuclear facilities.

6.1.3.2 Loss of communication facilities/systems

A limited assessment has been performed concerning the vulnerability of the communication means of OLP’s SAM organisation discussed in Section 6.1.2.4, subject to the set of external events considered in the present report. Note that an identified vulnerability does not mean that a specific communication means would fail, but is rather a recommendation to assess a particular item in more detail to strengthen its robustness or look for alternatives.

Table 6.3 gives an overview of a conservative expert judgement assessment on the vulnerability of the communication infrastructure in case of external events. Appendix 1.1.1A.14 discusses the vulnerability of the communication means presented in Table 6.3 in more detail.

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 of the emergency organisation due to consequences of accident/event on site may arise for to several reasons:

- Impairment/loss of communication means: see Section 6.1.3.2
- Impaired accessibility of nuclear facilities: see sections 6.4.2.2 and 6.1.3.4
  In case of high local dose rates, portable dosimeters, shielding, protective clothing, respirators and limitation of the exposure time will be used to keep the worker's dose within the regular limitations.
- Impaired accessibility of supporting facilities: see sections 6.1.3.5 and 6.1.3.6.
- Impaired accessibility of the OLP: see Section 6.1.3.1

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

On the OLP, only the HFR is equipped with a control room. The accessibility and habitability of this control room is addressed in Section 6.4.2.5. In case of the HCL there is no control room. However, tools and devices used for operational control are present in the HCL, too[40]. The accessibility and habitability of the HCL is addressed in Section 6.4.2.5 as well.

[40] Note that in the HCL-MPF for safety reasons (exothermic reaction) the first step of an isotope production cycle – dissolution of a target – must be completed once it has been started. In case of evacuation without proper action this might lead to a release inside the hot cell. As over-pressure cannot be excluded, this might lead to dispersion beyond the cell causing contamination inside the HCL-MPF building with mostly short-lived radionuclides. Restoration would mainly involve decontamination after proper decay, and repair of the hot cell.

Vulnerability of elements in SAM related to communication	External event: flooding	earthquake	fire[41]	extreme weather conditions	aircraft crash	cyber-attack	explosions/blasts
Availability of CAS room (single room)	-	-	-	-	-	o	-
Availability of ERO room (dual rooms)	-	-	+	-	+	-	+
Internal wired telephone infrastructure (KPN)	o	-	-	-	-	-	-
Internal wireless telephone infrastructure (GSM)	-	-	-	-	-	o	-
Connection to external wired telephone infrastructure (KPN)	-	-	o	-	o	-	o
Connection to external GSM infrastructure	-	-	-	-	-	o	-
National wired emergency line (Noodnet)	-	-	-	-	-	-	-
National wireless emergency network P2000 (Semafoon) & C2000 (Portofoon)	-	o	o	o	o	-	o
Local wireless communication network (Mobilofoon)	o	o	o	o	o	o	o
Pager (ERO representatives, BHV[42])	-	o	o	o	o	o	o
Intercom	+	-	-	-	-	+	-
Direct line CAS - HFR	+	-	-	-	-	+	-
Direct line CAS - fire department	+	-	-	-	-	+	-
Fax (ERO room)	-	-	-	-	-	-	-
Internal computer network infrastructure	o	-	-	-	-	-	-
External computer network infrastructure	-	-	o	-	o	-	o

'-' = vulnerable, 'o' = unknown, '+' robust

Table 6.3: Vulnerability of elements in SAM related communication

[41] here a fire outside the nuclear facility is meant, i.e. outside the OLP or in a non-nuclear facility on the OLP
[42] BHV: corporate certified first responder (Bedrijfshulpverlener)

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 crisis teams will meet and work in one of both prepared ERO rooms. Which ERO room is chosen depends on the location of the incident and the actual or predicted wind direction. Taking into account the prevailing wind directions (south to west), the ERO room in the Forum building would probably be less affected by incidents at the nuclear facilities on the OLP. In case of north-west to north-easterly wind directions, a choice for the ERO room on the JRC premises would be obvious. In case of flooding of the OLP, an extreme earthquake or extreme weather, both ERO rooms might be affected, impairing the ERO considerably.

As discussed in Section 6.1.2.4, the CAS has a crucial communicative role in case of an emergency, both internally and externally. The CAS is located in the HFR guard building (see Chapter 1). In case due to an external event the CAS room (and its facilities) is no longer available or habitable, currently no backup room is equipped to serve as an alternative for the CAS room.

Management of the accident in the facilities themselves is discussed in Section 6.4.2.2.

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 effects of a flood on the facilities on the Research Location Petten (OLP) have been assessed in Chapter 3. In case of a large flood, due to the elevation of the OLP the site itself would not or only shortly be flooded, effectively becoming part of an elongated dune island between Petten and Callantsoog. Though feasibility and efficacy of accident management measures would most probably hardly be affected, still the most vulnerable facilities would be shut down as precautionary measure accommodating to the forecasted flooding risk.

The critical parts of the nuclear facilities on the OLP are well protected against earthquakes. Even if in case of damage some alternatives may no longer be available, numerous alternatives remain to shut down these facilities in a safe way.

6.1.3.7 Unavailability of power supply

The direct consequences of the unavailability of off-site power as well as analyses of potential failures of emergency power supplies are discussed in Chapter 5. That chapter also provides information on the autonomy period of these systems and the ways in which that affects the development of accident scenarios. The effects of unavailability of power supply on communication means is discussed in Section 6.1.3.2. The effects on resources for the nuclear facilities in severe accidents conditions are addressed in the sections 6.1.3.3 and 6.1.3.4.

No detailed analysis has been performed (yet) on other aspects like food supply, accommodation of employees and emergency forces, or sheltering for injured/contaminated persons. The GBD building provides ample supply of radiological equipment.

6.1.3.8 Potential failure of instrumentation

High Flux Reactor (HFR)

During severe accidents the plant diagnostics and considerations concerning severe accident management strategies are keyed to a limited number of plant parameters. Most of the instrumentation important for that purpose is qualified for (severe) accident conditions and harsh environments. Currently, a project for implementation of Accident Management Procedures is running (project Accident Management-1). In that project the need for additional instrumentation is considered too.

During severe accidents at the HFR not all available instrumentation can be relied on. Accident management measures will be taken on the basis that any instrument that is believed to provide useful information should be used, even those not formally qualified for the concerned conditions. The operator has to follow proper emergency procedures trusting mainly the qualified instruments, assisted by application of portable instruments. As in severe accident conditions unqualified instrumentation will not fail right away, it may still be valuable and provide supplemental information.

It should be noted that during a severe accident for many instruments the environmental conditions may remain within their design basis. For example, for severe accident scenarios involving core damage the containment pressure and temperature conditions will not exceed those for design basis accidents for many hours. Qualified instrumentation will continue to function properly in that period of time. In some scenarios however (i.e. prolonged boiling of pool water) the environmental conditions will be so harsh that some instrumentation will not survive. Information on survivability of the instrument sensor or transmitter can be found from the data sheets, listing information on the environmental test data.

Furthermore, in the phase prior to occurrence of fuel damage the environmental conditions are in general within the design basis of the instrumentation. Instrumentation which is not harsh environment qualified is also available in that phase of the accident.

Other nuclear facilities

The other nuclear facilities have amongst others instrumentation to monitor building pressure and building activity. The qualification of this instrumentation is presented in Section 6.3.7.

6.1.3.9 Potential effects from nearby installations and emergency staff restrictions

Potential effects from nearby installations

Incidents and accidents at the Research Location Petten (OLP) either inside the facilities or during outside actions such as transport might result in releases of airborne radioactive or toxic substances and/or enhanced radiation levels in the vicinity of the accident location. Locally enhanced radiation levels and/or local concentrations of airborne toxic and/or radioactive substances may impact the operability of nearby facilities as well as required actions at the OLP, e.g. emergency response actions.

Except for the High Flux Reactor (HFR) and the Molybdenum Production Facility of the Hot Cell Laboratories (HCL-MPF), all nuclear facilities on the OLP can safely shut down their operations almost at once, leaving the facility in a safe and stable operational state. With respect to the HCL-MPF, once a molybdenum production cycle is started, operators need 2 hours to safely complete at least the first exothermic step and 8 hours to complete the full production cycle. In order to shut down the HFR into a safe and stable state, the HFR control room has to be staffed during 3 subsequent days. Smoke might enter the facilities through their ventilation systems. To prevent this from happening, the facility's ventilation inlets could be closed. However, over time this might impair the habitability of these facilities. To support operation in such conditions the HFR control room is provided with respirators and evacuation kids (suits and mask). Note that also the supporting facilities needed by the SAM organisation might be affected by accidents at the OLP or nearby areas.

With respect to the critical facilities identified above an assessment of the possible influences of releases at one location on work conditions in other facilities on the OLP (see Appendix A.15) did show that:

- the HFR might be affected by smoke from a fire or radioactive release at Covidien as well as by smoke from a fire at the central storage of ECN in case of south-east to south-westerly winds (↑)
- the HFR might be affected by smoke from a fire at the JRC complex in case of north to north-easterly winds (↓)
- the HCL-MPF might be affected by smoke from a fire or radioactive release at the Covidien building or smoke from a fire at the central storage of ECN in case of south-west to north-westerly winds (→)
- the HCL-MPF might experience minor impact by smoke of a fire or radioactive release at the HFR in case of north to north-westerly winds (↓)

- the CAS room, located at the HFR site, would experience the same effects as the HFR, including smoke of fire or radioactive release at the HFR

In case of a severe accident at the HFR leading to an internal release of radioactivity into the containment of the HFR reactor building – e.g. due to melt of an in-core experiment or even part of the core itself – to limit external exposure also staff present in neighbouring facilities might need to be evacuated temporarily, depending on the extent of the release (see Appendix A.15). The radionuclides involved in such internal releases are merely short-lived, causing dose rates to drop by several orders of magnitude within days. Hence, possible access restrictions for neighbouring facilities would be of short duration.

The CAS room is the only part of the SAM organisation without any planned backup provisions. It is recommendable to improve protection of the HFR control room, the HCL-MPF as well as the CAS room against toxics, smoke, etc.

Emergency staff restriction

Besides the on-site emergency forces as discussed in Section 6.1.1.3, in case of larger events, (e.g. lager fires or fires at multiple locations), support from off-site organisations can be requested (see Section 6.1.1.4). In case of limited or no off-site support (e.g. as result of a severe flooding, see Section 6.1.3.1), a lack of emergency staff might occur if several incidents have to be dealt with at the same time. This might be partially compensated by deployment of staff members of the nuclear facilities as well as members of the OLP BHV organisation, applying local fire extinguishing equipment.

6.1.4 Conclusion on the adequacy of organisational issues for accident management

Communication of incidents

The NRG management system, the nuclear facilities emergency guidelines and the OLP's site-wide emergency response organisation (ERO) guideline, provide clear guidance on the communication lines in case of an emergency and the responsibilities of the different actors. In case of severe accidents, additional plans are available – the OLP Disaster Response Plan (RBP) and National Plan for Nuclear Emergency Planning and Response (NPK) – that describe the communication between the different external and internal emergency response forces and their expert organisations on a detailed level. The different guidelines are evaluated – and if necessary – updated in regular intervals, and regular exercises are performed in order to train the emergency response organisation. All internal communications lines on the OLP come together in and are handled by the central alarm station (CAS), which can be considered as an efficient solution. However, because no backup room for the CAS is currently foreseen, inhabitability or a technical failure of the CAS room may impair the efficiency of the emergency response organisation. Furthermore, the present emergency procedures give no guidance how to deal with a temporary failure of the CAS. Furthermore, the CAS is the only (reported) location where the information on responsible staff members in case of an emergency (alarm roles) is stored. Therefore, it is recommendable to provide additional locations for this information, creating redundancy in severe accident management.

Currently, no detailed assessment exists concerning the availability of the different communication means foreseen in the emergency plans in case of external events. Communication on-site can be potentially impaired by external events (flooding, earthquake, lightning strike, fire, explosions), either directly or as consequence of (long-lasting) LOOP. Because communication is a cornerstone for the efficient operation of the SAM organisation, the OLP emergency plan should anticipate potential failures of the technologies it relies on.

A provisional assessment of the vulnerability of the communication network in case of severe accidents revealed a number of weaknesses. Recommendations to improve the reliability of the communication systems are:

- To improve the C2000 emergency network, a repeater could be installed inside the HFR.
- To improve the communication means of ERO the present communication means could be supplemented with additional communication means that are independent from on-site and land-based off-site network (e.g. walky-talky, satellite telephone/internet, cell phones). This particularly applies for the communication between CAS and HFR.
- The vulnerability of the GSM network as a valuable means for internal and external communication in case of a severe accident should be assessed. Its usefulness could be improved by clear guidance for its use for on-site communication is case of an emergency.

Communication to external emergency organisations

After severe flooding, failure of communication to off-site organisations may occur affecting both wired and wireless telephone networks (KPN, GSM) as well as the emergency communication via Noodnet, C2000 and P2000, either as direct effect of the flooding or as indirect consequence of a large, regional power failure. Application of satellite-based communication would strengthen the off-site communication.

On-site emergency organisation

In case of a severe accident or a threat that could potentially lead to a severe accident, the ERO crisis team assesses the accident situation based on information provided by the manager of the facility concerned, and communicates this with external emergency response organisations and their experts. Presently, the general responsibility of the ERO to coordinate such situations is described in the ERO documents. According to the facility-specific emergency procedures, the staff of the facility will be engaged in emergency response actions. Besides its role as communicator and advisor, ERO should also act as coordinator of the on-site response actions. In case off-site response organisations are unable to assist (e.g. after a large scale flooding), ERO will remain responsible for coordination of mitigating actions taken by the staff of the facilities in accordance with the facility emergency.

However, besides the fire department, the BHVers[43] and ERO, presently there is no official overall emergency response organisation that may combine all potential resources present at the different facilities. No procedures exist nor have exercises been performed concerning mutual support of the staff of the nuclear facilities enabling site-wide emergency response actions such as restoring of infrastructure and support facilities on the Research Location Petten (OLP). No specific ERO procedures exist for restoration of support functions (electric power, water supplies) or for providing first aid to wounded and/or contaminated personnel at designated locations at the OLP.

Furthermore, no written procedures exist that foresee activation of the ERO state in case of an external event such as approaching extreme weather condition or a large fire in the vicinity of a nuclear facility.

In summary, the following number of conclusion can be drawn:

- Currently, the ERO does not foresee a combined attempt of all facilities to handle emergency situations on site-level.
- No plans exist for using available equipment at the OLP for restoring safety functions such as confinement and shielding in the nuclear facilities in case off-site emergency forces are not available.
- Presently, the coordinating role of ERO is not triggered by threatening external events and does not include the handling of them.
- The role of ERO in restoring the support functions is not clear.
- No plans exist for sheltering of personnel, when personnel cannot leave the OLP.
- Both ERO rooms as well as the CAS room are vulnerable for external events like extreme earthquakes, flooding of the hinterland or severe weather.
- The CAS room is vulnerable for local events like fire, toxic releases or near-by explosion.

[43] Bedrijfshulpverleners (Corporate certified first responders)

Off-site emergency organisation

The emergency response guidelines provide detailed roles for external emergency forces. In case external events such as severe flooding, severe earthquake or extreme weather induce serious damage to the local infrastructure, such as roads, or power network, the (on-site) emergency response organisation may have to depend on the available resources on the Research Location Petten (OLP). In particular, in case of a severe flooding, it will impact a large region area surrounding the OLP over a longer period, affecting several relevant physical resources and impairing off-site emergency organisations. Even in case the OLP would still be accessible by external emergency organisations, it is questionable whether sufficient priority would be given in case of less severe accidents on the OLP. Such a situation is not anticipated in the OLP emergency response plans, and it is unclear whether the present provisions suffice in such a case (e.g. diesel fuel for emergency power is limited to ca. 3 days). Though crucial for SAM concerning organising its external support, the current provisions described in the emergency response plans would not be sufficient to sustain a reliable communication with off-site organisations over longer periods of time.

Resources on-site

Limited resources are available to the on-site emergency organisation (ERO, including the corporate Fire Brigade). In case of a flooding event, it should also be considered that the following external supplies to the nuclear facilities may be impaired or not available over a longer period of time:

- off-site power
- HFR cooling water supply (from the Noordhollands Kanaal)
- PWN water supply to fill the water make up building (Reinwaterkelder)

The High Flus Reactor (HFR), Hot Cell Laboratories (HCL) and the Decontamination & Waste Treatment (DWT) facility can anticipate on the loss of off-site power or cooling water as part of the concerning procedures. Emergency power supply can be provided by diesel engines on-site, with a limited amount of fuel present on location. Currently, no formal assessment or procedure exists for the case of a long-lasting LOOP[44], in order to verify that the amount of diesel supply present at the OLP is sufficient to enable shutdown of the HFR and to maintain confinement in the other nuclear facilities beyond a period of one week, and to elaborate robust diesel supply routes in case of severe flooding.

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[44] Core cooling of the HFR is ensured for very long periods under provision that the HFR reactor pool is not leaking and limited amounts of water can be supplied to the reactor pool

The current emergency guidelines do not describe possible measures to be taken in case of a flood warning, enabling the SAM organisation to anticipate a potential lack of external resources and replacement staff. In case of a flooding event, it should also be considered, that the following external resources related to the (physical) management/countermeasures of incidents may not available:

- external fire brigade
- fire engine with ladder
- drinking water supply
- electricity supply for on-site
  - radiation monitoring network
  - building ventilations
  - hand & foot monitor/TLT devices
- decontamination units
- ambulances/access to hospitals/medical emergency teams & units
- sick bay / reception area for evacuated/injured non-contaminated persons
- sick bay / reception area for evacuated/injured contaminated persons

6.1.5 Measures which can be envisaged to enhance accident management capabilities

Based on the conclusions in Section 6.1.4 and suggestions in the previous sections, a number of measures, studies and procedures have been identified that would enhance accident management capabilities.

The following measures can be envisaged to enhance accident management capabilities:

- The protection of the HFR control room against toxics, smoke etc. would improve the margin in case of ‘other events’.
- Provision of additional locations of cabinets with alarm procedures (alarm roles) and other contact information will create redundancy in Severe Accident Management.
- Installation of a C2000 emergency network repeater inside the HFR would increase emergency network capability.
- The possibility to provide long-term provisions for employees (e.g. food supply, sheltering) would increase the margin for the ERO.

- The possibility to improve the level of autonomy of the Emergency Response Organisation (ERO) would enhance the capability to handle threatening external events and severe accidents in case of lack of external emergency support due to extreme conditions in the hinterland.
- The possibility to assign a location for sheltering and treat (un-)contaminated and possibly injured people on site would improve the capability of the ERO to handle severe accidents in case of unavailability of external emergency support due to extreme conditions in the hinterland.

The following studies can be envisaged to enhance accident management capabilities:

- An analysis will be performed to evaluate the need and feasibility of a secondary communication room in case of failure of the main CAS room (centrale meldpost).
- An analysis will be performed to evaluate the availability of internal and external communication means in case of external events and long-lasting LOOP and measures will be proposed to meet potential failures.
- The possibilities to improve availability of replacement staff and internal resources that can be used in case of severe accidents will be investigated.

The following procedures can be envisaged to enhance accident management capabilities:

- Provide clear instructions in case of temporary failure of CAS.
- Provide guidance on GSM-based on-site communication, in case of emergency.
- Provide procedures for triggering of ERO by external events.
- Provide procedures for the management of accidents in case of inaccessibility of the Research Location Petten (OLP) for external emergency organizations.
- Define procedures that facilitate mutual support between facilities in case of severe accidents.

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

Chapter 6.2 applies only for the High Flux Reactor (HFR), since all other nuclear facilities on the Research Location Petten (OLP) have no (core) cooling function.

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

Accident Management strategies as last resort to prevent fuel damage as well as cooling restoration strategies are differentiated according to specific safety functions:

- Strategies to restore sub-criticality, see Appendix A.16.1
- Strategies for maintaining sufficient coolant inventory, see Appendix A.16.2[45]
- Strategies for primary heat removal, see Appendix A.16.3[45]
- Strategies for secondary side heat removal see Appendix A.16.4.

Strategies to restore sub-criticality

The HFR operational procedures (Bedrijfsvoorschriften) describe several alternative strategies to reach and maintain sub-criticality of the HFR core, ranging from emergency reactor scram (B-05) to application of the alternative shutdown system (B-24). No supplemental strategies were identified.

Strategies for maintaining sufficient coolant inventory

To maintain sufficient coolant inventory for heat removal of the core, the HFR operational procedures provide several alternatives, ranging from the primary system normally used for this purpose (B-09), the use of pool water (H-13 or H-04), denim water (B-09) and hydrofoor (PWN) drinking water (H-05) to recirculation of leak water (B-9, B-10). Beyond existing procedures, a full pallet of water supplies has been identified which by use of existing connections and by means of mobile pumps can be applied to replenish the primary system either directly or over the pool heat exchanger (see Appendix A.16.2.1, B).

To maintain sufficient coolant inventory for heat removal of spent fuel or the molybdenum production facilities, a subset of the above HFR operational procedures apply, ranging from the use of denim water (B-09), hydrofoor (PWN) drinking water (H-05), and re-circulated leak water (B-9, B-10). No supplemental strategies were identified.

Strategies for primary heat removal

For cooling of the core, the HFR operational procedure E-11 applies, describing both the normal heat removal via the three parallel heat exchangers by means of the primary pumps or, in case of loss of off-site power, both (emergency) decay heat removal pumps. In case of station black-out convection valves can be

[45] Note that the appendix A.16.2 and A.16.3 also present the strategies for maintaining sufficient coolant for heat removal of the molybdenum production facilities.

opened (H-04, H-13) for core cooling with pool water. In case all these strategies fail, one might consider creating/forcing a connection between the primary system and the pool water, beyond existing procedures, e.g. by removal of the observation windows (kijkglaasjes*)* in the reactor vessel head, or by removal of all in-core molybdenum production facilities and/or removal/lifting of the reactor vessel head (see Appendix A.16.3.1, B).

For spent fuel cooling the normal pool cooling system (Bassin Koelwatersysteem) BKWS (E-02) is used, supplemented by convection cooling. Beyond existing procedures, in case of loss of the normal pool cooling system cooling can be provided by water from the hydrant just outside the HFR area over a fire hose connection to the secondary side of the pool heat exchanger, while simultaneously heated water is drained to the secondary heat removal channel. Alternatively heated water might be pumped into two storage tanks, which then act as heat exchanger to the air. Finally, cooling might also be provided by evaporation from the pools, with pool water replenishment from storage tanks or PWN drinking water (see Appendix A.16.3.2, B).

For cooling of the molybdenum production facilities three dedicated systems are used: the normal cooling system (BEKWS) (L-11), the extended cooling system (U-BEKWS) (L-17) and the high pressure cooling system (HD-BEKWS) (L-11). The BEKWS and the HD-BEKWS pumps are connected to emergency power; if the U-BEKWS system pump fails, the BEKWS pump takes over. Occurrence of damage to experiment cooling hoses is not covered by HFR operational procedures. Beyond existing procedures, two actions have been identified to prevent target failure in such case: either heat transport by natural convection must be supported by assuring that the damaged cooling hose remains water-filled and submerged in the pool, or the concerned sample holders could be removed from their (in- or ex-core) positions and stored at the isotope table (see Appendix A.16.3.3, B).

Strategies for secondary side heat removal

Secondary side heat removal is normally provided by water from the Noordhollands Kanaal. After the heat exchanger in the primary pump building, the water is released into the North Sea. Both inlet and outlet of the secondary system are outside the Research Location Petten (OLP). HFR operators have experience with removal of blockages of the inlet, but not with removal of blockages of the outlet as such blockages so far have never occurred. As for the HFR secondary heat removal is no safety function, no operational procedures exist to provide for alternatives. Nevertheless a few alternatives have been identified. One could connect the hydrant just outside the HFR area – connected to the water make up building (Reinwaterkelder) – to the secondary side of the heat exchangers and pump the heated water into the secondary heat removal channel (to the North Sea). As a last resort one could inject (sea)water into the secondary cooling water system via the first de-aerating point. In that case one would need much flexible hose length and probably several mobile pumps (see Appendix A.16.4, B).

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

The Accident Management strategies mentioned in Section 6.2.1 apply both before and after occurrence of fuel damage.

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

The HFR reactor vessel is enclosed in the reactor pool. MELCOR analyses of severe accident progressions involving core melt have shown (1) that the heat produced in a relocated core is partly removed over the vessel wall towards the reactor pool, and (2) that the bottom of the reactor vessel will not fail. Most of the decay heat however will remain contained in the relocated core material, which therefore remains at en elevated temperature for a long period of time. The MELCOR analyses did show that the heat transfer to the reactor pool is sufficient to avoid melt-through of the bottom of the reactor vessel and, consequently, no core-concrete interactions will occur.

Even if in such conditions the bottom of the reactor vessel would fail, there is still the sub-pile room between the bottom of the reactor vessel and the containment basemat, so no relocated core material can ever reach the basemat of the containment.

6.2.4 Measures that can be envisaged to enhance capabilities for core cooling

The following procedure can be envisaged to enhance capabilities for core cooling:

- A set of Accident Management Procedures as supplement to the existing procedures (bedrijfsvoorschriften) will be developed and a training program should be implemented. Examples of recommended issues to be addressed are:
    - accident management measures which are possible at the HFR, but currently not mentioned in a separate procedure
    - possible leak repair methods for larger pool leakage
    - use of autonomous mobile pumps

6.3 Maintaining 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

High Flux Reactor (HFR)

The pressure of the HFR primary system during normal operation is low (maximum approx. 5 bar gauge after the primary pumps). The maximum primary system pressure during accidents is limited by design, as the system has an open connection to the reactor building via the expansion tank. Therefore fuel damage or meltdown at high pressure during accidents is excluded.

Other nuclear facilities

The other nuclear facilities have no equipment or systems in which fuel damage at high pressure conditions could occur. Therefore elimination of fuel damage / meltdown at high pressure does not apply.

6.3.1.2 Operational provisions

This section does not apply because fuel damage or meltdown at high pressure during accidents is excluded by design, see Section 6.3.1.1.

6.3.2 Management of hydrogen risks inside the containment

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

High Flux Reactor (HFR)

As the cladding of the HFR fuel does not contain zirconium, in the HFR the possibility of hydrogen production due to a zirconium water reaction in case of a severe accident in the reactor building or any other building is excluded by design. During normal operation small amounts of hydrogen are produced due to oxidation of the aluminium cladding of the fuel plates. This oxidation process is dependent on fuel type, thermal hydraulic conditions and other boundary conditions. Though the exact amount of hydrogen produced in the HFR has not been quantified yet, based on experience this results in concentrations far below the explosion limit witch is the range of 8 - 12 Vol. % hydrogen. Therefore hydrogen production poses no risk to the reactor building. If the off-gas system is available, the system is easily able to remove the hydrogen.

Other nuclear facilities

As there is no zirconium in the other nuclear facilities, hydrogen production due to a zirconium-water reaction in case of a severe accident is impossible. Furthermore, there is no storage of hydrogen in any of the buildings.

The Research Laboratory of the Hot Cell Laboratories (HCL-RL) is equipped with a water pool of app. 58 m^3 that is used for storage of processed Uranium targets with negligible heat production. The pool water is circulated by pumps through a purification system that has no cooling function. In case of loss of water from the pool the targets will not melt and a possible hydrogen production due to oxidation of the aluminium cladding poses no risk for the building.

6.3.2.2 Operational provisions

Not applicable, see Section 6.3.2.1.

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

High Flux Reactor (HFR)

The reactor building of the HFR is provided with a water lock that opens at an overpressure of 0.5 bar which limits the maximum overpressure in case of severe accidents. This overpressure is well below the leak test pressure of the reactor building (0.55 bar).

MELCOR analyses of a severe accidents have shown that in case of a closed stack the reactor building pressure might increase to a maximum value of 0.4 bar gauge (scenario: loss of secondary cooling, no reactor scram, loss of all primary pumps, loss of decay heat pumps, no injection of water in the primary system. This means that the water lock will stay closed even under severe accident conditions.

The strategies for protecting the containment function of the HFR are presented in Appendix A.16.5. In this unlikely case prevention of overpressure requires steam / gas relief from the containment, for which the following design provisions are available:

- Off-gas re-circulation filter system, furnished with absolute filters and active carbon filters. In recirculation mode the filters of the off-gas system will remove the radioactive aerosols.
- Filters of the ventilation system.

Hot Cell Laboratories – Research Laboratory (HCL-RL)

From the maintenance hall, loading locks and the piping on the first floor the building exhaust air is conducted through a pre-filter and an absolute filter. In addition, the cell exhaust ventilation system is equipped with an active carbon filter, which is used when radioactivity is detected. Furthermore, the building ventilation system is equipped with two single centrifugal fans, both in permanent operation. The flow is controlled by inlet valves. When the building ventilation would stop, the supply fan (as well as the off-gas fans) will stop automatically too in order to prevent overpressure.

Hot Cell Laboratories – Molybdenum Production Facility (HCL-MPF)

The building exhaust air is carried through a pre-filter, an absolute filter and an active carbon filter. To protect the under-pressure in the building, the building ventilation can operate only when the exhaust fans are operating. When the building ventilation would stop, the supply fan will stop automatically too in order to prevent overpressure.

Jaap Goedkoop Laboratories (JGL)

Compared to the HCL facilities the JGL facility deals with much less nuclear material (it is a so-called B Lab which nuclear inventory is limited). The building exhaust air is carried through absolute filters. When the building ventilation would stop, the supply fan will stop automatically too in order to prevent overpressure.

Waste Storage Facility (WSF)

Inside the Waste Storage Facility (WSF) radioactive waste and fissile material is stored in several closed storage areas below ground floor level. Most of the radioactive material is enclosed in drums as well. The ventilation system creates an under-pressure in the underground storage areas, and the filtering system removes any airborne radioactive particulates. Air dust contamination is continuously monitored.

When the ventilation system would stop, the under-pressure created by that system will vanish. As the WSF has no forced air supply, in the building no overpressure will arise. As the radioactive material is enclosed in the storage areas, no acute situation will develop.

Decontamination & Waste Treatment (DWT) facility

In the Decontamination and Waste Treatment Facility (DWT) processing and storage of radioactive waste and materials takes place in rooms partly below ground floor level. Most barrels are liquid tight. As there is no forced air supply, in the building no overpressure will arise when the ventilation system would stop.

GBD building

The GBD building is used for research and consultancy concerning handling of radioactive waste and radiation. In order to do this the building is equipped with laboratories for radiological analyses. NRG-RE also calibrates measurement systems and provides dosimetry. As there is no forced air supply, in the building no overpressure will arise when the ventilation system would stop.

Low Flux Reactor (LFR)

The Low Flux Reactor (LFR) is not operated anymore. In that state the ventilation system is in shutdown mode. This means that the ventilation shut-off valves are closed and as a result the system is isolated, so no (possibly present) radioactivity can escape. As there is no forced air supply for pressure build-up, there will be no overpressure in the building. The fuel elements are stored in the storage pits (pluggennest) below ground floor level and covered by plugs, and the experimental converter plate is locked in its usual shielded position on the irradiation trolley in front of the LFR, so no acute situation can develop.

STEK hall

The STEK hall is out of service. Currently radioactive waste awaiting transport to COVRA is stored in the STEK hall, enclosed in containers. When the ventilation system would stop, there will be no overpressure in the building, because there is no forced air supply for pressure build-up in the STEK hall. No acute situation can develop, because the radioactive waste is stored in closed containers.

6.3.3.2 Operational and organisational provisions

High Flux Reactor (HFR)

In addition to the design provisions mentioned in Section 6.3.3.1 there are no operational and organisational provisions to prevent overpressure in the containment. Note that even in case of severe accidents the reactor building pressure will stay below the pressure at which the water lock opens, see Section 6.3.3.1.

Hot Cell Laboratories – Research Laboratory (HCL-RL)

Strategies that can be followed at the HCL-RL for ensuring the confinement function are presented in Appendix A.17.1. The strategies for prevention of overpressure of the buildings are:

- Switching-off of the ventilation air supply fan manually in case the automatic switch-off action fails (and possibly closing the inlet and outlet, see appendix A.17.1, strategies 1 and 2).
- Recovery of cell ventilation in case of loss of the original power cable. A cable with plug is present in the technical room (location of the exhaust fans), see Appendix A.17.1, strategy 8. This cable can be connected to the cell exhaust fan motor in case of damage of the original power cable (as cause for loss of cell ventilation). The socket to be used is connected to the on-site emergency power grid.

Hot Cell Laboratories – Molybdenum Production Facility (HCL-MPF)

Strategies that can be followed at the HCL-MPF for ensuring the confinement function are presented in Appendix A.17.2. The strategy that can be used to prevent overpressure of the building is to switch-off the ventilation air supply fan manually in case the automatic switch-off action fails (and possibly closing the inlet and outlet, see Appendix A.17.2, strategies 1 and 2).

Jaap Goedkoop Laboratories (JGL)

Strategies that can be followed at the JGL for ensuring the confinement function are presented in Appendix A.18. The strategy that can be used to prevent overpressure of the building is to switch-off the ventilation air supply fan manually in case the automatic switch-off action fails (and possibly closing the inlet and outlet, see Appendix A.18).

Other facilities

These buildings have no forced air supply, so when the ventilation system would stop, no overpressure in these buildings will arise (see above Section 6.3.3.1).

6.3.4 Prevention of (re)criticality

Different from a power reactor, concerning the nuclear installations in the Research Location Petten (OLP) criticality plays a role at several distinctly different facilities in various ways. A summary of the re-assessment of criticality in the nuclear installations on the OLP is presented in Appendix A.19.

6.3.4.1 Design provisions

High Flux Reactor (HFR)

The HFR may only be operated if the core configuration complies with the requirements stated in Section 4 of the HFR VTS. In case of a power failure the control rods are designed to fall into the core automatically by gravity as well as by drag force of the primary coolant.

The storage vaults for fresh LEU fuel elements and fresh control rods are sub-critical by design in any operational condition. The same holds for the bulk storage vault for fresh HEU Molybdenum production targets, and for the collected uranium filters from the Molybdenum production process in MTR 2/33 containers in the HFR pool. For the racks in the HFR pool containing (partially) irradiated LEU fuel elements and control elements sub-criticality was shown under conservative assumptions.

Low Flux Reactor (LFR)

All fuel of the LFR, permanently shut down in December 2010, is stored in the storage pits (pluggennest), which configuration is sub-critical by design. Also the experimental converter plate – still in place at the reactor's irradiation trolley – which contains 360 g HEU, is sub-critical by design.

Hot Cell Laboratories (HCL)

For the 4 storage pipes in the HCL-RL used to store uranium filters from molybdenum production sub-criticality by design was shown under all conditions. For the racks in the HCL-RL pool containing some old fuel elements and at maximum two racks filled with uranium filters from molybdenum production sub-criticality was shown under conservative assumptions.

Waste Storage Facility (WSF)

Each of both zones containing fissile materials in the WSF – viz. the trenches and the storage pipes – is sub-critical by design.

Jaap Goedkoop Laboratories (JGL) and STEK hall

Both at the JGL and the STEK hall criticality is no issue.

General concerning criticality safety analyses

In all criticality safety analyses deformation or displacement has been excluded. It is recommended to assess such deviations in future criticality safety analyses.

6.3.4.2 Operational provisions

High Flux Reactor (HFR)

In case of failure of the reactor shutdown mechanism, an alternative shutdown system is available in the form of a cadmium plate that has to be placed next to the (west) wall of the reactor vessel. The present procedure for placement of this cadmium plate will be trained in the current situation, and it will be appraised whether the time needed to execute this procedure is acceptable.

With the cadmium plate in place, the reactor will become sub-critical and due to xenon poisoning will remain so for some 45 hours. This provides ample time to decide on execution of further actions to prevent re-criticality, dependent on the cause of malfunction. To this end it is recommended to design and build cadmium plugs for insertion in the HFR core as additional instrument to maintain sub-criticality.

Hot Cell Laboratories (HCL)

Sub-criticality in the HCL-MPF is guaranteed by a combination of physical constraints (distances, shielding) and procedurally constraints limiting the maximum uranium amount in a cell to the safe mass of 700 g ^{235}U equivalent.

Waste Storage Facility (WSF)

The fissile material in the WSF trenches is stored in cupboards and racks, and could get relocated under accidents conditions. Therefore, it is recommended that the fissile material in the WSF trenches be stored in such a way that its location under accident conditions is assured.

Jaap Goedkoop Laboratories (JGL) and STEK hall

Both at the JGL and the STEK hall criticality is no issue.

6.3.5 Prevention of basemat melt through

6.3.5.1 Potential design arrangements for retention of the corium in the pressure vessel

High Flux Reactor (HFR)

If the fuel in the reactor vessel is severely damaged the material is relocated to a lower part of the vessel. The HFR reactor vessel is encapsulated in the reactor pool. MELCOR analyses of severe accident progressions involving core melt have shown (1) that the heat produced in a relocated core is partly removed over the vessel wall towards the reactor pool, and (2) that the bottom of the reactor vessel will not fail. Most of the decay heat however will remain contained in the relocated core material, which therefore remains at en elevated temperature for a long period of time. The MELCOR analyses did show that the heat transfer to the reactor pool is sufficient to avoid melt-through of the bottom of the reactor vessel and, consequently, no core-concrete interactions will occur.

The MELCOR analyses did not encompass the situation that all primary and pool water are lost. In such situations still the Accident Management measures stipulated in Section 6.2.2 apply. Even if the bottom of the reactor vessel would fail there is still the sub-pile room between the bottom of the reactor vessel and the containment basemat, so no relocated core material will reach the containment basemat.

Other nuclear facilities

At the other nuclear facilities there is no fuel cooling function needed and therefore no melting and relocation of fuel material can occur. Consequently, prevention of basemat melt through of relocated fuel material does not apply.

6.3.5.2 Potential arrangements to cool the corium inside the containment after reactor pressure vessel rupture

High Flux Reactor (HFR)

See Section 6.3.5.1. Besides note that the Accident Management measures after occurrence of fuel damage are mentioned in Section 6.2.2.

Other nuclear facilities

See Section 6.3.5.1.

6.3.5.3 Cliff edge effects related to time delay between reactor shutdown and core meltdown

High Flux Reactor (HFR)

Chapter 6.1.5 describes the Accident Management measures to prevent core melt and the Accident Management measures after occurrence of a core melt. From these Accident Management measures it can be concluded that the HFR has strategies to mitigate severe accidents. One specific timeline cannot be given because all scenarios differ.

For the severe conditions considered in this report cliff edge effects for all nuclear facilities have been identified in other chapters of this report:

- cliff edge effects in case of earthquake are described in Chapter 2
- cliff edge effects in case of external flooding are described in Chapter 3
- cliff edge effects in case of loss of primary UHS and total loss of AC power (referred to as loss of primary UHS with SBO2) 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

For loss of electrical power supply a distinction is made between the following cases (see Chapter 5 of this report):

- Loss of offsite power (LOOP). Meaning that there is no supply from external grid(s).
- Loss of offsite power and emergency AC power supply (Station Black-Out-1, SBO1). Meaning that in addition to the loss of supply from external grid(s) also the normal emergency power system providing AC power is not available.
- Loss of all electrical power supply (Station Black-Out-2, SBO2). Meaning that in addition to SBO1 also the diverse permanently installed backup AC power sources are not available.

6.3.6.1 Design provisions

High Flux Reactor (HFR)

For ensuring the isolation of the reactor building it is important that the isolation valves to the environment are closed. The isolation is possible by closure of the ventilation valves (to the ventilation building), closure of the valves of the off-gas system (to the ventilation building) and/or closure of the stack valves. The valves are opened by compressed air.

In case of failure of electrical AC and DC power and simultaneous failure of the compressed air system the ventilation valves (to the ventilation building), the valves of the off-gas system (to the ventilation building) as well as the stack valves will close automatically, as these are spring closed. For closure of the ventilation valves, the valves of the off-gas system or the stack valves no compressed air is needed.

For protection against overpressure the reactor building of the HFR has a water lock which opens at an overpressure of 0.5 bar. This water lock is passive, needs no electric power for actuation, and opens therefore also in case of failure of electrical AC and DC power. Thus containment integrity is assured in such a situation.

Hot Cell Laboratories – Research Laboratory (HCL-RL)

In case of failure of electrical AC and DC power:

- The motor-operated isolation valves of the building and cell fans at the outlet (connection to the environment) will be closed, because the valves are fail-safe (spring closed). The valves can be operated also manually.
- The supply fan opening at the inlet of the ventilation system (connection to environment) can be closed manually by closing the sealing doors (kneveldeuren).

For closure of the valves no compressed air is needed.

- In case of LOOP, the systems of the HCL-RL equipped with emergency power are connected to emergency power bus bar A. The following main components of the HCL-RL are supplied with electrical energy by this bus bar:
- Manipulators
- Emergency lighting on the ground and first floors
- Pumping equipment and the fans of the ventilation system
- Emergency air compressor
- Radioactive release monitoring equipment.

In case of LOOP-SBO1 the consequences of failure of power supply for the HCL-RL is failure of the intake and exhaust fans due to which the under-pressure regime would be lost. A leakage might develop at the passage openings of the manipulators although this leakage is counteracted by the loss of the under-pressure regime in other parts of the building. In such a situation radioactivity might leak to other places in the building. In case of relatively high wind velocities radioactivity might even be sucked out of the building and be released, though the radiological consequences would still be relatively small.

In case of LOOP-SBO2 the batteries of the fire detection system have a stand-by time of 72 hours. The batteries of the alarm and the radiation monitoring system as well as the building's intercom system have a stand-by time of approximately 30 minutes. From that time on any radioactive releases will not be monitored and recorded anymore. The battery stand-by time of 1 hour of both the centralised and the decentralised emergency lighting system suffices for evacuation of the building in an orderly fashion.

Hot Cell Laboratories – Molybdenum Production Facility (HCL-MPF)

In case of failure of electrical AC and DC power:

- The motor-operated isolation valves of the air outlet, located before and after the building fans and cell fans (connected to the outside air) will be closed, because the valves are fail-safe (spring closed). The valves can also be operated manually.
- The motor-operated isolation valves of the inlet will be closed, because the valves are fail-safe (spring closed). The valves can also be operated manually.

For closure of the valves no compressed air is needed.

Note that in case of LOOP the emergency power facilities of the HCL-MPF are fed from bus bar B during a loss of electrical power. The following main components of the HCL-MPF are supplied with electrical energy by emergency bus bar B:

- Pumping equipment and intake and exhaust fans
- Emergency lighting
- Control panels of cells and pumps of cells 01/11 and 02/12
- Warning and monitoring systems and the central intercom
- Fire alarm system and CO2 fire fighting system
- HCL-MPF emergency air compressor.

The Control panels of cells and pumps of cells 01/11 and 02/12, the warning and monitoring systems and the central intercom as well as the fire alarm system and CO_2 fire fighting system are equipped with a no-break power supply to guarantee an uninterrupted power supply during the start-up of / switch-over to the emergency power system. The discharge time of the batteries of the fire alarm system is 72 hours. The other no-break batteries have a discharge time of around 30 minutes. The batteries of the decentralised emergency lighting system have a discharge time of at least 1 hour.

In case of LOOP-SBO1 the consequences of failure of power supply for the HCL-MPF are a failure of the intake and exhaust fans due to which the under-pressure regime would be lost. The cells 01/11 are isolated, but a leakage might develop at the passage openings of the manipulators although this leakage is counteracted by the loss of the under-pressure regime in other parts of the building. In such a situation radioactivity might leak to other places in the building. In case of relatively high wind velocities radioactivity might even be sucked out of the building and be released, though the radiological consequences would still be relatively small

Jaap Goedkoop Laboratories (JGL)

In case of failure of electrical AC and DC power:

- The motor-operated isolation valves of the outlet of the ventilation system will be closed because the valves are fail-safe (spring closed). The valves can also be operated manually.
- The motor-operated isolation valves of the inlet of the ventilation system will be closed, because the valves are fail-safe (spring closed). The valves can also be operated manually.

For closure of the valves no compressed air is needed.

In case of LOOP the following systems are supplied with electrical energy by emergency bus bar B:

- Air conditioning systems for building and off-gas systems for glove-boxes
- Wall sockets in laboratories for experiments with radioactive materials
- Warning and monitoring systems and central intercom
- Fire warning system
- Emergency lighting system.

In case of LOOP-SBO1 failure of electrical power supply does not imply the dispersion of radioactivity.

The subsequent failure of electronics of active experiments and of CO gas monitoring does not pose an acute problem, as the radioactivity remains confined.

In case of LOOP-SBO2 the batteries of the fire detection system have a stand-by time of 72 hours. The battery stand-by time of 1 hour of the emergency lighting system is sufficient for evacuation of the building in an orderly fashion.

Waste Storage Facility (WSF)

Inside the Waste Storage Facility (WSF) radioactive waste and nuclear fuel material is stored in areas below ground floor level. Most of the radioactive material is enclosed in drums as well. The ventilation system creates an under-pressure in the storage areas and the filtering system removes any airborne radioactivity.

In case of LOOP the ventilation system, the filtering system and the radiation monitors of the WSF are connected to emergency bus bar A. In case of LOOP-SBO with complete loss of electrical power, the ventilation system would stop and the under-pressure created by that system would vanish. Also the filtering system would stops functioning. The building isolation valves of the ventilation system would close. The WSF has an UPS for the radiation monitoring system which feeds the system for one hour. The radioactive material is enclosed in drums. Therefore without the above systems properly working still no acute situation will develop. Only if the drums are damaged, radioactivity might be set free in the building unnoticed by the monitoring system and by-passing the filter system be released.

Decontamination & Waste Treatment (DWT) facility

In case of LOOP the ventilation system, the filtering system, the monitoring system and the emergency lighting in the DWT is connected to emergency bus bar A. As the drums are packed in leak tight vessels and stored below a concrete cover, no radioactivity will be set free in case of failure of electrical AC and DC power. Only if the vessels are damaged, radioactivity might be set free unnoticed by the monitoring system and by-passing the filter system be released.

GBD building

In case of LOOP the following systems of the GBD building are supplied with electrical energy by emergency bus bar B:

- Air conditioning systems
- No-break batteries for information systems
- LAN data
- CRM engineering workstation

In case of LOOP-SBO the no-break sets of the above information systems have a limited stand-by time of approximately 15 minutes which only causes a possible loss of data. The ventilation shut-off valves close and in the fume-hoods the under-pressure will be lost, but no radioactivity will be set free when the emergency power supply would fail.

Low Flux Reactor (LFR)

The Low Flux Reactor (LFR) ceased operation in December 2010. The normal electrical energy for LFR hall monitoring is supplied by substation ECN 4. The LFR hall is not equipped with emergency power. In case of LOOP the monitoring system will become unavailable. According to procedure, in case the LFR is not operated the ventilation system is in the shutdown mode. This means that the ventilation shut-off valves are closed due to which the system is isolated and no (possibly present) radioactivity would escape. LOOP will not affect the system isolation.

STEK hall

For the STEK hall the normal electrical energy for monitoring is supplied by substation ECN 4. It is no problem when the delivery of electrical energy fails, since the radioactive waste is stored in closed shielded containers. The ventilation shut-off valves of the STEK hall close at LOOP and no (possibly present) radioactivity would escape.

6.3.6.2 Operational provisions

High Flux Reactor (HFR)

The strategies which can be performed at the HFR for protecting the containment function are presented in appendix A.16.5.

In addition to the measures mentioned in Section 6.3.6.1 the following strategies can be performed in case of loss of electrical AC and DC power and loss of compressed air:

- Reduce pressure in the reactor building and/or mitigate fission product release by venting via absolute filters and active carbon filters in the ventilation building (LBG).
- Inundation of the Swan Lake (Zwanenmeer) when the U-bend piping in the Swan Lake is damaged.

It is recommended to describe such actions in a Containment Function Restoration Procedure.

Hot Cell Laboratories – Research Laboratory (HCL-RL)

The strategies which can be performed at the HCL-RL for ensuring the containment function are presented in Appendix A.17.1.

In addition to the measures mentioned in Section 6.3.6.1 the following strategies can be performed in case of loss of electrical AC and DC power and loss of compressed air:

- The off-gas isolation valves can be closed manually.
- For the cell and building ventilation fans, the valve at the pressure side of the common outlet of the two fans can also be closed, especially in case closure of the motor operated isolation valves at the building outlet fails. This valve can be closed manually as well.

Hot Cell Laboratories – Molybdenum Production Facility (HCL-MPF)

The strategies which can be performed at the HCL-RL for ensuring the containment function are presented in Appendix A.17.2. In addition to the measures mentioned in Section 6.3.6.1 there are no operational provisions which can be performed in case of loss of electrical AC and DC power.

Jaap Goedkoop Laboratories (JGL)

The strategies which can be performed at the JGL for ensuring the containment function are presented in appendix A.18. In addition to the measures mentioned in Section 6.3.6.1 there are no operational provisions which can be performed in case of loss of electrical AC and DC power.

Other facilities

In addition to the measures mentioned in Section 6.3.6.1 there are no operational provisions which can be performed for ensuring the containment function in case of loss of electrical AC and DC power.

In case of a long-lasting LOOP, sooner or later emergency power will be lost in most of the nuclear facilities on the OLP. Though these facilities will be in safe shutdown isolated state, under-pressure will be lost and no radiation readings will be available anymore. It is recommended to develop a function restoration procedure for maintaining the confinement function of the HCL-RL, the HCL-MPF, JGL, WSF, DWT and STEK hall.

6.3.7 Measuring and control instrumentation needed for protecting containment integrity

High Flux Reactor (HFR)

The information on the HFR instrumentation in this section is derived from the HFR instrument data sheets, the HFR instrument wiring diagrams and the HFR alarm listings.

The containment instrumentation that is important in case of an accident is listed in Appendix A.20.

The necessary measurements for the under-pressure situation of the containment under accident conditions are fully covered by PIXTA-V9. For gauge pressure the PIA-V20 is present but not yet readable because it is a parameter of the remote monitoring system (RMS) that is expected to become operational from mid-2012 on. The same applies for the containment activity monitor RIA-GM4.

There are two containment temperature instruments present but they have no reading in the control room. The reading has to be done in the ventilation building (LBG). In normal conditions the operators use the ventilation system's in- and outlet temperature as indication. However, in accident conditions the ventilation system is in isolation mode, so these readings give no indication on containment temperature.

The activity monitor RIRXSA-GM1 monitors the containment and is located on the ground floor of the ventilation building (Luchtbehandelingsgebouw, LBG), so it is shielded from DBA or severe accident conditions.

In case radioactivity within the containment exceeds a value of $3.7.10^5$ $Bq.m^{-3}$, the activity monitor is switched-over to sampling from the delay line in the LBG, whereas the ventilation system is switched-over into isolation mode. From that moment on an indication of the containment activity can only be derived from the off-gas activity monitor RIRXSA-GM3 that samples 10% of the original ventilation flow. However, from mid-2012 on RIA-GM4 will be available too for monitoring of the containment activity in case of accident conditions. The activity monitors are not seismic qualified. It is recommended to qualify the monitors for earthquake resistance.

For monitoring emergency radiation levels, inside the reactor building four high range radiation sensors (watertight) are installed. Three are located at the upper part of the containment and one is located near the personnel airlock. These emergency monitors are connected to the central radiological monitoring (CRM) system and its readings can be obtained over the NRG network as well as over the internet. The readout of the emergency monitoring system in the control room is obtained over the NRG network that is also backed-up by emergency power. If pre-set values are exceeded an alarm signal is provided in the control room. The system is powered by the normal electricity grid and is backed-up by emergency power; the monitors have their own battery backup. Hence, during accidents with loss of normal electricity grid, in the control room still information of the emergency monitoring system on activity in the confinement area or release to the environment is available.

In case of severe accidents the containment air can reach high temperatures (max. approximately 90 °C). For these conditions the radiation sensors are not formally designed. However, tests have shown that they do function properly at these temperatures.

In case due to high radiation, flooding, fire, or other reasons the HFR control room and/or the electronic systems in the basement of the reactor outbuilding are unavailable, the relevant operating conditions of the reactor can still be monitored in order to ensure the safe shutdown conditions. This can be done by the Remote Monitoring System (RMS) (operational from mid-2012), located in a building outside the HFR. That building is protected against increased radiation levels as well as an internal fire at the HFR. As far as reasonably achievable, its power supply, electronics, wiring and cabling are independent of those in the reactor outbuilding. The RMS is a monitoring system only, it provides no means for actions.

In the RMS the following containment parameters are monitored:

- Radioactivity in the reactor hall RIA-GM04 (1.2. 103 - 5. 108 Bq.m-3)
- Radiation in the reactor hall RIA-104 (10-9 - 10 Sv. h-1)
- Pressure in the reactor hall PIA-V11 (-80 mm tot 6 m H2O).

Hot Cell Laboratories – Research Laboratory (HCL-RL)

The instrumentation of the HCL-RL building important in case of an accident is listed in Appendix A.21. The under-pressure situation of the HCL can be monitored at the ventilation information panel, nick-named Rikkel Nikkel[46] cabinet. That panel is located in the central aisle of the HCL-RL facility. Since it dates from the early days of the HCL and is not earthquake resistant, a project is underway to revamp the ventilation system in order to meet present day requirements.

Under-pressure in the central hall (OT) and the central cell exhaust can be read from two Magnehelic differential pressure indicators. For their operation they are not dependent on any energy source. Their readings are presented to the building management system (Gebouwbeheersysteem, GBS), which initiates alerts for the off-site paging system if pre-set limits are exceeded. The cabling rolled out for this monitoring system appears to be vulnerable for lightning strikes, hence improvement is considered. Supplemental to both instruments, under-pressure can be read-out on local Rixen differential pressure indicators that are earthquake resistant and also require no energy source.

The main ventilation and off-gas flow to the HCL-MPF stack is monitored by radioactivity detectors. Their data are monitored on a panel adjacent to the ventilation panel in the central aisle of the HCL-RL facility. The detectors as well as their lead shielding casings are not earthquake resistant. An air dust contamination detector monitors the building's ventilation system for possible radioactivity. Alarms from the monitors are presented to the ECN-GBS. The readings of the detectors are presented to the NRG's central radiological measurement system (CRM) that can be accessed from any place on the OLP via NRG intranet by authorised staff.

Hot Cell Laboratories – Molybdenum Production Facility (HCL-MPF)

The HCL-MPF building instrumentation important in case of an accident is listed in Appendix A.21. Apart from the common stack, the ventilation system of the HCL-MPF is completely independent of that of the

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[46] "Rickel Nickle, the adventures of a robot" was a Dutch children's television series broadcasted in the season 1961/1962. Possibly, the Rikkel Nikkel panel is nick-named after a gadget in that series.

HCL-RL. Under-pressure in the transport hall is measured by a local Rixen differential pressure indicator, earthquake resistant and not requiring any energy source for local read-out. It is equipped with an electronic module that presents its signal to a panel in the central aisle of the HCL-RL facility as well as to a second panel in the entrance hall of the HCL-MPF (not earthquake resistant). Where the HCL ventilation system only sends information to the GBS, the control system of the HCL-MPF ventilation normally acts as a stand-alone system. As for maintenance purposes this system can be remotely accessed from the central GBS computer in ECN building 31 at the OLP, this system might be vulnerable for tampering over the NRG data network.

The main ventilation flow to the HCL-MPF stack is monitored by radioactivity detectors for noble gases and iodine (lead shielding casings are not earthquake qualified). Read-out is on a monitor by a panel adjacent to the ventilation panel in the central aisle of the HCL-RL facility (not earthquake qualified) and a slave panel in the entrance hall of the HCL-MPF. Alarms from the monitors are presented to the GBS and its readings are presented to the CRM. A project is underway to revamp the HCL-MPF ventilation system.

Jaap Goedkoop Laboratories (JGL)

The JGL building instrumentation important in case of an accident is listed in Appendix A.21. The design of the JGL ventilation control system is identical to that of the HCL-MPF including the possibility of remote access via the NRG Intranet. Under-pressure in the central building part and each of the laboratories is monitored by differential pressure indicators with an electronic module that presents its signal to a central panel in the JGL entrance hall (not earthquake qualified) and the GBS.

The main ventilation flow to the JGL stack is monitored by a radioactivity monitor for noble gases. Its readings are presented to the panel in the entrance hall and to the CRM. It is recommended to evaluate in more detail the earthquake qualification of the instruments for building pressure and radioactivity.

Other facilities

In the buildings of these facilities pressure measurements and radiation monitors are installed. It is recommended to evaluate in more detail the earthquake qualification of these instruments.

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

Since only the HFR requires core and spent fuel cooling to prevent core melt or fuel damage, simultaneous core melt/fuel damage accidents at different nuclear facilities cannot occur on the Research Location Petten (OLP). Due to the distinct differences between the nuclear installations on the OLP, simultaneous (re-)criticality accidents (see Section 6.3.4) at different installations due to common cause are very unlikely.

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

The accident management measures for protecting containment integrity are presented in Section 6.3.3. Most accident management measures are related to closure of the valves of the building ventilation system to the environment and switch-off supply fans manually. For the HFR only a part of the strategies are written down in procedures. For the other facilities no procedures exists to provide the necessary guidance to protect the building's containment integrity in case of an accident.

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

The following measures can be envisaged to enhance accident management capabilities:

- The installation of a Remote Monitoring System for the HFR will be completed by mid-2012 and will provide the capability to monitor the HFR under extreme adverse conditions.
- Provisions will be made to add neutron absorbers to improve HFR reactivity control in emergency situations.
- The storage facility of fissile material in the WSF trenches should be strengthened in such a way that deformation is excluded under all accident conditions.

The following study can be envisaged to enhance accident management capabilities:

- The currently available criticality safety analyses for the various storage locations for fissile material will be extended in the light of possible deformation of the storage facilities due to extreme events.

The following procedures can be envisaged to enhance accident management capabilities:

- A Function Restoration Procedure for maintaining the containment function of the HFR will be developed (as supplement to the existing procedures).
- Develop function restoration procedures for maintaining the confinement function of the HCL-RL, the HCL-MPF, JGL, WSF, DWT and STEK buildings.
- The placement of the Cd plate next to the HFR vessel will be trained in the current situation.

6.4 Accident management measures to restrict the radioactive releases

The ENSREG prescribed table of contents was developed for a nuclear power plant. However, the nuclear installations on the Petten Research Park (OLP) comprise of research reactors, radiological laboratories, and waste storage and treatment facilities. Consequently, it was necessary to deviate from this specification to broaden its applicability. The deviations are specified in Table 6.4 below.

ENSREG Guideline		Present Report	
6.4	Accident management measures to restrict the radioactive releases	6.4	Accident management measures to restrict the radioactive releases
6.4.1	Radioactive releases after loss of containment integrity	6.4.1	Radioactive releases after loss of confinement
6.4.1.1	Design provisions	6.4.1.1	Design provisions
6.4.1.2	Operational provisions	6.4.1.2	Operational provisions
6.4.2	Accident management after uncovering of the top of fuel in the fuel pool	6.4.2	Accident management after loss of primary confinement
6.4.2.1	Hydrogen management	6.4.2.1	Hydrogen management
6.4.2.2	Providing adequate shielding against radiation	6.4.2.2	Providing adequate shielding against radiation
6.4.2.3	Restricting releases after severe damage of spent fuel in the fuel storage pools	6.4.2.3	Restricting releases
6.4.2.4	Instrumentation needed to monitor the spent fuel state and to manage the accident	6.4.2.4	Instrumentation needed to monitor in support of accident management
6.4.2.5	Availability and habitability of the control room	6.4.2.5	Availability and habitability of the control room
6.4.3	Conclusion on the adequacy of measures to restrict the radioactive releases	6.4.3	Conclusion on the adequacy of measures to restrict the radioactive releases
6.4.4	Measures which can be envisaged to enhance capability to restrict radioactive releases	6.4.4	Measures which can be envisaged to enhance capability to restrict radioactive releases

Table 6.4: Deviations from the ENSREG Table of Contents

6.4.1 Radioactive releases after loss of confinement

6.4.1.1 Design provisions

To mitigate a radioactive release after loss of the confinement function the following provisions are available:

High Flux Reactor (HFR)

- Use of the off-gas recirculation filter system (absolute filters and active carbon filters) in order to mitigate the release
- Use of the filters of the ventilation system

Hot Cell Laboratories – Research Laboratory (HCL-RL)

- Use of the pre-filters and absolute filters (first floor the building exhaust air).
- Use of the active carbon filters

Hot Cell Laboratories – Molybdenum Production Facility (HCL-RL)

- Use of the pre-filters and absolute filters
- Use of the active carbon filters

Other facilities

- Use of the absolute filter system of the building

6.4.1.2 Operational provisions

The following Accident Management measures are applicable:

High Flux Reactor (HFR)

(see also appendix A.16.5)

- Reduce pressure in the reactor building by opening the RB valves to the ventilation building (LBG).

Hot Cell Laboratories – Research Laboratory (HCL-RL)

- Recovery of cell ventilation in case of loss of the original power cable. A cable with plug is present in the technical room (location of the exhaust fans), see Appendix A.17.1, strategy 8. This cable can be connected to the cell exhaust fan motor in case of damage of the original power cable (as cause for loss of cell ventilation). The socket to be used is connected to the on-site emergency power grid.

- Spraying the outside of the building by the fire brigade to scrub possible fission products.

Other facilities

- Spraying the outside of the building by the fire brigade to scrub possible fission products.

6.4.2 Accident management after loss of primary confinement

This section is renamed to cover also accidents in further nuclear installations than reactors and fuel storage pools. According to the ENSREG list of contents, this section should treat *accident management after uncovering of the top of fuel in the fuel pool*, for other installations that state would loosely be comparable to loss of primary confinement.

Of the nuclear installations on the Petten Research Park (OLP) only the HFR has a reactor and spent fuel storage pool in operation. The LFR ceased operation in December 2010 and its fuel is in dry storage in storage pits (pluggennest), awaiting transport to COVRA. Of the other installations on the OLP, only the HCL-RL features a storage pool, though this pool is not comparable to a fuel storage pool of reactors: the HCL-RL storage pool does contain irradiated fuel and targets, but these do not need any cooling:

- Prior to its transport from the HFR to the HCL, the fuel to be stored in the HCL-RL pool has been cooled typically for at least half a year, and the amounts of fuel are small
- The targets contain only small amounts of uranium and normally have been irradiated for only short periods of time so their decay heat generation is small.

Consequently, loss of water of the HCL-RL storage pool will only lead to increased radiation levels, but will not lead to a release of radioactivity in the building.

6.4.2.1 Hydrogen management

Not applicable for any of the installations on the OLP (for HFR: see Section 6.3.2).

6.4.2.2 Providing adequate shielding against radiation

High Flux Reactor (HFR)

The Accident Management measures that are applicable for the HFR to refill the reactor pool, pool 1 and pool 2 are presented in Appendix A.16.2. Note that after loss of cooling of the pools the fuel assemblies will be remain covered with water for approximately 700 hours, see Chapter 5.

A decrease of the water level in the reactor pool will cause an increase of the dose rate both inside and outside of the reactor building. Direct radiation will only have consequences for operators near the edge of the pool. Due to the thick concrete walls of the reactor pool the exposure at lower parts of the reactor hall and the control room will only be caused by indirect scattered radiation even if the core is completely uncovered by pool water. Since the top side opening of the reactor pool is relatively narrow, primary scattering of radiation will be mainly at the reactor hall ceiling. It is estimated that even when the core is covered by only 1 m of water the dose rate at floor level in the reactor building will not exceed 10 µSv/hr.

Another source of radiation inside the reactor building is caused by airborne radioactivity, for instance after failure of an experiment in the core. When electrical power is available the airborne radioactivity will be removed by the ventilation system, operating either in normal mode or in recirculation mode. When the ventilation system cannot be used, the proper action to be taken depends on the need for human presence in the reactor building. This is discussed in more detail in Section 6.4.2.5 below (Availability and habitability of the control room).

Hot Cell Laboratories (HCL)

Due to an accident normal radiation shielding systems may be (partly) lost, such as:

- windows, walls or roof of the hot cells
- the water and/or concrete cover of the HCL-RL storage pool
- containers that provide shielding

For the HCL-RL no calculations of radiation levels were available for events resulting in loss of radiation shielding. As loss of all water from the HCL-RL storage pool would be an extreme situation in this respect, for the present report some preliminary calculations were performed to assess that situation. In those calculations presence of the maximum allowable amount of radioactive material in the HCL-RL pool was assumed. The dose rate directly above the pool was estimated at 25 Sv/hr, due to lack of shielding water. Directly around the pool dose rates up to 500 mSv/hr were calculated, mainly caused by reflection of radiation from the ceiling above the pool. At 4 m distance in the adjacent room, i.e. behind a 60 cm thick concrete wall, dose rates of ca. 0.25 mSv/hr were calculated. These preliminary results suggest that even in case of loss of all HCL-RL storage pool water it would still be possible to carry out emergency actions, although these have to be well-prepared. In scenarios with far more additional structural damage – such as might be caused by an airplane crash – all actions near and inside the HCL-RL would probably require additional shielding provisions.

It is recommended to assess the potential radiological impact on emergency workers in case of lowered water levels in the HCL-RL storage pool.

For the HCL-MPF none of the scenarios studied for the Complementary Safety Margin Assessment lead to failure of these shielding provisions, except for an airplane crash. However, if one of the cells – or floor – would be severely damaged, dose rates of over 10 Sv/hr may occur as preliminary calculations did show. In such circumstances, emergency actions in and near the HCL-MPF can only be carried out if additional shielding is provided (e.g. using transport containers).

Jaap Goedkoop Laboratory (JGL)

Due to the low activity content, severe accidents at the JGL have no relevant consequences, neither on-site nor off-site.

Decontamination & Waste Treatment (DWT) facility

In general, the radioactive waste treated at the DWT is low radioactive waste, not requiring additional shielding. Only the containers with resins of the HFR cooling water treatment system have a relatively larger exterior dose rate. Doses to staff are easily limited by keeping distance to and minimizing presence near these containers.

In case of airborne activity in case of a fire, fire fighters and rescue teams can use respiratory protection to avoid (low) doses due to inhalation of airborne radioactive material.

Waste Storage Facility (WSF)

In case of a severe accident at the WSF causing both severe damage to the shielding trench covers and plugs of the pipe storage area, dose rates to rescue workers or firemen may easily rise to 75 mSv/hr (at 15 m from the length wall, or double at 10 m). Doses due to inhalation of airborne radioactive material should be avoided using adequate respiratory protection. To support possibilities for fire-fighting and recovery actions it is recommended to assess the possibilities to prepare and conduct countermeasures in case of severe accidents involving radiation shielding, covering of debris, etc.

Low Flux Reactor (LFR)

In case of a severe accident at the LFR involving a fierce fire, dose rates to rescue workers and firemen would mainly be caused by exposure to the unshielded calibration sources (1 mSv/hr at 5 m distance, 6 mSv/hr at 2 m distance). Doses due to inhalation of airborne radioactive material will be relatively small but would easily be avoided by using adequate respiratory protection.

STEK hall

In case of a severe accident at the STEK hall involving damaging of a few type-B transport containers causing a spill of high-level liquid waste from molybdenum production, fire-fighters might receive a dose rate of some 5 mSv/hr (at 5 m from the spill, 15 mSv/hr at 2 m).

6.4.2.3 Restricting releases

Restricting releases

This section is renamed to cover also accidents in further nuclear installations than reactors and fuel storage pools. According to the ENSREG list of contents, this section should treat *Restricting releases after severe damage of spent fuel in the fuel storage pool*, for other installations this would mean all actions to restrict releases to the environment in general.

High Flux Reactor (HFR)

The spent fuel pools of the HFR are all inside the reactor building. The Accident Management measures applicable for restricting releases after severe damage of spent fuel in the fuel storage pools are presented in Appendix A.16.5.

Hot Cell Laboratories (HCL)

In virtually all circumstances releases to the environment can be prevented by the normal isolation measures, see Appendix A.17. That appendix also presents supplemental Accident Management measures to be applied as the procedural measures might fail.

Jaap Goedkoop Laboratory (JGL)

The largest releases are related to fires. As the amount of flammable material is small, internal fires pose no problem extinguishing. Any release will be filtered (see Section 6.4.1.1). No additional measures have been identified to restrict releases.

Decontamination & Waste Treatment (DWT) facility

For DWT focus of the measures to restrict radioactive releases is on fire fighting (including measures such as closure of ventilation intakes). In case of failure of tanks, containers and basins, measures such as pumping of contaminated water from failed basins and tanks or from floors flooded by radioactive contents of the failed containers is envisaged.

Waste Storage Facility (WSF), Low Flux Reactor (LFR), and STEK hall

Concerning restricting of radioactive releases, presently only repression plans to fight fires at the WSF by the corporate fire brigade exist.

6.4.2.4 Instrumentation needed to monitor in support of accident management

This section is renamed to cover also accidents in further nuclear installations than reactors and fuel storage pools. According to the ENSREG list of contents, this section should treat *Instrumentation needed to monitor the spent fuel state and to manage the accident*, for other installations this would mean all instrumentation needed to support accident management to restrict releases.

High Flux Reactor (HFR)

The pool instrumentation that is important in case of an accident is listed in Appendix A.20.2. Two of the reactor pool level monitors LITSA-B12/B15 are not qualified for severe accidents. They are however able to operate up to a pool temperature of 80 ºC. Furthermore, the reactor pool level monitor LITSA-B3 is qualified for severe accidents.

Also some of the level monitors of pool 1 and pool 2 are not formally qualified for severe accident conditions (LITRSA-B13 and LITRSA-B14). However, based on engineering judgment it is expected that these monotors will survive up to a containment temperature of approximately 70 ºC. Furthermore, the other reactor pool level monitors are certainly qualified for severe accident conditions.

The radiation monitors around the pool will survive DBA environment conditions but are not formally qualified for severe accident conditions. This is not a real problem because in that case the containment radiation monitors are available as secondary source. Furthermore, the actual situation in and around the pools can be observed by CCD cameras that can be monitored and controlled from the control room.

Other facilities

Inside these buildings fire detectors, under-pressure detectors (except STEK hall) and gamma or air dust contamination detectors are installed, often connected to the central monitoring system (CRM). Readings of these detectors might be used to monitor accident development. Readings of the exhaust ventilation monitors – located after the ventilation filters – might provide additional release information.

6.4.2.5 Availability and habitability of the control room

On the OLP, only the HFR is equipped with a control room. In case of the HCL there is no control room. However, tools and devices used for operational control are present in the HCL, too. Furthermore, for safety reasons at least the first step of an isotope production cycle in the HCL-MPF must be completed once it has started.

High Flux Reactor (HFR)

A scenario assuming failure of an experimental apparatus or materials in the HFR core has been assessed for its exposure consequences in the HFR control room. Calculations revealed that in case of such an accident operators in the control room might receive a dose between 3 mSv at the operating console to 26 mSv directly at the window facing the reactor in 30 minutes time. In such conditions prolonged presence in the control room would require clear necessity. As monitoring the situation by means of the planned Remote Monitor System would provide a good alternative, completion of that system is desirable.

In case of a severe accident involving core damage and/or core melt the reactor building would become filled with radioactive noble gasses and other volatile radionuclides. In such conditions dose rates in the control room would initially exceed 1 Sv/hr, dropping to several tens of mSv/hr in 4 days' time. Though the automatic recirculation of the off-gas system would reduce the airborne aerosol activity, the noble gas activity inside the containment of the reactor building would be unaffected. Only if the containment would have been vented, human presence in the control room would be justifiable.

Hot Cell Laboratories (HCL)

The HCL has no control room. An emergency panel to (de)activate alarms as well as to control the ventilation system and the fire stops is located at the reception of the HCL building. The Rikkel Nikkel[46] panel, located in the northern aisle of the U-shaped corridor surrounding the HCL-RL hot cell hall, provides the same functions. Presence in the HCL-RL in accident conditions is not justified. The same holds for presence in the HCL-MPF, although in that case safety issues concerning discontinuing a started cycle must be weighed against prolonged exposure.

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

High Flux Reactor (HFR)

For the HFR only part of the strategies is written down in procedures. The actions mentioned in existing procedures and the alternative actions not yet described in procedures are adequate measures to restrict the radioactive releases from the containment.

Alternative ways to cool the spent fuel pools and possible supplemental measures to enhance the robustness of the spent fuel pool cooling can be found in Chapter 6.2.

Hot Cell Laboratories (HCL)

For the HCL focus of measures to restrict the radioactive releases is on fire fighting and maintaining the under-pressure regime. These measures are efficient to restrict radioactive releases for the large majority of potential accidents involving radioactive releases. Actual response plans focus on relatively small accidents. For large fires, the accident will be scaled-up and the regional emergency response organisation will be activated.

Filtering of the exhaust gases is also effective to restrict releases. However, for severe accidents the ventilation system might not be available, e.g. due to unavailability of power or damage to the system. For severe accidents involving exposure of radioactive materials to a large fire and failure of the ventilation system, the release is mitigated by extinguishing the fire. If the corporate fire brigade is not able to fight the fire successfully, the Safety Region Noord-Holland Noord has sufficient means at its disposal to fight the fire, and stop the release eventually.

Jaap Goedkoop Laboratory (JGL)

Maximum releases pose no threat to the environment, the present measures are adequate.

Decontamination & Waste Treatment (DWT) facility

For DWT focus of the measures to restrict the radioactive releases is on fire fighting (including measures such as closure of ventilation intakes). In case of failure of tanks, containers and basins, measures such as pumping of contaminated water from failed basins and tanks or from floors flooded by radioactive contents of the failed containers is envisaged.

These are efficient measures to restrict radioactive releases for the large majority of potential accidents involving radioactive releases. Nevertheless, even if such emergency response actions would not be successful, dose consequences would be limited.

Waste Storage Facility (WSF), Low Flux Reactor (LFR), STEK hall

The following measures to restrict radioactive releases are implemented:

1. confinement of accidental released airborne radioactive materials to the building and an active ventilation system
2. exhaust ventilation filter unit removing airborne radioactivity
3. in case the release is caused by a fire, various fire fighting options are available

General

During accidents, water may become radioactive contaminated. Therefore contaminated water should be collected at the nuclear facilities. Assuming operability of the water treatment building of the DWT facility, that provision could be used to collect and process radioactive contaminated water resulting from mitigating actions and to release (controlled) the purified waste water. The availability of the water treatment facility of DWT should be ensured during and in the aftermath of a severe accident at the OLP, and the availability of sufficient storage capacity for water resulting from accident mitigation should be assessed. It might be necessary to have additional temporary storage capacity available at the facilities to collect radioactive contaminated water in case the underground pipes are blocked or damaged.

6.4.4 Measures which can be envisioned to enhance capability to restrict radioactive releases

The following studies can be envisaged to enhance capabilities to restrict radioactive releases:

- The potential radiological impact on emergency workers due to lowered water levels in the HCL storage pool will be analysed;
- Evaluation of the possibilities to prepare and conduct countermeasures like use of radiation shielding, covering of debris, collect, store, and process contaminated water, decontamination of equipment and persons in case of severe accidents.

7 Other extreme hazards

7.1 Selected events

The following extreme events were identified for this analysis:

- explosion and fire related hazards:
    - internal explosion;
    - external explosion, under which a sinking ship with explosion and a resulting tidal wave;
    - internal fire;
    - external fire;
    - airplane crash;
    - toxic gases.
- electrical related issues:
    - large grid disturbance;
    - millennium-bug kind of failure of systems, renamed as failure of systems by introducing computer malware.
- water related issues:
    - internal flooding;
    - blockage of cooling channel inlet.

The following information is elaborated about each of these hazards:

- General description of the event,
- Description of how the event could lead to consequences for the safety systems,
- Elaboration on these consequences.

In the following sections, the last two topics have been taken together in a single subparagraph.

7.2 Internal explosion

7.2.1 General description of the event

Internal explosions are defined as being those explosions that originate from plant systems and plant storages. To protect safety relevant systems against the impact of internal explosions, the following measures are included in the design to prevent occurrence of internal explosion and to reduce their impact:

- Apply as far as possible inflammable gases and liquids instead of more obvious but combustible ones;
- Reduce number and volume of explosive materials;
- Limit the releases of explosive materials in case of disturbances;
- Subject the storage of explosive materials to special precautions;
- Monitor risk areas combined with automatic safety measures;
- Ventilate risk areas;

7.2.1.1 High Flux Reactor

Most gas cylinders stored in the reactor containment building contain mixtures of inert or noble gases and form no threat for the integrity of the installation. The following gas cylinders remain:

1. One 10 litres cylinder $CH_4/CO_2/N_2$ (64.4/32.4/3.2) on the first platform;
2. Three 10 litres 97.6 % CO stored in the basement;
3. One mobile 50 litres cylinder N_2/H_2 (95/5) backing gas TIG welding;
4. Two methane/argon cylinders for the portal monitor in the “SC” room.

A special instruction from the Reactor Safety Commission has been made for the CO cylinders, and the number allowed and the mutual distance they are to be placed.

In all three cases release and complete dispersion of the gas in the reactor containment building or in the cellar of the reactor containment building does not lead to an explosive mixture. In all three cases the volumetric percentage is well below the lower explosion limit.

Instantaneous release however, i.e. complete failure of the bottle, and direct ignition may lead to an explosion inside the reactor containment building.

Frequencies and maximum pressure spikes have been calculated (REMAIN2 project, 2010). The calculated pressure spikes generated by the explosions are significantly higher than the static design pressure of the containment (50 kPa) and containment failure might occur. This failure consists however only from an opening through the “water slot”; the dome itself can withstand higher pressures. However direct damage to the dome cannot be excluded entirely for this scenario.

Recommendation: Pressurized gas cylinders as mentioned should be removed from the reactor hall as much as possible. Welding equipment should only be moved into the building for the period that it is used, and removed afterwards. Carbon monoxide should only be used in very small quantities in the building. Larger stocks should be stored outside the reactor building.

7.2.1.2 Hot Cell Laboratory

The only natural gas supply of the HCL is the one supplying the radiation monitors; here the natural gas is used as a purge gas. The flow capacity of this line is so small that it cannot cause a gas cloud explosion of a significant size. The total natural gas consumption of the HCL is about 0,6 m^3/hour.

No other explosive materials are reported.

7.2.1.3 Other NRG facilities

In the other facilities under consideration for the safety margin analysis:

1. Low Flux Reactor
2. Decommissioning and Waste Treatment Facility
3. Jaap Goedkoop Laboratory
4. Waste Storage Facility
5. STEK hall

no explosive materials are stored or handled.

7.2.2 Potential consequences for the plant safety systems

7.2.2.1 High Flux Reactor

Beside the loss of containment, there might also be an impact on other safety functions like reactivity control and decay heat removal, for example as a result of the shock wave and/or fire, also damaging the cabling of the reactor protection system. Additional investigation is required to assess the potential contribution to the core damage frequency as a result of explosions.

Recommendation: The effect of internal explosions on the safety functions of the HFR besides loss of containment will be investigated.

7.2.2.2 Other NRG facilities

The other NRG facilities contain no explosive materials of significance, so this event does not pose any threat to their safety systems.

7.3 External explosion

7.3.1 General description of the event

Explosion pressure waves may generally result from accidents in nearby facilities or means of transportation. The damage may be due to pressure wave impact or impact of missiles.

Outside the OLP, there is no industrial activity, and therefore no potential industrial source of explosion. There is a gas station, at about 800m distance from the HFR as the crow flies, also supplying LPG. This gas station is supplied by tank trucks; no pipelines are present. However, high dunes are separating the gas station from the OLP, so any explosion from this source is directed upwards, leaving the OLP unaffected. The possibilities for causing an explosion on the OLP consist of gas tanks (including hydrogen) and gas supply for heating purposes.

Transportation accidents are thinkable as either road accidents on the nearby Westerduinweg or shipping accidents on the sea. Railways accidents can be excluded on the grounds of distance (8 km).

Road accidents, including those involving LPG trucks from the gas station mentioned above, can be excluded as well, as the high dunes are separating the Westerduinweg road from the OLP over the full relevant length of the road.

In case of explosion of a shipload on the sea close to the OLP or even stranded on the beach, the distance to the OLP and the dunes will protect the site. The distance from the sea to the DWT and WSF facilities is about 240m, and the distance to the HFR and its primary pump building about 360m. According to the Ministry of Infrastructure and the Environment, there are no shipping lanes close to the Petten coast. The lanes for various ships including container ships are about 27 km from the coast, and those for ships carrying hazardous chemicals about 90 km from the coast. To protect the oil drilling platforms and off-shore wind farms close to the coast, a tugboat is permanently on stand-by in the harbour of Den Helder. This makes the probability of stranding of a ship with explosive chemicals very small. In case it would still happen, the dune strip between the sea and the OLP will protect the OLP sufficiently. In case the chemicals would not explode at the location of the stranded ship, but form a gas cloud instead, drifting towards the OLP, the elevated dunes will cause sufficient dispersion by the time the cloud arrives at the OLP.

Directly adjacent to the OLP site, a naval artillery test range is in use. Direct hit by a military missile can be excluded, as the cannon is mechanically fixed in the sea-directed position. The transport route of the ammunition is over the OLP site. Although the test range has been taken into account in earlier HFR safety reviews, investigations into the nature and amount of explosives present during testing are not yet completed. Impact distance (pressure wave) has to established.

Recommendation: The on-going investigations on the explosives present in the naval artillery test range during testing will be monitored, and it should be made sure that the safety functions of the OLP nuclear facilities remain unaffected during accidents with these explosives.

Part of OLP, but not part of the NRG facilities under consideration for the safety margin assessment, are the organisations Netherlands Energy Research Foundation ECN, the Joint Research Centre JRC of the European Commission and the pharmaceutical company Covidien.

At ECN, various kinds of hazardous material are stored. The storage is distributed over a large number of locations that are established according to the guideline PGS15[47]. Besides this, a number of storage tanks for bulk storage of liquids and gases are present, including hydrogen and propane (to be considered as potentially explosive). In a risk analysis performed in 2010, the risks of the hazardous materials stored at the ECN premises have been analysed. The risk contours of the storage locations have been determined, and it has been shown that also the 10^{-8} risk contours[48] do not enter the NRG premises, but remain on ECN grounds. Therefore it can be stated that an explosion in one of these storage facilities will not affect any of the NRG facilities under consideration.

JRC has one gas storage and two distribution stations, in which the maximum gas volume is 10.000 l. In the distribution stations, the material of concern is primarily hydrogen, in gas cylinders or cylinder

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[47] "Publicatiereeks Gevaarlijke Stoffen, part 15", governmental guideline for the storage of hazardous materials.

[48] A risk contour of 10^{-8} means that inside this contour the probability of a person to die from the explosive force on that location is 10^{-8}.

bundles. In the storage, various mixtures and inert gases are stored. The possible effects on the HFR are mentioned in the environmental permit by the local authority, and described as not significant. Also for the High Pressure Gas Testing Facility at the JRC premises, agreements have been made with the nuclear licensing authority KFD, including a strength analysis of the building and a Hazard and Operability Analysis (HAZOP). However, impacts on the other OLP facilities under consideration (HCL, WSF etc.) are not mentioned. But as all other OLP facilities are located on larger distances from the JRC premises than the HFR, it can be assumed that the HFR analysis will cover the other facilities as well.

A possible explosion source may be the pressure reducing station of the natural gas distribution network close to the JGL and HCL. This building dates back to the early days of the OLP (1964), and has been classified as EX zone, with special instructions to prevent explosions. However, information on the effects of an explosion or fire originating from this source could not be found.

Recommendation: The effects of an explosion/fire originating from the natural gas pressure reducing station located on the OLP will be analysed.

The scenario of explosion at the cooling water inlet at the Noordhollands Kanaal is covered by Section 7.11 (Blockage of cooling water inlet). The distance of the cooling water inlet to the OLP together with the dune row inbetween protect the OLP form direct impact of such an explosion.

7.3.2 Potential consequences for the plant safety systems

The event external explosion does not have consequences for the safety systems of the OLP facilities as far as information was available. The artillery test range, the ammunition transports, and the gas reducing station remain to be checked.

7.4 Internal fire

7.4.1 General description of the event

7.4.1.1 High Flux Reactor

The most important objective of a fire hazard analysis is to demonstrate that the safety systems required to shut down the reactor, to remove decay heat and to prevent radioactive releases are protected from potential fires. First, safe shutdown of the reactor needs to be accomplished. Safe shutdown condition is the point in reactor shutdown where criticality and reactor coolant inventory, temperature and pressure can be maintained over a period of time. For the HFR this implies:

- Sub-criticality conditions as in cold shutdown
- Removal of decay heat
- Containment function

In order to define the different levels of severity of a fire event, and finally to analyse their effects on the safety systems, fire compartments and fire cells are defined for the various plant buildings.

A fire compartment is a building or a part of a building comprising one or more rooms or spaces, constructed to prevent the spreading of fire to or from the remainder of the building for a given period of time. A fire compartment is completely surrounded by a fire barrier.

A fire cell is a subdivision of a fire compartment in which fire separation between items important to safety is provided by protection features (such as limitation of combustible materials, spatial separation, fixed fire extinguishing systems, fireproof coatings or other features) so that consequential damage to the other systems is not expected. For each building the fire compartments and cells have been determined and described in a fire analysis on the HFR. Those cells containing equipment necessary for safe shutdown or that could be a fire threat to the surrounding fire cells have been identified. A fire cell is regarded not threatening the surrounding fire cells if the combustible load is small or if the fire barriers between the exposing fire cell and exposed fire cells are very effective. The cells identified in this way are listed in Table 7-1

Building	Room designation	Description/content
Reactor hall	Sub-pile room	Control rod drives
	Off-gas filter station	Valves off-gas system
	Basement	Cabling
	Reactor hall	Cabling
Reactor outbuilding	RGB-034	Control room
	RGB-035	Room behind control room panels
	RGB-035A	Room behind control room panels
	RGB-K02	Water supply waterworks
	RGB-K04 & RGB-K17	Cabling and reactor protection and control system
	RGB-K04A	Cabling
	RGB-K11	Battery room
	RGB-K12	Emergency and uninterrupted power supply room
	RGB-K16	Containment penetration panel
Primary pump building	PPG-Cell 12	Decay heat removal pumps
	PPG-G01	Control cabinets decay heat removal pumps and cabling
	PPG-013	Switchgear room
	PPG-K04	Cabling
	PPG-K05	Cabling
	PPG-K01 and PPG-K02	Cabling
Air Treatment Building	Monitor room	Gas monitors 1, 2 and off-gas monitor
	Ventilation system room 3	Stack valves
	Ventilation system room 10	Containment intake valves and bypass valves
	Ventilation system room 15	Containment exhaust valves
Emergency power building		All equipment
Secondary pump building		All equipment

Table 7-1: Relevant fire cells in the HFR buildings

An internal fire can also cause failure of electric cables, like short or open circuit, short to ground, hot shorts and secondary ignition. These have been evaluated as well. Here three types of cables are distinguished: those that support reactivity control, those that support decay heat removal, and those that support containment of radioactivity.

7.4.1.2 Hot Cell Laboratory

The safety relevant event is loss of containment function of the hot cells. If no fire occurs in the cell, the containment function of the hot cells is maintained as long as the cell ventilation system or off-gas system keeps functioning, depending on which system is used. In case of fire in a cell, the containment function will be maintained if the cell ventilation system continues to operate and the amount of extinguishing agent introduced into the cell does not exceed the extraction capacity of the ventilation system. Loss of containment function can take several forms which are not all equally serious. The least serious scenario is if the under-pressure in the cell fails or becomes equal to the pressure in the space outside the hot cell. More serious is when an overpressure occurs in the cell but the extraction system including the filters continue to function normally. The most serious situation occurs if the fire causes the filters to fail and radioactive materials are discharged into the environment.

A fire outside the hot cells can cause loss of containment function if the cell ventilation fails or the PVC pipes are affected. The latter is only possible at the F-cells and G cells. However, a large fire in the lead cell hall with the effect of damaging the PVC pipes of the cell ventilation system is very unlikely because of the small amount of combustible material in this space. A fire resulting in loss of the ventilation system is however possible. One should think of loss of power supply caused by fire or a fire in the room where the ventilation system fans are located. A fire with loss of power supply is possible with a fire in the switch room (37), the basement and the vertical cable shaft, the pipe corridor on the first floor, and with a fire in the ventilator room. A fire in the switch room will generally result in loss of power supply, regardless of whether the fire is extinguished quickly or after a longer time. In the other rooms, it may be possible that non- energy related components will catch fire first, posing a threat to energy supplies only after some time. In this case, the possibility of extinction is to be considered.

In general, the fire fighting method in a hot cell is to stop the air supply, the full opening of the exhaust and manual supply of extinguishing material. It may be that too much extinguishing agent is supplied and the under-pressure in the cell is lost, or worse, that overpressure in the cell occurs. In addition, the ventilation system may fail due to the high temperatures of the gases to be discharged. In the cell exhaust duct of the concrete cells A, B, C, D and E spark catchers are installed. This is not the case with the G-cells, the F-cells and in cells operated by Covidien. During a fire in the G-cells, including the hot cell L2, the operator will supply CO_2 as extinguishing agent. Here the air supply is not closed and the air exhaust remains operational.

With regard to the actinide lab, the suction of the boxes and the extraction of the cells in the creep laboratory are connected to the off-gas system. Failure of the off-gas system causes therefore loss of containment function of these boxes or cells. Without direct intervention, a fire in an A-box could lead to a fire outside the box due to the damaging of the flammable gloves. Once the fire is outside the box, a fire alarm will be generated by the fire detectors present in the lab. Besides this, a fire could also emerge in one of the electrical cabinets present in the laboratory. Also in this case the operator can attempt to extinguish the fire. A fire alarm will be generated by the detectors in the laboratory. In both cases, at not extinguishing in time the fire may spread to the ceiling, and eventually to other combustible materials (floor) in the laboratory. Depending on whether or not extinguishing in time by the fire brigade, the windows and roof will collapse.

7.4.1.3 Other NRG facilities

Also for the other facilities under consideration, being:

1. Low Flux Reactor;
2. Decommissioning and Waste Treatment Facility;
3. Jaap Goedkoop Laboratory;
4. Waste Storage Facility;
5. STEK hall;

the safety relevant event is loss of containment function. Relevant rooms in all buildings are kept at under-pressure, except for the LFR and the STEK hall. In all buildings some fire load is present, therefore a fire is principally possible. Chemicals are stored in protective cabinets as required. For the LFR, DWT and WSF so-called gauge scenarios have been defined: fire scenarios with normative source terms for consequences for the vicinity.

7.4.2 Potential consequences for the plant safety systems

Fire prevention, detection and control in the workspace are key concerns in the design and facilities at NRG, as fire may result in a loss of containment of radioactive materials and distribution of radioactivity.

Essentially, dealing with fire at NRG rests on three pillars, namely:

- Fire prevention through minimal use of combustible materials and minimization of sources of ignition;
- Early detection of a fire;.
- Extinguish the fire extinguishing systems ranging from manual to incoming deployment of the fire brigade.

The site has its own fire brigade which is standby 24/7. In case of a fire alarm, firemen are ordered to go to the location of the fire alarm and to suppress the fire if necessary. The regional fire brigade provides assistance during fire events. Smoke detectors are present in almost all rooms, which contain safety relevant equipment. The distance between the fire brigade housing and the HFR is about 1 kilometre.

7.4.2.1 High Flux Reactor

The following consequences of internal fire have been identified in a fire analysis on the HFR:

- A fire in the fire cell RGB-K04 or related cells in the primary pump building might cause a loss of primary flow information and thus disabling of the corresponding shutdown signals. The shutdown function 'outlet temperature reactor too high' is still available. In addition to this, the operator can initiate a manual scram after some minutes.
- A fire in fire cell RGB-K04 or RGB-K17 of the reactor outbuilding might disable all shutdown functions, but will finally result in a loss of power supply to the control rod magnets, which causes a scram. It is expected that the operator will initiate a manual scram in case of a fire in one of the basements rooms concerned in the reactor outbuilding. A procedure how to handle in case of fire as recommended in the fire analysis has been implemented, however this has not been documented.
- A fire in the control room might disable several shutdown functions. In spite of this, a shutdown will be initiated because of the loss of power supply to the control rod magnets. An alternative for the operator is to initiate a scram from the sub-pile room by cutting off the power to the control rod magnets causing the control rods to drop into the core.
- A fire in the cells PPG-G01, PPG-K04 or PPG-K05 of the primary pump building might disable not only the decay heat removal pumps but also the primary pumps, resulting in a loss of forced primary flow. The recommendation made in the fire analysis, to separate the power supply and the control of the decay heat removal pumps as much as possible from PPG-G01, PPG-K04 and PPG-K05, has been realized. However, this has not been documented yet.
- In general a fire in the Air Treatment Building will not threaten the reactor. Given a fire in the reactor building the operators will shut down the reactor. In most cases a fire in one of the cables of the ventilation system will result in closure of the containment valves or a transfer to isolation mode.
- A fire in the emergency power building will not disturb the operation of the HFR. Also the distance between the HFR and the emergency power building is such that a fire in the emergency power building will not threaten the HFR. In case of a loss of the emergency power building the normal shutdown procedure to achieve a stable shut down of the HFR can be used.
- A fire in the secondary pump building causing failure of the secondary cooling water flow during normal operation will generate an alarm in the control room. The operator will shut down the reactor

manually. If the operator ignores the alarm an automatic power decrease will be initiated due to a high reactor inlet temperature. If the automatic power decrease fails, the temperature of the primary coolant will rise further until a scram is initiated due to a too high reactor outlet temperature.

- Fire in the cable channels has been analysed, and here a recommendation has been formulated to separate power cables in a fire-resistant way.

Recommendation: The fire analysis of the HFR will be updated in order to document the implemented measures.

It can be seen that these events do not pose a threat to the reactor as long as they are single, separate events. However, when combined with other failures like loss of off-site power or the failure of certain components, core damage cannot be ruled out, as shown in the quantitative fire analysis. In this analysis, event trees have been made for six buildings: the Reactor Building, the Reactor Outbuilding, the Primary and Secondary Pump Buildings, the Air Treatment Building and the Emergency Power building. These event trees are shown in Annex G.1: it can be seen that for the first five buildings the fire must always be combined with either a simultaneous failure of all three scram systems (manual, automatic and alternative) or a simultaneous failure of the pool water injection valves and the convection valves. If this happens, a LUHS scenario develops. The handling of this scenario is described in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink). For the Emergency Power and Air Treatment Buildings, a LOOP and a failure of both the pool water injection valves and the convection valves must occur simultaneously. In this way, a combined LOOP/LUHS scenario develops, as described in the same report.

7.4.2.2 Hot Cell Laboratory

The HCL compartmenting consists of brick walls, fire valves or flow valves (“doorstroomkleppen”). In general, redundant components are in the same room. Also the normal and emergency power distribution are in the same area. The plastic bootings of the manipulators are vulnerabilities with respect to the inclusion of a fire in a hot cell. The use of synthetic bootings requires the quick extinguishing of a fire in a hot cell. In the actinide laboratory, the gloves at the front and rear of the cabinets are made of a combustible material. Keeping the fire in the burning cabinet is therefore dependent on the rapid extinguishing of a fire in a cabinet or of the glove penetrations be sealed when the gloves are not used.

The safety relevant consequence of an internal fire is loss of containment function of the hot cells. The consequences of an internal fire have been analysed for a fire on the following locations:

- Outside the hot cells;

- In the concrete hot cells (A to E);
- In the G-cells;
- In the Covidien cells;
- In the actinide lab.

Besides this, also fires causing loss of power supply have been taken into account.

With fire in a given area different scenarios are possible depending on whether or not detection, whether or not extinguishing, whether or no closure of air supply and so on. Like for the HFR analysis, use has been made of the event tree technique. In the analysis, event trees have been made for a fire outside the hot cells, in a hot cell and in the actinide lab. These event trees are shown in Appendix A.11. In the event tree for a fire outside the hot cells, it can be seen that a loss of containment function is caused directly if the power supply in the building is affected by the fire. In this case an SBO situation develops, as has been described in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink). The event tree on the hot cells shows that either an excess of extinguishing agent or a failure of the ventilation system will cause loss of containment function. The event tree on the actinide lab shows that containment function is lost if the fire is not extinguished by either the HCL personnel or the fire brigade.

Loss of containment will not always lead to radioactive releases, however in most cases accident management actions will be required. These are described in Chapter 6 of this report (Severe accident management).

7.4.2.3 Other NRG facilities

Also for the other facilities under consideration loss of containment function is the safety relevant consequence. Like for the HCL, it will not always lead to radioactive releases.

Accidents in these buildings may lead to the release of liquid radioactive material leading to an increased exposure of employees and liquid emissions in the area. Also at various points in the building contaminated radioactive material is present that could cause release of airborne radioactive materials in case of fire. To prevent this, accident management actions will be required. These are described in Chapter 6 of this report (Severe accident management).

7.5 External fire

7.5.1 General description of the event

External fires may originate from:

a) forest/dune fire in the neighbouring dunes,
b) the facilities of the other organisations at the OLP, and
c) ships or trucks carrying flammable material on nearby transport ways.

Transport of flammable material by ship, rail and road is covered in Section 7.3 of this report (External explosion).

According to the NRG fire department an ignition of the buildings by external fire ("duinbrand") is possible, however only by indirect means ("vliegvuur" sparks). The rationale is the distance of woodlands / scrubs to the plant. The fire load of the beach grass / marram grass ("helmgras") surrounding the buildings is so low that a spreading of the fire from the trees / scrubs to the buildings via the grass is not possible. Forest fires have occurred in the neighbouring dune forests during the past few years. The forests are however on such a distance from the OLP buildings, that the fires can only be transferred to the OLP buildings by setting fire on the beach grass. Figure 7-1 shows the forest and beach grass areas on and around the Petten research site.

The HFR confinement is made of steel, and has as such a very low probability of catching fire.

Next to the direct possible loss of the control room (causing loss of control and scram) and primary pump building by fire (loss of normal cooling), the smoke can cause operating personnel to be forced to leave the control room as this room cannot be isolated completely. The habitability of the control room is impacted and evacuation of the control room might be necessary. The Emergency Diesels will also suffer from smoke, as the air intake is disturbed.

External fire causing "Loss of off-site power" is not to be expected as offsite power is realised through ground cables and not by overhead lines and an open switch yard.

Part of OLP, but not part of the NRG facilities under consideration for the safety margin assessment, are the organisations Netherlands Energy Research Foundation ECN, the Joint Research Centre JRC of the European Commission and the pharmaceutical company Covidien. The flammable gas storage at these organisations is covered in Section 7.3 of this report (External explosion). A fire in the hot cells of Covidien is similar to a fire in the HCL, described in Section 7.4 of this report (Internal fire).

A possible fire source may be the pressure reducing station of the natural gas distribution network close to the JGL and HCL. This has been covered in Section 7.3 of this report (External explosion).

7.5.2 Potential consequences for the plant safety systems

No consequences to the safety systems of the OLP facilities under consideration are to be expected from dune fires. Sparks may land on OLP buildings, but no fire load is available to cause the buildings to heat up sufficiently. Also, the whole fire fighting strategy on the site is aimed at preventing this by spraying water on the buildings.

At Covidien, small amounts of gases are available as allowed for laboratories, like methane for radiation counters. The two cyclotrons contain a small amount of hydrogen. Like the HCL, the Covidien laboratories are classified as so-called B-objects for the Dutch nuclear law ('Kernenergiewet'), for which a fire analysis is part of the licensing procedure. Similar consequences as for the HCL apply here, as described in Section 7.4 of this report (Internal fire).

For the HFR, control room habitability may become an issue with an external fire. As the control room cannot be isolated completely, smoke entering the room could become a problem. This aspect is described in Section 7.7 of this report (Toxic gases).

It appeared that in case of a dune fire the OLP was informed inadequately and only at a late stage. A policy and procedure are missing.

Recommendation: establish a communication protocol from the regional fire brigades to the OLP in case of fire or other external hazards.

Figure 7-1: Aerial view of the Petten research site (OLP), showing the forest and beach grass areas.

7.6 Airplane Crash

7.6.1 General description of the event

7.6.1.1 High Flux Reactor

In the initial plant design, Airplane Crash (APC) has not been considered as a relevant load case since the site is located outside the direct influence range (flying zones) of commercial and military airports. However, from 2001, this event has been given consideration. According to the Nuclear Safety Rule NVR 3.1, "Guideline for the protection against external effects", the ruling scenario for the HFR would be a crash of a heavy military fighter plane. This accident is also chosen for the ruling scenarios for the other nuclear installations at the Petten site and has been analysed to the level of source terms and effects of the radioactive releases on the vicinity.

In case the aircraft crashes in such a way that the reactor hall is not affected, but only one or more of the other HFR buildings, then the reactor will be shut down, but kept intact. The hit buildings and their equipment will however be lost.

In case the aircraft crashes directly into the reactor containment, the reactor pool, shutdown systems and cooling systems could be damaged in such a way that core damage cannot be prevented as well as release of radioactivity. In the analysis, this type of crash has been assumed as the decisive event. Damage of the other buildings of the HFR has limited additional impact on the release and dose.

The fighter plane would be of the heaviest type, weighing 20 tons, having an impact velocity of 774 km/h on these installations, and carrying of 900 kg kerosene. Heavy parts of the plane will hit the reactor pool inside the reactor building after penetration of the dome.

In the analysis it is assumed that the reactor core is still covered by a few meters of water after the crash. However, it is more likely that major cracks will occur in the pool walls and/or bottom resulting in substantial leakage through these cracks and ultimately provides an uncovered core. Additionally, if heavy equipment, like parts of the dome or parts of the plane, will fall into the reactor pool, it will damage the installations inside this pool as well as the pool walls/bottom.

Besides the mechanical damage there is also damage by the kerosene fire. Another effect of the kerosene fire is the formation of a smoke plume that will take radioactive material up into the air, resulting in lower concentrations of radioactive materials at ground level nearby the site and therefore lower maximum individual doses for the nearby population. However a larger area will be affected.

The fighter plane is seen as having maximum destructive effect, because of the fact that all weight is concentrated in the engine. A large passenger aircraft will be able to destruct a wider area, and has a much larger amount of kerosene. The fighter plane, has only a small impact area to consider (cross section of 7 m^2), and therefore the probability of having a severe damage of two or more buildings is small. With a large passenger aircraft this becomes much more probable. However, the study referenced above indicates, damage of the other buildings besides the HFR reactor hall has limited additional impact on the release and dose.

7.6.1.2 Other NRG facilities

For the other facilities, APC always includes (partial) destruction of the building and therefore loss of containment function and potential release of radioactive material. Only if a building is not directly hit, but just affected by the kerosene fire, a radioactive release may be prevented. In that case, the event would be similar to the event Internal Fire, described in Section 7.4 of this report.

7.6.1.3 Potential consequences for the plant safety systems

The consequences of an APC for the safety systems of the OLP facilities can be such that core damage cannot be prevented as well as release of radioactive material. For these cases, the National Plan for the Control of Nuclear Accidents (Nationaal Plan Kernongevallenbestrijding) will come into force. This involves organisations like fire brigade, community health service (GGD) and police, with whom preparatory consultations have taken place. Also, Severe Accident Management actions will be initiated, to minimize the consequences for the vicinity, as described in Chapter 6 of this report (Severe accident management).

7.6.1.4 High Flux Reactor

The reactor pool contains three installations with irradiated fuel; the pool side facility, the spent fuel storage and the reactor. The reactor has the largest inventory of radioactive materials of these three installations. Therefore the radioactive release would be dominated by the damage of the fuel in the reactor. In the decisive scenario analysed, the damage of the reactor vessel (deformation and rupture) would result in blockage of the coolant flow in several fuel elements which leads to significant damage of the fuel in the reactor core. Noble gases and iodine from the melted fuel would be released in the pool water. A large amount of radioactive material would be released from the pool water and will escape into the atmosphere via the damaged dome of the reactor building. In Chapter 6 of this report (Severe accident management) is described how to mitigate these consequences, including the Disaster Response Plan OLP and personnel issues (an important part of the personnel will be incapacitated in this scenario).

However, as indicated in the previous paragraph, the decisive scenario analysis mentioned does not entirely cover the most serious consequences. In the analysis it was assumed that the core remains covered by pool water, while it is now seen as more probable that no pool water will be left to cover the core.

Recommendation: The analysis of the consequences of radioactive release caused by air plane crash on the HFR will be extended under the assumption that all pool water is lost.

In case the aircraft crashes in such a way that the reactor hall is not affected, but only one of the other HFR buildings, then the reactor is not threatened immediately, as long as the decay heat removal pumps and the convection valves can be operated. However, if the APC also would cause other failures like loss-of-offsite power or the failure of the secondary cooling system, core damage cannot be ruled out, and the event trees from the fire analysis referenced in Section 7.4 Internal Fire also apply here (Appendix A.10). A LUHS scenario, or combined LUHS/LOOP scenario could develop, as described in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink).

7.6.1.5 Other NRG facilities

Also for the other facilities besides the HFR, an airplane crash would mean immediate loss of containment. For all facilities except for JGL and STEK, APC is included in the safety analysis. For the HCL, WSF and LFR, the radioactive inventory is such that also protective measures outside the OLP could be necessary. For the DWT, JGL and STEK, these amounts are much lower, and a release from these facilities will be so small that the affected area remains limited to the OLP or its close vicinity.

7.7 Toxic gases

7.7.1 General description of the event

In case of an accident with toxic gases, they 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 NRG,
- Chemical present on the premises of the other organisations at the OLP,
- Chemicals present outside the OLP

The chemicals could be released by a leakage, a fire (Chapter 5 and 6) or an explosion (Chapter 3 and 4).

In the cellar of the HFR building, carbon monoxide is present in the form of maximum three 10 litre cylinders (97,6 % CO).

At ECN, various kinds of hazardous material are stored. The storage is distributed over a large number of locations that are established according to the PGS15[49]. Besides this, a number of storage tanks for bulk storage of liquids and gases are present, of which the chemicals hydrogen fluoride (HF), sulphur dioxide (SO_2) are known as toxic. The risks of the hazardous materials stored at the ECN premises have been analysed. The risk contours of the storage locations have been determined, and it has been shown that also the 10^{-8} risk contours do not enter the NRG premises, but stay on ECN grounds, see Section 7.3.1. Therefore it can be stated that a toxic release from one of these storage facilities will not affect any of the NRG facilities under consideration.

Both at JRC and Covidien no toxic chemicals of sufficient significance are available.

Chemicals released by ship or truck accidents in the vicinity of the OLP do not pose a threat as both the beach, the channel Noordhollands Kanaal and the nearest public road are at such a distance that a potential toxic cloud will be dispersed before arrival at vulnerable OLP buildings. Also, there is no industry present in the vicinity that could potentially be a source of a toxic chemical release.

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[49] "Publicatiereeks Gevaarlijke Stoffen, part 15", governmental guideline for the storage of hazardous materials.

7.7.2 Potential consequences for the plant safety systems

As the HFR control room cannot be isolated, a toxic cloud could potentially incapacitate control room personnel. In that case, the HFR will continue to run as long as no malfunctions or resource shortages occur. Although the reactor will shut down after one or one and a half day when the automatic control system will reach its limits, in case of a toxic gas release event the reactor should be shut down as soon as possible. This is a manual action. Cooling water pumps will be switched on automatically after shutdown.

Recommendation: The protection of the control room against toxics, smoke etc. would improve the margin in case of other events.

Remark: A modification plan (CR-1) to implement the above mentioned measure is in preparation (review).

7.8 Large grid disturbance

7.8.1 General description of the event

Large grid disturbances, including overvoltage, voltage interruption /short circuit and frequency transients, could cause damage to the house transformers and the electrical systems behind it; in the worst case causing their permanent unavailability or causing damage to safety systems such as the decay heat removal pumps.

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 (a higher voltage than usual is offered over the grid);
- 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 Cooperation Electric Utilities 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.

The OLP is connected to the public electrical grid through two separate 10 kV connections. These are connected to the regional 50 kV grid through a transformer at the switchyard near Schagen, about 10 km away. In the field the 10 kV feeders are rolled out separately at a distance of ca. 1 km from each other.

The 50 kV grid is connected to the main 150 kV grid near Schagen as well, through another transformer. A more detailed description can be found in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink).

The protection consists of an undervoltage protection on the 0,4 kV internal grid of 90% during 4 seconds.

7.8.2 Potential consequences for the safety systems

Large grid disturbance problems typically occur in large high voltage grids, like the 150 and 380 kV grids where the power plants are connected to. The distance of the OLP to the nearest 150 kV connection, together with the double down transforming to 50 and 10 kV respectively, is a sufficient natural protection of the OLP safety relevant systems.

Theoretically, large grid disturbances could result in the damaging of equipment (transformers, busbars), resulting in long unavailability of the facilities, however without safety impact. 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 for restoration of offsite power,
- Unavailability of safety related electrical components, with as most serious consequence the loss of the UPS, supplied by the emergency power supply 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 OLP. Until then, the plant will be in a situation of Loss Of Off-site Power (LOOP), as described in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink).

7.9 Failure of systems by introducing computer malware

7.9.1 General description of the event

In this report, “failure of systems by introducing computer malware” implies accidental invasion of malware in the OLP computer systems. Malware, short for malicious software, is software (or script or code) designed to disrupt computer operation, gather sensitive information, or gain unauthorized access to computer systems. It is a general term used to describe any kind of software or code specifically designed to exploit a computer, or the data it contains, without consent. The expression is a general term used by computer professionals to mean a variety of forms of hostile, intrusive, or annoying software. Malware includes computer viruses, worms, trojan horses, spyware, dishonest adware, most rootkits, and other malicious programs.

This accidental invasion could be accomplished either by users on the NRG Petten premises, or by those with an internet connection.

The safety related systems of the facilities are mostly controlled by non-computerized electronic hardware. Irradiation experiments and other experiments with radioactive material are often equipped with PCs, but these are never connected to the facility’s safety systems. Programmable logic controllers (PLCs), components potentially vulnerable for accidental malware contamination, are known to be used in some cases in the safety systems of the facilities under consideration, but they are not connected to the NRG Intranet / Internet. An exception here is the ECN-GBS, a computerized domotics system that operates via the main ECN-NRG computer network on the OLP. From the central engineering station of this system it is possible to execute control actions in certain ventilation systems and in doing so on the under-pressure situation of these buildings.

At the moment however, there is no complete overview of the existence and connection of PLCs in the facilities of the OLP under consideration. There is also no policy yet to acquire and maintain this overview or the vulnerability for malware in general.

Recommendation: All safety systems will be checked for the presence of PLCs and their vulnerability for malware. A general policy of checking vulnerability for malware with facility hardware adaptations and upgrades should be established.

The HFR control room is equipped with a computerized presentation system (DACOS) for retrieval and analysis of experimental data that cannot be accessed from outside. Theoretically it could be accessed from inside, and false information could hypothetically enter through this access, and therefore cause confusion under the control room personnel. There is no connection however to the HFR's safety systems. The same applies to the other computerized process presentation system (DAS) for process information in the HFR control room.

The electrical main switchboards of the HFR are equipped with a computerized parameter control and monitoring system. It is not connected to an external network. Its control capability is blocked except for the possibility to set parameters of the thermal overload protection. Inadvertent use of this facility is prevented by a hardware jumper regime.

7.9.2 Potential consequences for the plant safety systems

The safety systems of the OLP facilities are mainly controlled by non-computerized electronic hardware. Also the HFR's reactor protection system and other safety related systems are not processor or PLC controlled. The process and experiment information computers are solely supportive for plant control: it only shows information, but does not control plant systems. Although some equipment have PLC's in their start-up circuits their failure will not affect safety, because they are programmed in a fail-to-safe mode. They are not connected to any control network and even in the event that they would fail it will never prevent manual start-up. All this offers sufficient protection on forehand for the reactor and its safety systems.

Theoretically the experiment presentation system (DACOS) and the process information system (DAS) in the HFR control room could be accessed from inside, and false information could hypothetically enter through this access, and therefore cause confusion under the control room personnel. However, according to their procedures they will only take action if the DAS or DACOS information is confirmed by readings of the analogue instrumentation on the control console and wall panels.

From the central engineering station of this system it is possible to execute control actions in certain ventilation systems and in doing so on the under-pressure situation of these buildings. In case such an action would succeed, the containment function would be lost un these facilities. For follow-up on this event reference is made to Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink).

It is not known whether PLCs are totally absent in the safety systems of the other facilities under consideration, besides those mentioned. They need to be located systematically, and their vulnerability for malware checked.

7.10 Internal flooding

7.10.1 General description of the event

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 facility. In the HFR buildings, this might originate from leakage of the cooling water or demin water systems. Furthermore, the public water supply system (PWN) is a potential source of flooding in all buildings.

The design criteria to mitigate floods and their consequences focus on:

- Appropriate design: this means that by plant layout and system design, mitigation is effective e.g.
    - by proper design to prevent leakages
    - by flow paths to accumulation areas
    - by drain systems
    - by protecting electrical systems against water ingress when submerged or wetted
    - by thresholds, protecting other areas
- Procedures: normal operating and/or emergency operating procedures will support/guide operator actions to control floods.
- Detection systems: means for detection of leakage or floods should be available. For leakage detection these means can refer to system process parameters like the level, flow, temperature and pressure of the water. For detection of floods means can be based on flow or mass detection of drain systems or water level indication in (assigned) accumulation areas.

7.10.1.1 High Flux Reactor

Identified water sources

Water originates from bulk stored inside or from systems having large mass flow capacity while externally supplied. Large water volumes “stored” inside the HFR building are:

- The primary cooling water system
- The pools (reactor pool, pool 1 and pool 2).

External sources with large quantities of water feed are:

- The secondary cooling water system
- The public water supply system

For the separate buildings the following systems are potential flooding or spray sources:

The Reactor Building:

- The primary cooling water system
- The pools and connecting systems
- The public water supply system (fire fighting system)
- Scour water

The Primary Pump Building:

- The primary cooling water system
- The pool cooling water system
- The secondary cooling water system

The Reactor Outbuilding:

- The public water supply system (fire fighting system)
- The overflow of the primary pump building in case of a very large leakage flow
- Central heating system

The Secondary Pump Building:

- The secondary cooling water system

For these systems leakages flow rates and released water volumes have been determined for the largest pipes.

Affected areas

From plant and system layout, for each building areas are selected for their large volumes of water passing or collected (end of flow paths) and listed in Table 7-2.

Building	Affected area
Reactor building	reactor building basement
	sub-pile room
	pipe corridor
Primary pump building	connecting duct PPG-K02
	debasement PPG-K02
	primary pump building basement PPG-K01
Reactor outbuilding	reactor outbuilding basement
Secondary pump building	secondary pump building basement

Table 7-2 Areas of the HFR affected by internal flooding.

Per building, the areas are sometimes connected to adjacent ones by open passages. In the reactor building basement, the various areas are connected through the drain system. Next to that, an open connection exists between primary pump building and reactor outbuilding (important in case of release of very large water volumes in the Primary Pump Building with the reactor protection system located in the Reactor Outbuilding).

Removal of water from the Reactor Building is performed by the drain system through a central pit in the sub-pile room. When the storage tanks of this system in the Primary Pump Building are filled, overflow to the area PPG-K02 will occur.

Affected systems/components

In the areas identified, the following safety related systems have been selected based on their location in the building and their impact on safety related functions when exposed to floods or sprays.

- **Reactor Building:**
  - The nuclear safety channels; especially the electronic cabinet of nuclear channel 10 to 12 (about 0.6m above basement floor). Submergence of these components may result in spurious signals to the reactor protection system. Basically this will be indicated by the reactor protection system as an invalid signal compared to the redundant channels NC 5 to NC7.
  - Control drive actuators and magnets are situated at an elevation of about 2 meter in the sub pile_room. Submergence of these components may induce reactor scram, as the system will be short-circuited. When this system is sprayed by water, the system will also fail because it is not made waterproof. Reactor will scram automatically.
  - The off gas filter-station, important with respect to filtering of containment air. Submergence of this system will cause failure of this system but confinement is still assured.

- Experiment cabinet COBO4, can be sprayed by water if scour water/demin water from platform three is leaking on this system. The system will fail and temperature control of the experiment is lost, causing reactor shutdown.
- Cabling and tubing of nuclear safety channels, control rod magnets and motors and off-gas system. If these cabinets are sprayed by water, these systems could fail, as they are not waterproof. Losing the functionality of these systems will result in a reactor scram.

- **Primary Pump Building:**
  - Cable trays to components in the primary pump building carrying electrical power and control cabling. Submergence of these cable trays if splices are exposed, will cause short-circuiting of the power and control system. The reactor will be shut down, and in the worst-case primary cooling and decay heat removal will be stopped.
  - Isolation valve O-032 of the off-gas system in PPG-K02 provides isolation from the off-gas system of the PPG in case isolation of the reactor building is required. Submergence of this valve will cause the valve to fail but since the valve is designed to be fail-safe, the valve will be closed. The confinement function is assured.

- **Reactor Outbuilding:**
  - The reactor protection system is situated in the basement. Submergence of this system will cause failure of the power supply (220V) and loss of instrumentation of the reactor protection system. This will result in a reactor scram and control room black out.
  - The Uninterrupted Power Supply and the bus bar of the emergency power are situated at basement level. Submergence results in loss of DC power, emergency power, instrumentation etc. This will cause a flood induced scram since the magnets will be de-energised, resulting in a drop of the control rods into the reactor core.
  - Process monitoring panel in RGB-K04 could be sprayed by water and because the system is not waterproof, it will result in a loss of monitoring process parameters in the control room. The reactor will scram but shutdown conditions cannot be monitored.
  - Electrical cabinet of emergency communication/alarm annunciation systems in RGB-K17 could be sprayed by water. Since the system is not waterproof, it will result in a failure of these systems. All alarms and controls used in the control room is lost and the reactor will scram.

7.10.1.2 Other NRG facilities

For all buildings, it is assumed that leaking water is flowing down and ends in the cellar of the building. The water level cannot rise above ground level as it flows outward through the doors.

Internal flooding generally causes failure of electrical systems, failure of power supply and inaccessibility of flooded areas. For most areas the flooding has no consequences; this has been described in detail in Chapter 3 of this report (Flooding). Here the remaining consequences are summarized:

- Blocking of the ventilation of the "pluggennesten[50]"; no further consequences.
- Floating of contaminated waste tanks in the cellars of the HCL MPF, DWT (Water Treatment Building) and JGL, followed by rupture of the connecting pipes and release of the tank contents.
- Release of contaminated sump water from the cellar of the DWT Decontamination Building.

7.10.2 Potential consequences for the plant safety systems

7.10.2.1 High Flux Reactor

The consequences of internal flooding of the HFR buildings have been analysed. A summary follows below.

Reactor building

Because water is collected either in the pit of the sub-pile room or the pipe corridor, the drain system will immediately start removal. In case this system fails, this volume will be spread over the entire free area (basement, sub-pile room and pipe corridor). Because all electrical equipment is installed at a level of 1 m or higher, none of this equipment will become submerged, as the amount of water is limited. The connections of the nuclear channels 10 to 12 (high flux measurements) are installed at an elevation of about 0.55 m, just high enough not to be submerged. Failure of these nuclear channels will be detected by the reactor protection system, but will not affect safety systems.

Secondly, the off-gas system will be flooded and therefore fail certainly. This means that filtering of the containment atmosphere will stop as well as the release of filtered air to the environment.

[50] Storage facility consisting of vertical storage pipes in the floor.

It is noticed that alarms indicating flooding will occur when the drain system starts removal of water from the reactor building. A second indication is given when the drain tanks spill over into the area PPG-K02.

The safety functions are maintained as follows:

- Shutdown: the reactor will scram automatically when the water level of the pool decreases more than 0.5m.
- Core cooling: decay heat removal system and secondary cooling system are functional. Permanent leakage from primary system to the pool is small and is compensated by the fill- and drain system. Core cooling is assured for a long period because from the cooling water supply tanks (ca. 165 m3 available).
- Confinement: isolation of the containment is functional.

Primary pump building

In the primary pump building, water will be accumulated in the following sequence: first the basement of area PPG-K02 will be filled up, then the connecting duct while the remaining volume will fill the entire basement of the primary pump building. Overflow to the reactor outbuilding will occur within 30 minutes. Consequences of this event are:

- The reactor will be scrammed due to loss of secondary cooling (high outlet temperature primary system). Primary system and pools establish the ultimate heat sink by which residual and decay heat are removed.
- Electrical and I&C equipment in both primary pipe building and reactor outbuilding are flooded within 30 minutes. This will cause failure of power supply (380V), failure of the reactor protection system and of UPS (flooded batteries). As a consequence additional scram and a control room blackout will be initiated.
- The isolation valve O-032 of the off-gas system will be submerged. This valve will fail in closed position with the result that primary pump building will be isolated from the reactor building.

Reactor outbuilding

After fill-up of the room K16 by which the conducting plate of the reactor protection system from out-building to reactor building is submerged, the water level in the outbuilding starts rising. As no flood indication is installed, a flood will be detected only when spurious signals from the reactor protection system occur or when the reactor protection system or DC systems fail. Again then the reactor will be scrammed automatically and control room blackout will occur.

With regards to the safety functions, the following is concluded:

- Shutdown: when the power supply is lost, the magnets of the control rod drive system are de-energized and the control rods will drop into the core (flood induced scram). Until loss of the reactor protection system occurs, it is assumed that single spurious signal that originate from flooded control system cables, do not affect the system.
- Core cooling: primary system, its content and pools (when in connection with the reactor core) establish the ultimate heat sink.
- Confinement: Off-gas- and ventilation system are available so confinement functionality is assured.

Secondary pump building:

The basement of this building can be flooded without affecting any safety relevant system. In continuation water will be spilled outside the building. As secondary cooling is lost, the reactor will scram and residual heat will be removed by the primary system and the pools, as described in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink).

It has been checked that the electrical switchgear room and the installed equipment will not be flooded in this situation.

Additionally, the internal flooding event for each building can occur combined with other failures, like loss of off-site power or the failure of certain components. In the quantitative internal flooding analysis, the combination of events that lead to core damage have been analysed. Event trees have been made for two buildings: the reactor outbuilding and the primary pump building. These event trees are shown in Appendix A.12: it can be seen that for both buildings the flooding must always be combined with either a simultaneous failure of all three triggers of scram systems (manual, automatic and alternative) or failure of the convection valves. If this happens, a LUHS scenario develops. The handling of this scenario is described in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink).

7.10.2.2 Other NRG facilities

For the other NRG facilities, it is assumed that the cellar of the buildings is flooded due to the internal flooding event. As described in Chapter 3 of this report (Flooding), the consequences are as follows.

Hot Cell Laboratory:

Due to flooding of the cellar the ventilation of the "pluggennesten" is blocked. This has no further consequences.

The following systems are flooded:

- the storage pool and the "pluggennesten": no consequences, because the drums are water tight.
- the second storage pool with "filterbussen": no consequences, because the pool is already filled with water.

In the MPF part, the liquid waste tanks will float due to flooding of the cellar as they are not bolted down, resulting in pipe break and ultimately release of the contents of the tanks.

Low Flux Reactor:

Flooding of the cellar with the fuel storage will result in water entering the storage "kokers". The fuel storage drums are submerged. Fuel (used as well as fresh) does not become critical. Containers are watertight, so possibility for releases does not exist.

Decommissioning and Waste Treatment Facility:

Water Treatment Building: Flooding of the pools in the cellar will cause its contents to be released, but this will not give nuclear contamination to the environment because the basins contain clean water.

DWT Storage Building: Flooding of the cellar will cause the drums with filter resins to become submerged. Since the drums are water tight, there is no contamination.

Jaap Goedkoop Laboratory:

Flooding of the cellar will cause submerging the power supply for the laboratories. This results in electricity failure in the laboratories. This has no nuclear consequences. Also the communication equipment will be submerged. The communication between JGL and the outside world will fail. Furthermore, flooding of the cellar with waste tanks will cause these tanks to float, possibly resulting in pipe break and ultimately the release of the contents of the tanks (loss of containment) and therefore contamination.

Waste Storage Facility:

Flooding the cellar will cause the pipes with drums with highly radioactive material to be submerged. The contents will however remain subcritical. The radioactive material will remain inside the pipes, as there are no flow paths for the water beyond the pipes.

STEK hall:

Flooding the cellar will have no consequences, as the cellar is empty.

The recommendations on this event are included in the recommendations of Chapter 3 of this report (Flooding).

7.11 Blockage of cooling water inlet

7.11.1 General description of the event

Blockage of the cooling water inlet could basically be caused by two initiating events:

1. ship grounding
2. biological phenomena

Blockage by ice during cold weather is addressed in the Chapter 4 of this report (Extreme weather conditions). In the sections below, these phenomena are described, together with their impact on the plant's safety systems.

7.11.1.1 Ship grounding

In this event, a ship on the channel Noord-Hollands Kanaal loses its normal direction and runs ashore exactly at the location of the cooling water inlet, blocking it.

7.11.1.2 Biological phenomena

This hazard is defined in terms of impact on the plant of organic material in intake water. The material may be algae, water plants, fish, mussels etc. The event is rather theoretical, as never any plants or animals showed op in quantities anywhere near the amount needed to cause a cooling water inlet blockage.

The intake is inspected weekly. Sudden blockage will be detected by the secondary water level measurement in the Secondary Pump Building.

7.11.2 Potential consequences for the safety systems

Blockage of the cooling water inlet at the Noord-Hollands Kanaal is an initiating event for the LUHS scenario. The handling of this has been described in Chapter 5 of this report (Loss of electrical power and loss of ultimate heat sink). However, a long term Loss of UHS scenario can be excluded for the causes named above, as it is always possible to remove the organic material or the stranded ship with limited additional resources within a sufficiently short period of time. The loss of heat sink has no immediate consequences for the operation of the reactor, as described in the same report. In this case cooling can be maintained over the pool via the convection valves; sufficient means are available to cool or replace the heated pool water. A postulated long-term Loss of UHS is also elaborated in Chapter 5.

List of tables

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List of abbreviations

AC	Alternate current
APC	Air Plane Crash
ASDS (ASS)	Alternative Shut Down System
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
AUHS	Alternate Ultimate Heat Sink
BAC	OLP Alarm Centre (Bedrijfs Alarm Centrale)
BDBA	Beyond Design Basis Accident
BEKWS	Pool Facility Cooling Water System
BKWS	Pool Cooling Water System
BLEVE	Boiling Liquid Expanding Vapour Explosion
CAS	Central Alarm Station
CMP	OLP Fire Brigade Alarm Centre (Centrale Meldpost)
COVRA	Centrale Organisatie Voor Radioactief Afval (Dutch organisation for radioactive waste storage)
CRM	Centrale Radiologische Monitoring (NRG-RE)
CSA	Complementary Safety margin Assessment
DACOS	HFR Data Acquisition System (experiments)
DAS	HFR Data Acquisition System (process)
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DC	Direct current
DWT	Decontamination and Waste Treatment
ECC	Emergency Coordination Centre
ECN	Energie-onderzoeks Centrum Nederland (Netherlands Energy Research Foundation)
EMP	Electromagnetic Pulse
EMS	European Macro-seismic Scale

ENSREG	European Nuclear Safety Regulator Group
EOP	Emergency Operating Procedure
EPS	Emergency Power System
EPSS	HFR Electrical Power Supply System
ERO	Emergency Response Organisation
EU	European Union
GGD	Gemeentelijke Geneeskundige Dienst (community health service)
HAVA	Hoog Actief Vast Afval (high level radioactive waste)
HAZOP	Hazard and Operability Analysis
HCL	Hot Cell Laboratory
HD-BEKWS	High pressure pool facilities cooling system
HEU	High Enriched Uranium
HFR	High Flux Reactor
HSC	HFR Safety Committee
I&C	Instrumentation & Control
IAEA	International Atomic Energy Agency
JGL	Jaap Goedkoop Laboratory
JRC	Joint Research Centre
KFD	Kernfysische Dienst (Dutch nuclear regulator)
KNMI	Koninklijk Nederlands Meteorologisch Instituut (Dutch national meteorological institute)
KOTS	Koelwateropslagtanks (cooling water storage tanks)
LAVA	Laag Actief Vast Afval (Low level radioactive waste)
LBG	Luchtbehandelingsgebouw (Air Treatment Building)
LEU	Low Enriched Uranium
LFR	Low Flux Reactor
LMH	Lage Montagehal (construction hall of HFR)
LNG	Liquid Natural Gas
LOCA	Loss Of Coolant Accident
LOOP	Loss Of Off-site Power

LPG	Liquid Propane Gas
LSO	Laboratorium voor Sterk stralende Objecten (Dutch abbreviation for HCL)
LUHS	Loss of Ultimate Heat Sink
MAVA	Middel Actief Vast Afval (Intermediate level radioactive waste)
MCR	Main control room
MMI	Modified Mercalli Intensity scale
MPF	Molybdenum Production Facility
n.a.	not applicable
NAP	Normaal Amsterdams Peil (Amsterdam Ordnance Datum)
NBP	Nucleair Basis Peil (Nuclear Reference Level)
NC	Normally Closed (circuit breaker)
NDE	Non-destructive examination
NO	Normally Open (circuit breaker)
NPP	Nuclear Power Plant
NRG	Nuclear Research and consultancy Group
NV	HFR Uninterruptible Power System for decay heat removal
NVR	Nucleaire Veiligheids Regel (Nuclear Safety Rule)
OLC	Operational Limits and Conditions
OLP	Onderzoeks Locatie Petten (research location Petten)
PCV	Pool Cooling Valve
PE	Polyethylene
PEKWS	Primary facility cooling water system
PGA	Peak Ground Acceleration
PIE	Postulated Initiating Events
PKWS	Primary Cooling Water System
PLC	Programmable Logic Controller
PPG	Primair Pomp Gebouw (Primary Pump Building HFR)
PSF	Pool Side Facility
PVC	Poly Vinyl Chloride

PWN	Provinciale Waterleidingmaatschappij Noord-Holland (Public water supply company)
RBP	Research Location Petten Disaster Response Plan
	(Rampenbestrijdingsplan Onderzoeks- en bedrijvenlocatie Petten)
REWAS	Reserve Elektrisch Warmte Afvoer Systeem
RGD	Rijks Geologische Dienst (Dutch governmental geological organisation)
RL	Research Laboratory
RPS	Reactor Protection System
RSC	Reactor Safety Committee
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SBO	Station Black-Out
SC	Short Circuit
SKWS	Secundair Koel Water Systeem (Secondary Cooling Water System)
SPG	Secundair Pomp Gebouw (Secondary Pump Building HFR)
SSC	Structures, Systems, and Components
STEK	Snel Thermisch Experiment Krito (decommissioned test facility)
TGB	Technische Grondslagen voor Bouwvoorschriften (Dutch Building Code)
TIG	Tungsten Inert Gas (a welding method)
U-BEKWS	Extended pool facilities cooling system
UCW	Uranium Containing Waste
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply
VTS	Veiligheids Technische Specificaties (Technical specifications related to safety)
VZO	Uninterruptable Power supply System for reactor instrumentation and control
WENRA	Western European Nuclear Regulators' Association
WSF	Waste Storage Facility

8 Appendices

A.1 Letter to NRG from the Ministry of Economic Affairs, Agriculture and Innovation (reference ETM/ETD/ 1107457)

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Ons kenmerk
ETM/ED / 11074571

Uw kenmerk

Bijlage(n)
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Datum 1 juni 2011

Betreft Uitvoering stresstest

Geachte heer Stol,

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 (Hoger Onderwijs Reactor in Delft, Hoge Flux Reactor en Lage Flux Reactor in Petten, in samenhang met de andere nucleaire installaties op de Onderzoekslocatie Petten). 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 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 Hoge Flux Reactor, de Lage Flux Reactor en de andere nucleaire installaties in Petten.
De huidige Europese afspraken hebben betrekking op kerncentrales en kunnen niet zondermeer op geheel andere installaties, zoals een radionuclidenlaboratorium, worden toegepast. De uitgangspunten voor de stresstest echter wel. Binnenkort zal een afspraak met u worden gemaakt, om de uit te voeren stresstest op de Onderzoekslocatie Petten nader met u te bespreken

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Directie Energie en Duurzaamheid

en specificeren. Daarbij zal de hieronder genoemde planning 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 aan een ernstige storing van de elektriciteitnet, een bosbrand of een neerstortend vliegtuig. Tevens zal rekening moeten worden gehouden met moedwillige verstoringen.

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

Als resultaat zal de stresstest het volgende moeten opleveren:

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

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

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

Om de betrouwbaarheid en de verantwoording van het hele proces te verhogen, zullen de nationale rapporten over kerncentrales, zoals gevraagd door de Europese Raad, worden onderworpen aan een 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. In lijn hiermee zal ik collega's in het buitenland betrekken bij het beoordelen van de stresstest voor onderzoeksreactoren.

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Directie Energie en
Duurzaamheid

Ons kenmerk
ETM/ED / 11074571

Bij het hele EU stresstest process zullen de nationale autoriteiten 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.

Het is de bedoeling de resultaten van de stresstest 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
Naar analogie van het EU voorstel, zal ik voor de stresstest voor de Onderzoeklocatie Petten een planning hanteren waarmee de resultaten uiterlijk medio 2012 definitief zullen zijn.
De eerste prioriteit ligt bij het uitvoeren van de stresstest voor kerncentrales en onderzoeksreactoren.

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

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

Hoogachtend,

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

A.2 ENSREG Declaration and Safety annex I & II EU “Stress Test” specifications

ENSREG
European Nuclear Safety Regulators Group

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.

[1] See annex II

Annex I

EU "Stress tests" specifications

Introduction

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

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

Definition of the "stress tests"

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

This reassessment will consist:

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

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

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

By their nature, the stress tests will tend to focus on measures that could be taken after a postulated loss of the safety systems that are installed to provide protection against 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.

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Technical scope of the "stress tests"

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

a) Initiating events
- Earthquake
- Flooding

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

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

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

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

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

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

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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

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...)

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 emergencyoperating 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;

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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.

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

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

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

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

Two situations have to be considered:

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

For each of these situations:

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

d) Loss of the primary UHS with SBO

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

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

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

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

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

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

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

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

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

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

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

The following aspects have to be addressed:

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

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

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

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

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Annex II

The national nuclear safety authorities should remain associated with this process to facilitate an overall coherent response with respect to prevention, management and mitigation issues. They would share within ENSREG any recommendation that they believe will contribute to the overall response to the stress test exercise.

Progress on these issues should be included in the report to be made by the Commission to the December 2011 European Council.

A.3 Listing of EPS emergency power loads

EPS (NSC) Bus	Connected Loads	Assigned power (kVA)	Installed power (kVA)	System	Remarks
A	DWT [ECN 4] K1(NS)	32	20	Under-pressure	
	HCL [ECN 6] K3(NS)	90	49	Under-pressure	
	HFR [ECN 2] K2A(NS)	300	119	Safety systems and ESFAS emergency cooling	
	AV-board EPS	25	14	EPS support systems	Priority at ECN 3
	total EPS bus A:	447	202		
B	MPF (ECN 3 K7A)	115	52	Under-pressure	
	Covidien (Covidien 1 B1D)	80	41	Under-pressure	Ex. Tyco
	Substation ECN 3 Main board K3C (NS)	215	199	i.e. Under-pressure JGL	
	AV-board EPS	25	14	EPS support systems	Priority at ECN
	total EPS bus B:	435	306		
C	HFR (ECN 2 K2R)	10	10	NV 2	
	EPS bus A	447	188+14	See bus A	in of failure generators A en B
	total EPS bus C:	**457**	**212**		

A.4 HFR buildings: elevation & flooding effects

Description	Room/building	Description	Elevation +NAP	Effect	Consequence
Reactor Outbuildings	Connection duct	Duct	5.00	Water on the roof	Water in cellars
Reactor Outbuildings	Hall	Reactor hall	7.00	Pushes door open	Loss of containment
Reactor Outbuildings	Emergency exit	Emergency exit	7.00	Corricor floats	Loss of containment
Reactor Outbuildings	Corridor	Corridor transfer hall	7.00	Corricor floats	Loss of containment
Reactor Outbuildings	Corridor BNCT	Corridor to BNCT	7.00	Corricor floats	Loss of containment
Reactor Outbuildings	Transfer hall	Building 102.45	5.00	Hall floods	None
Reactor Outbuildings	Cellar	Cellars	5.25	Cellar floods	Station black out (SBO2)
Ventilation Building	Ground	Inlet duct	5.00	Inlet blocked	No ventilation
Primairy Pump Building	Omvormer	Converter	5.25	Short circuit of converter	No emergency power
Primairy Pump Building	Noodkoeling	Battery clamps	5.40	Short circuit of batteries	No emergency power
Primairy Pump Building	Trafo	Trafo 1 - 4	6.00	Short circuit	Station black out (SBO2)
Secundairy Pump Building	Ground	Trafo	5.75	Short circuit	No secondary cooling

A.5 Other buildings: elevation & flooding effects

Description	Room/building	Description	Elevation +NAP	Effect	Consequence
HCL	Cellar	Pluggennesten	3.90	Ventilation blocks	No ventilation pluggennesten
HCL	Cellar	Storage pool	3.90	More water in pool	None
HCL	Ground	Hot cells en pluggennesten	3.90	Pluggennesten floods	None
HCL	Ground	Storage pool filterbussen	3.90	Storage pool floods	None
HCL	Ground	Hot cells and pluggennesten	4.10	Short circuit	No under pressure
HCL	Ground	Hot cells and pluggennesten	5.90	Ventilation blocks	No ventilation
MPF	Cellar	Liquid waste tanks	3.90	Tanks float	Release of contents
MPF	Ground	Equipment	4.90	Water in hot cells	Release of contents
MPF	Floor 1	Technical area	6.90	Short circuit	No under pressure
MPF	Floor 1	Technical area	6.90	Emergency power floods	No under pressure
Emercency Power Building	Ground	Diesels	4.10	Water in air inlet	No diesel
Emercency Power Building	Ground	Diesels	4.30	Short circuit	No diesel
Drinkwaterpompst.	Ground	Drinking water	7.10	Water in drinking water	None
Water treatment	Sump	Waste water	7.85	Sump floods	Release of contents
Water treatment	Cellar	Basins	7.85	Basins flood	No contamination
Water treatment	Sink	Tanks	7.85	Tanks float	Release of contents
Water treatment	Ground	Equipment	8.35	Short circuit	No under pressure
DWT	Cellar	Filter resins	7.85	Filters storage floods	No release
DWT	Ground	Empty	8.85	Short circuit	No under pressure
Pipe Cleaning	Pipe Cleaning	Pipe Cleaning	9.35	Pipes flood	None
Pipe Cleaning	Centrifuge	Centrifuge	8.85	Tank under centrifuge floods	Release of contents
WSF	Cellar	High active waste	7.85	Tanks submerge	Remain subcritical
WSF	Ground	Drums	7.85	Drums submerge	Release if drums leak
WSF	Ground	Drums	8.35	Short circuit	No under pressure
STEK	Cellar	Empty	6.66	Floods	None
STEK	Ground	Storage	6.66	Storage containers submerge	None
STEK	Ground	Storage	7.16	Short circuit	No ventilation
LFR	Sink	Fuel storage	6.66	Fuel storage floods	None
LFR	Sink	Fuel storage	6.66	Storage drums submerge	Remain subcritical
LFR	Ground	LFR	7.16	Short circuit	No under pressure
Alg. Lab.	Cellar	Telecommunication	5.85	Floods	No communication
Forum	Em. Coord. Cent	ECC	5.50	Inaccessible	No communication and coordination
Forum	Cellar	Electricity	5.50	Short circuit	No electricity
JRC	Em. Coord. Cent	ECC	4.40	Inaccessible	No communication and coordination
Jaap Goedkoop Lab	Cellar 1	Communication	3.90	Floods	No communication
Jaap Goedkoop Lab	Cellar 1	Power labs	3.90	Short circuit	No nuclear consequences
Jaap Goedkoop Lab	Cellar 2	Waste tanks	3.90	Tanks float	Release of contents
Jaap Goedkoop Lab	Floor 2	Technical area	9.90	Floods	No under pressure

A.6 Safety functions and SSCs of the facilities

HFR:

- Control of reactivity: a reactor core and spent fuel are present. Main systems or components are the control rods, the control rod drive mechanisms and activation by the RPS. Spent fuel is kept in a safe subcritical configuration by design and procedures;
- Fuel cooling: a reactor core and spent fuel are present. Main cooling systems are the primary coolant system, the decay heat removal system and pool cooling system;
- Confinement: several, e.g. the barrier concept.

HCL:

- Control of reactivity: spent fissile material is kept in a safe subcritical configuration by design and procedures;
- Confinement: a significant amount of the radioactive materials is present. Main systems are: matrix and cladding (to a limited extent), UCW filter-container, pool, hot-cells, ventilation, HCL containment;
- Pools in which fuel containing experiments are stored only have a shielding function and not a cooling function.

WSF:

- Control of reactivity: spent fuel and other fissile material are kept in a safe subcritical configuration by design and procedures;
- Confinement: a significant amount of radioactive materials and a limited amount of spent fuel are present. Main safety systems are UCW filter-containment, pipes and trenches.

LFR:

- Confinement: an amount of the radioactive materials is present;
- The reactor core and its spent fuel have been brought in a permanent shutdown configuration. In addition, the near term relocation of the spent fuel is being planned. Due to the status of the facility neither the control of reactivity and the heat removal function are not relevant safety functions.

JGL:

- Confinement: a very limited amount of radioactive materials is present;

- Main safety systems include ventilation system and glove boxes.

DWT:

- Confinement: a very limited amount of radioactive materials is present.

STEK:

- Confinement: the facility is used (only occasionally) for short-term storage of radioactive waste.

A.7 Wind speeds in Bft, m/s and km/h with typical land conditions

Beaufort number	Speed [m/s]	Speed [km/h]	Land conditions
0	0 – 0.2	0 - 1	Calm. Smoke rises vertically.
1	0.3 – 1.5	1 - 5	Smoke drift indicates wind direction and wind vanes are not moving.
2	1.6 – 3.3	6 - 11	Wind felt on exposed skin. Leaves rustle and wind vanes begin to move.
3	3.4 – 5.4	12 - 19	Leaves and small twigs constantly moving, light flags extended.
4	5.5 – 7.9	20 - 28	Dust and loose paper lifting. Small branches begin to move.
5	8.0 – 10.7	29 - 38	Branches of a moderate size move. Small trees in leaf begin to sway.
6	10.8 – 13.8	39 - 49	Large branches in motion. Whistling heard in overhead wires. Umbrella use becomes difficult. Empty plastic rubbish bins tip over.
7	13.9 – 17.1	50 - 61	Whole trees in motion. Effort is needed to walk against the wind.
8	17.2 – 20.7	62 - 74	Some twigs broken from trees. Cars veer on road. Progress on foot is seriously impeded.
9	20.8 – 24.4	75 - 88	Some branches break off trees, and some small trees blow over. Construction/temporary signs and barricades blow over.
10	24.5 – 28.4	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	28.5 – 32.6	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	> 32.7	>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.

A.8 Building standards

Facility code.	Facility name	Building name	Lightning protection standard	Wind resistance standard	Roof load standard
HCL	Hot Cell Laboratory	LSO	NEN 1014, LP3	TGB 1955 (TGB 1972)	TGB 1955
WSF	Waste Storage Facility	Radioactive waste building	NEN 1014, LP3	TGB 1955 **)	TGB 1955 **)
JGL	Jaap Goedkoop Laboratory	Jaap Goedkoop laboratory	NEN 1014, LP3	TGB 1955 **)	TGB 1955 **)
STEK	STEK hall	STEK hall	NEN 1014, LP3	TGB 1955	TGB 1955
HFR	High Flux Reactor	Reactor building	NEN 1014, LP2	n.a.	n.a.
		Reactor outbuilding	NEN 1014, LP2	TGB 1955	TGB 1955
		Primary pump building	NEN 1014, LP2	TGB 1955 **)	TGB 1955 **)
		Storage tanks	none	TGB 1955 **)	TGB 1955 **)
		Storage building	n.a. (NEN 1014, LP3)	n.a. (TGB 1955 **)	n.a. (TGB 1955 **)
		Secondary pump building	NEN 1014, LP2	TGB 1955	TGB 1955
		Chlorine bleach building	n.a. (NEN 1014, LP2)	n.a. (TGB 1955 **)	n.a. (TGB 1955 **)
		Air treatment building	NEN 1014, LP2	TGB 1955	TGB 1955
		Security lodge	n.a. (NEN 1014, LP2)	n.a.	n.a.
		NDE building	n.a. (NEN 1014, LP3)	n.a.	n.a.
LFR	Low Flux Reactor	Fermi building	NEN 1014, LP3	TGB 1955 **)	TGB 1955 **)
DWT	Decontamination and Waste Treatment	Decontamination building	NEN 1014, LP3	TGB 1955	TGB 1955
		Water treatment building	NEN 1014, LP3	TGB 1955	TGB 1955
		Waste treatment and recycling building	NEN 1014, LP3	TGB 1955 **)	TGB 1955 **)
ECC	Emergency Coordination Centre	Forum	NEN 1014, LP3	TGB 1955	TGB 1955
		Joint Research Centre (JRC)	NEN 1014, LP3	TGB 1955 **)	TGB 1955 **)
		General laboratory	NEN 1014, LP3	TGB 1955 **)	TGB 1955 **)
EPS	Emergency Power Supply	Emergency power building	NEN 1014, LP3	TGB 1955 **)	TGB 1955 **)

**) Value adopted from LSO building. More recent building standard may apply for newer structures.

A.9 Heat dissipation by decay heat production of core and spent fuel

This annex includes the assessment of the decay heat removal form the core of the HFR and from the radioactive materials stored in the HFR pools. These radioactive materials include spent fuel and remains of experiments and of isotope production. For the remaining of this annex these stored radioactive materials will be referred to as “spent fuel”.

The emphasis in this assessment is put on produced heat, to be removed by heat up and evaporation of water and the timeframes for both processes.

It presents the premises, sequences under consideration, their assessment and the final results as used in Chapter 5 of this report (Loss of safety functions).

Next to this, an assessment is performed to find input data for the situation that water of the pool will be heated up and replaced by colder water before boiling of starts.

9.1 Inputs for the calculation

9.1.1. Geometry

The core in the reactor vessel is cooled by the primary cooling system (100 m^3 of water). In case of emergency (such as a loss of secondary cooling), cooling of the core can be achieved by pool water. There are two valves in the reactor vessel, one above and one below the core. Opening these valves allows pool water to enter the vessel and cool the core by natural convection.

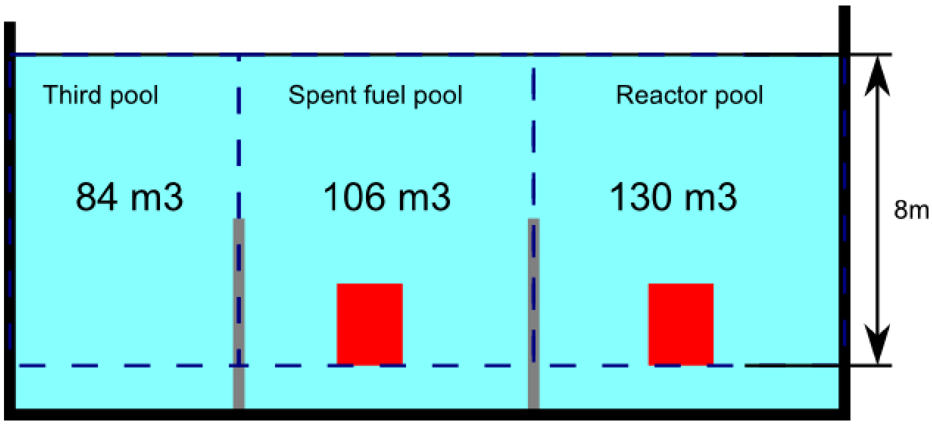


Figure 8-1: Simplified geometry of the HFR pools.

There are in total 3 pools: the reactor pool, a pool where the spent fuel is stored and a third pool. The pools are separated by doors that do not reach to the top of the pools, as shown in Figure 8-1. The volumes of these pools are calculated from the bottom of the reactor core, as water below heat-producing elements can stay stratified and subcooled. The area of the pools in total is 40 m^2 , the individual pools are 10.5, 13.25 and 16.25 m^2 respectively.

Figure 8-2 shows the volumes that have to be boiled off to reach the top of the fuel.

- When the water levels falls by 4 meters, a radiation shielding limit is reached.
- Once the water level is 4.62 meters below the original level, the pools become separated. From that point on the water level in the individual pools will fall a different rates, depending on the heat production within that pool.
- After the separation of the pools, the water level has to fall another 2.17 meters before the top of fuel is uncovered.

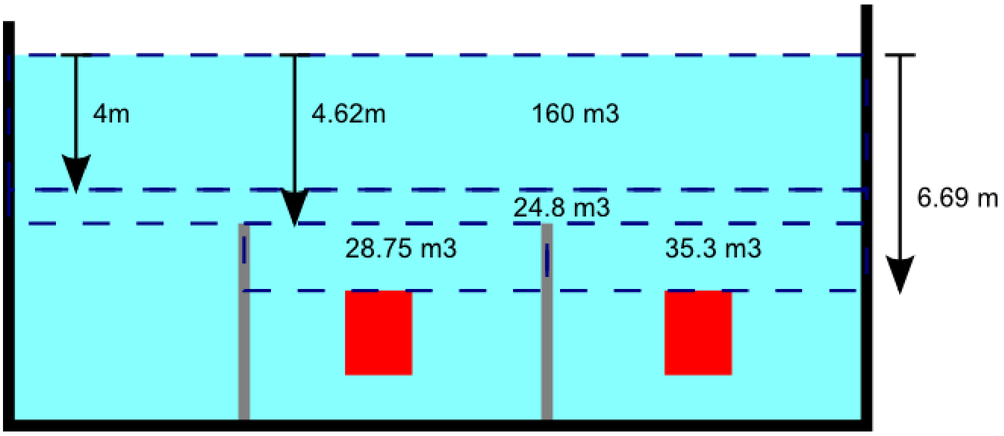


Figure 8-2: Volumes to be boiled off from the HFR pools.

9.1.2 Heat production by the core

For the first 200,000 seconds (55.5 hours) after reactor shutdown, the heat production of the core is taken from existing thermohydraulic analyses of the HFR using the RELAP code. RELAP implements the standard ANS-79 decay heat curve plus a contribution for fission power produced by delayed neutrons directly after shutdown.

Decay heat production beyond 200,000 seconds is approximated by an exponential model taken from Todreas and Kazimi:

$$P = 0.66P_0(T^{-0.2} - \left(T + T_p\right)^{-0.2})$$

Where T is the time (in seconds) since shutdown, T_p is the time the reactor has run at full power and P_0 the full power level of the reactor. The full power level is 51.5 MWth.

Two modifications have been made to achieve a closer fit with the curve from RELAP. Both modifications are conservative:

- The curve is scaled by a constant multiplier in order to match the RELAP curve at T= 200,000 seconds.
- T_p is taken as 1 year, although the fuel in the HFR stays in the core for at most 6 months. Using the higher value resulted in a better fit with the RELAP curve in the range from T=0 to T=200,000 seconds.

The resulting curves are shown in Figure 8-3 and Figure 8-4. Figure 8-4. shows the correspondence between both curves in the short run.

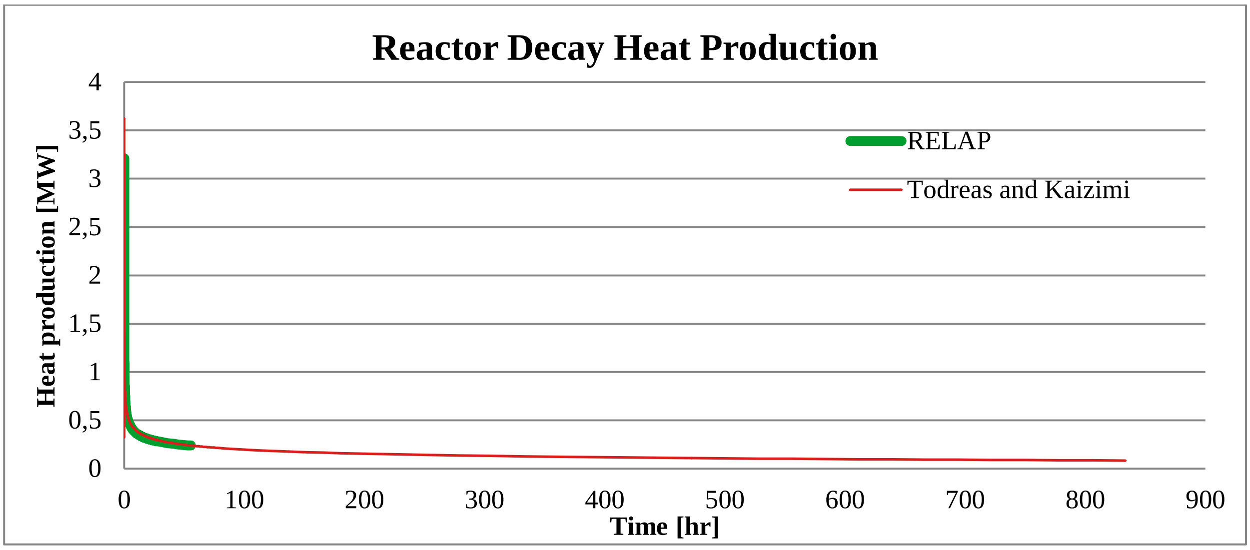


Figure 8-3: Reactor decay heat production.

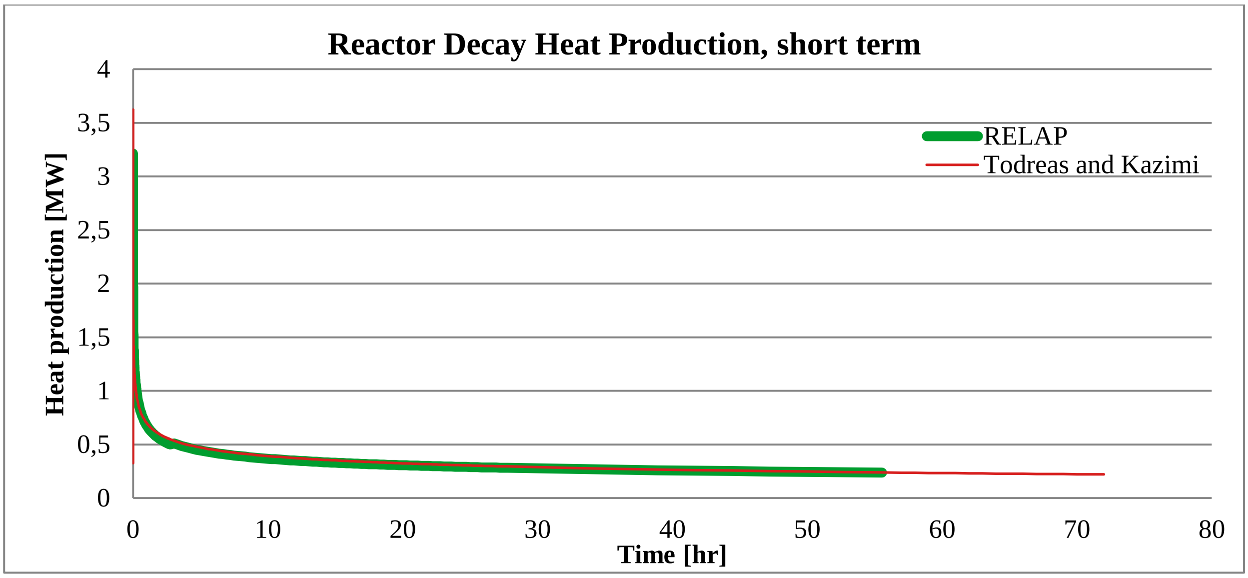


Figure 8-4: Reactor decay heat production, short term.

9.1.3 Heat production by the spent fuel

The spent fuel is assumed to produce a constant 100 kW of heat, based on the HFR Dokpak from 1992. The current amount of fuel stored in the pool is smaller than it was in 1992. This maximum capacity is determined by the LCO conditions indicating allowed maximum amounts of spent fuel and remains of experiments and isotope production.

When the reactor is not operating, the fuel elements of the core are placed in the spent fuel pool. This is treated as a separate case. The core will be placed in the spent fuel pool not earlier than 3 hours after reactor shutdown. Therefore, the calculations for this case assume that the loss of ultimate heat sink occurs 3 hours after reactor shutdown, with the core placed in the spent fuel pool. The heat production curve for this case is taken as a constant 100 kW plus the heat production curve of the core from 3 hours on. This heat production takes entirely place in the spent fuel pool.

9.1.4 Power scenarios

Energy can only be stored in the primary system if there is electric power available to circulate primary water through the core region. This leads to three different scenarios:

A. Power is available. In this scenario all water in the primary system will be heated to the threshold value.

B. The primary system runs on battery power for 2 hours. In this scenario, the primary system will absorb all core heat production in the first 2 hours, after which all heat gets deposited in the reactor pool.

C. No power is available. In this scenario, the primary system is not taken into account in the calculations

9.1.5 Pool isolation and pool mixing

In the main calculations, the pools are considered as a single well-mixed volume, heated by a single heat source. For the first period after the loss of ultimate heat sink, it is not certain that mixing takes place fast enough to ensure that the entire pool will reach saturation temperature at the same time. For this reason, it is possible that boiling starts early in some parts of the system, before the entire pool has reached saturation temperature.

To get a conservative estimate for the earliest onset of boiling, every scenario is also considered under the assumption that the pools are initially isolated from each other. Under this assumption, one pool will start to boil before the other pools, and this time can be considered as a conservative estimate of the earliest likely onset of boiling.

9.2 Results

9.2.1 Primary system power available

Assessment

The first sequence to consider is the **complete mixing situation** in which the reactor decay heat will be dissipated in the primary system until saturation occurs.

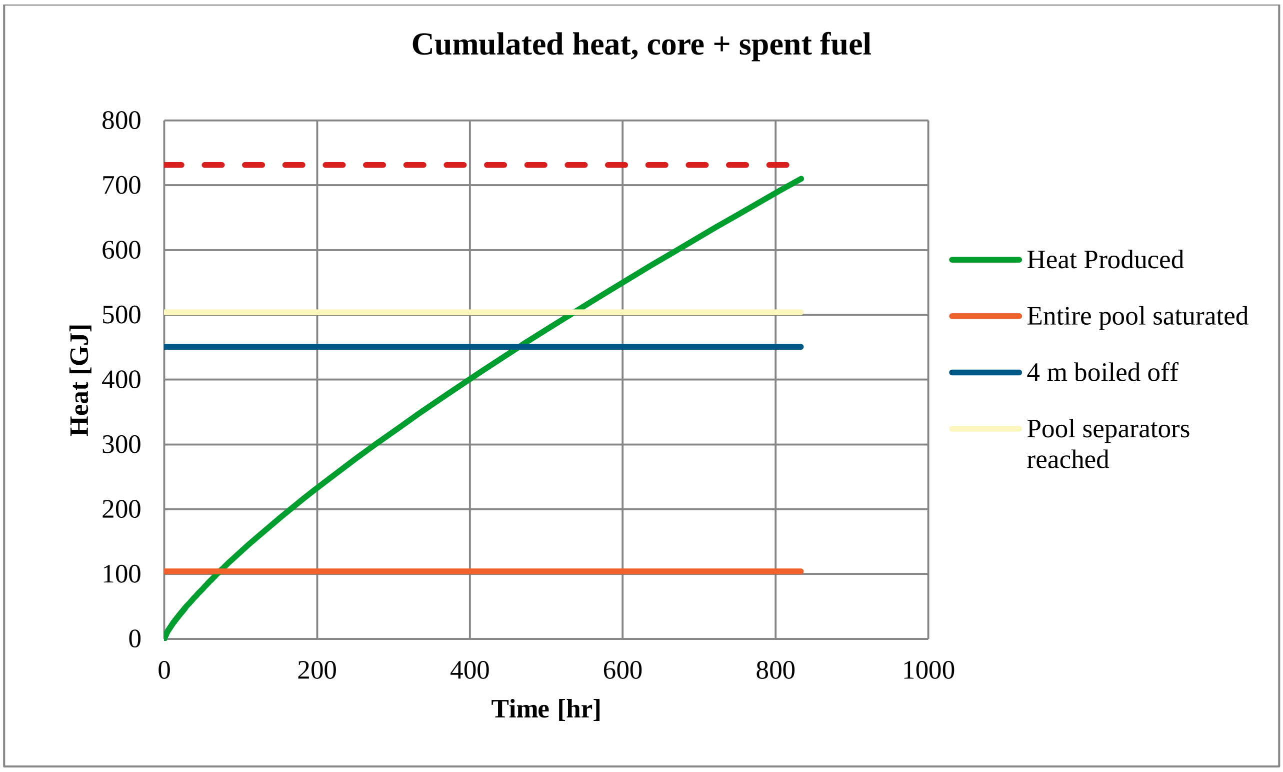


Figure 8-5: Total heat production, for the mixing situation.

Figure 8-5 shows the combined heat production of core and spent fuel together. First the primary system is heated by the decay heat of the core and becomes saturated after 12 hours. Then water of all pools is used to dissipate the heat of the core. In parallel the spent fuel heat up the pools as well. After 73 hours, the combined heat of spent fuel and core will cause the entire pool of 320 m^3 be saturated, because perfectly mixing of pools is assumed. This can be seen as a best estimate of the onset of boiling.

As the figure shows, the water level will drop by 4 meters after around 470 hours. After 50 hours the water level reaches the top of the pool doors. Now the pools are isolated and it is assumed that the heat of the spent fuel will only contribute to boiling in the spent fuel pool, and heat from the core will only contribute to the heating of the reactor pool. This is illustrated in Figure 8-6.

The top of the spent fuel will be reached first, after approx. 700 hours. The spent fuel pool is smaller than the reactor pool, while the spent fuel heat production is on this time scale comparable to the core heat production.

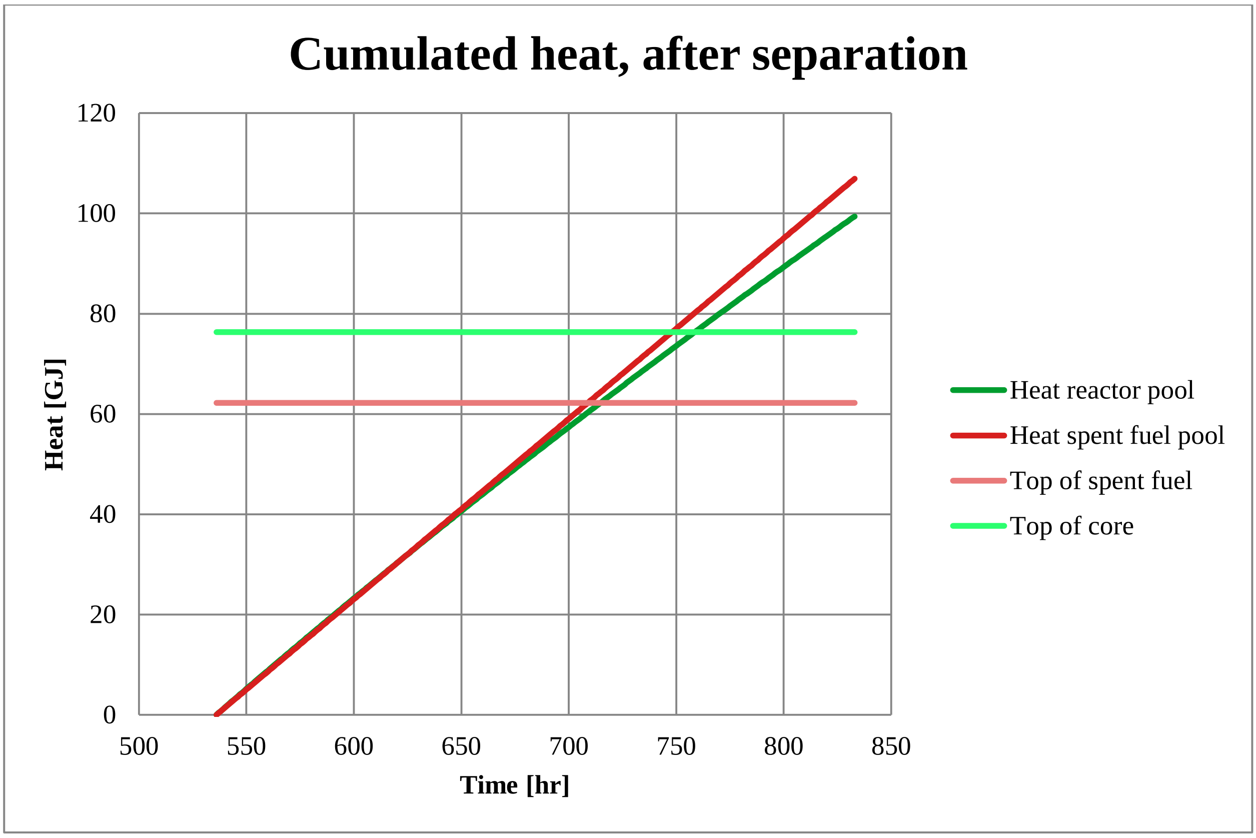


Figure 8-6: Heat production after separation of pools.

When power is available, extra water can be replenished from on-site reserve stock to the pools during boil-off. Figure 8-5 indicates the final level of heat that can be removed in case this extra water is supplied before the pool doors are reached. This results in an additional 330 hours of boil off of water.

The second sequence differs from the preceding one in that **complete isolation situation** is assumed. The primary system is heated by core heat and becomes saturated after 12 hours. This will be followed by heating up of the water of the reactor pool (130 m^3) only until boil off starts. In parallel the other pools will be heated up by the spent fuel, again until boil off starts. In this way the timing of boil off for both pools is determined, and the earliest of these can be considered as aconservative estimate for the onset of boiling in the entire pool. In this sequence the reactor pool boils off first after 47 hours.

The sequence is illustrated by Figure 8-7.

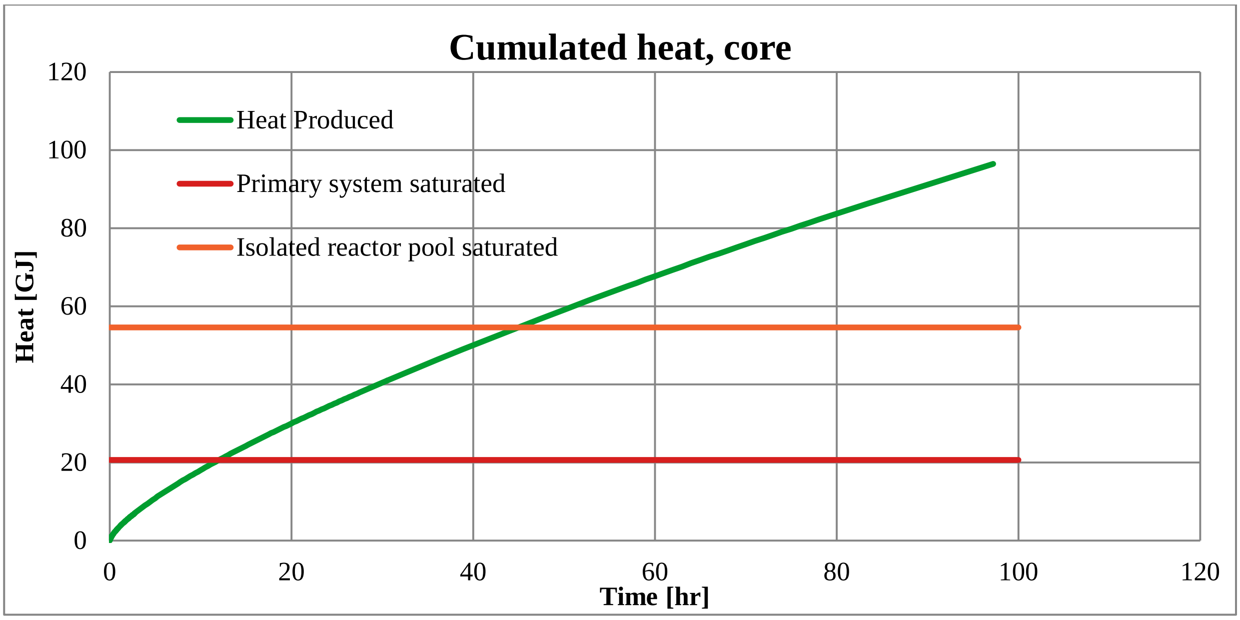


Figure 8-7: Core heat production, for the isolation situation.

Conclusion

These sequences and the results that can be derived from the figures Figure 8-5 to Figure 8-7 are summarized in Table 8-1.

In case power is available and maximal amount of heat can be dissipated in the primary system:

- Boil off will start after approx. 45 hours for the most conservative case and after 73 hours for the estimate based on the assumption of perfect mixing.
- At the 72 hour mark, water has been boiling off for about 1 day in the most conservative estimate. The amount of water boiled off at 72 hours is conservatively estimated at 10 m3, which would result in a 0.25 meter water level drop.
- The -4 m level will be reached after 464 hours.
- Fuel uncovery occurs after 709 hours in the spent fuel pool; more than one day later followed by core uncovery. This is due to the fact that at that time the heat output of both sources are almost equal, while the remaining amount of water in the reactor pool is larger.
- 330 hours of additional time are gained by supplying of extra water that already is stored on-site.

Step	Sequence	Mixing situation			Isolation situation		
		Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)	Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)
1	T rise of primary system 50 to 100 ˚C	100	21	12	100	21	12
2	Heating up reactor pool water from 37 ˚C to 100 ˚C	320[51]	84	73	130	34	45
3	Heating of remaining pools, starts in parallel with 1&2	320	84	73	190	50	137
4	Evaporation of 4 m of water from the pools	160	346	464			
5	Evaporation to separated pools	25	54	536			
6	Evaporation in separate pool, until first fuel uncovery	29	62	709			
A	*Extra time if water is injected*	*105*	*227*	*330 extra*			

Table 8-1: Sequence A summary

[51] For the mixing situation the pool water volumes are combined to only one volume of 320 m^3 heated up by the combined sources (so step 2 and 3 are merged to one step)

9.2.2 Two hours of power available for primary system pumps

Assessment

Compared to the preceding sequences, this paragraph deals with the complete mixing and isolation situations, while the primary system only absorbs heat produced by the core during the first 2 hours after start of the event. Both sequences are illustrated by Figure 8-8 to Figure 8-10.

The difference between the preceding mixing sequence is that the primary system is heated up by the decay heat of the core for only 2 hours instead of 12 hours. So the primary system does not becomes saturated. For the remainder the sequence is the same.

- Fuel uncovery occurs after 709 hours in the spent fuel pool; more than one day later followed by core uncovery. This is due to the fact that at that time the heat output of both sources are almost equal, while the remaining amount of water in the reactor pool is larger.

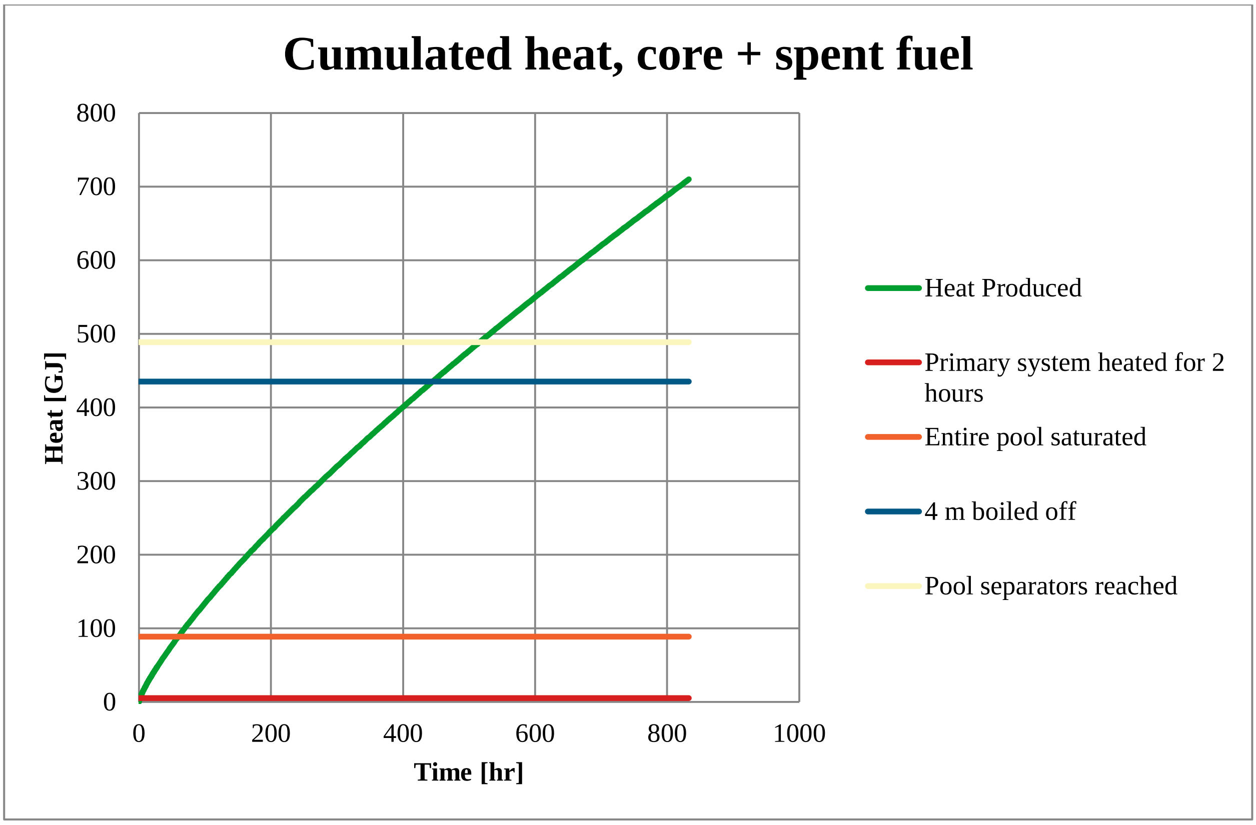


Figure 8-8: Total heat production, for the mixing situation

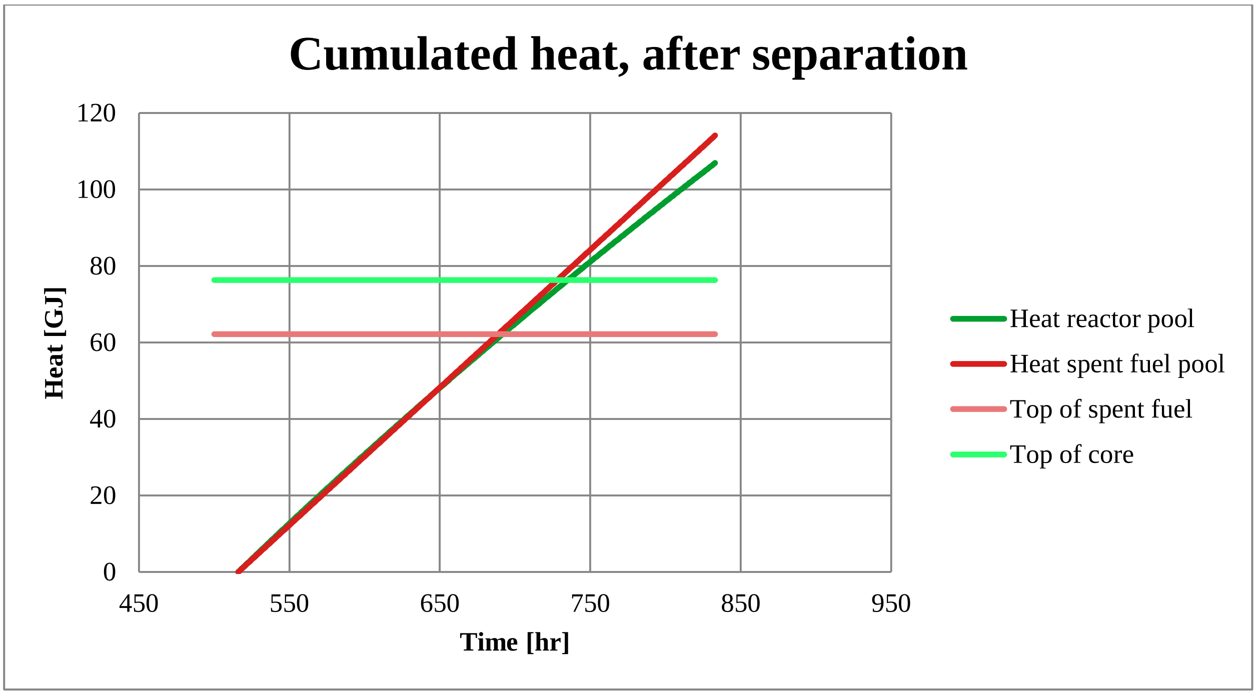


Figure 8-9: Heat production after separation of pools.

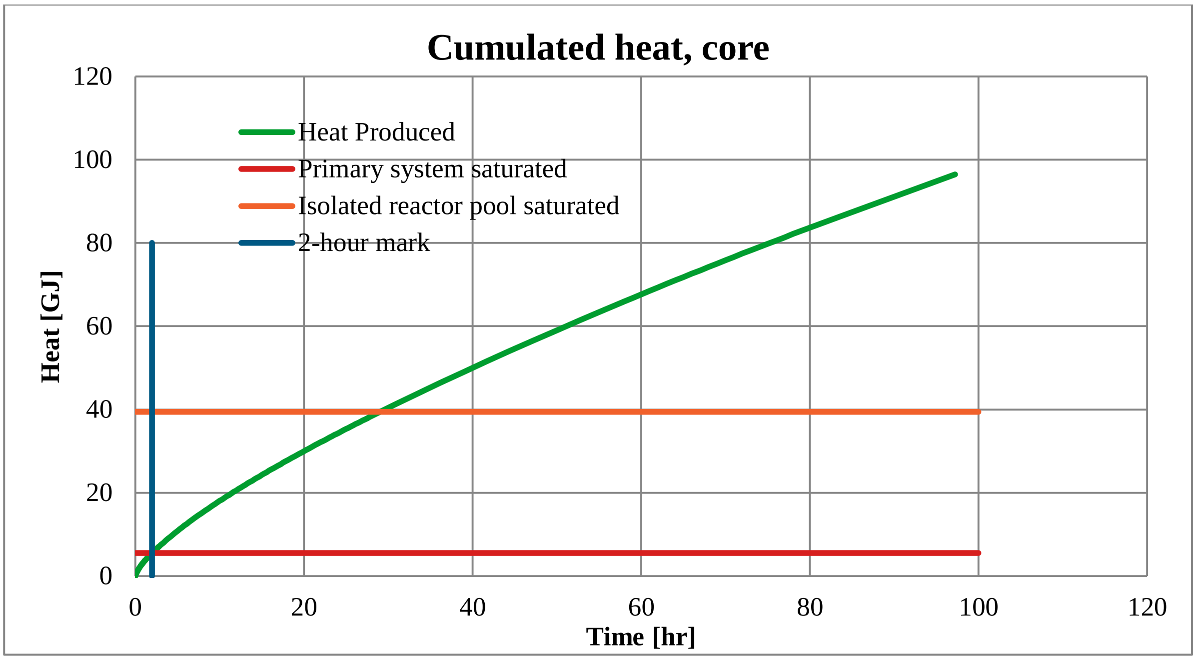


Figure 8-10: Core heat production, for the isolation situation.

In the isolated situation the reactor pool boils off first after 31 hours. The sequence is illustrated by Figure 8-10.

Conclusion

These sequences and the results that can be derived from Figure 8-8 to Figure 8-10 are summarized in Table 8-2.

Step	**Sequence**	**Mixing situation**			**Isolation situation**		
		Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)	Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)
1	2 hours of core heat to primary system	100	5.5	2	100	5.5	2
2	Heating up reactor pool water from 37 ˚C to 100 ˚C	320	84	60	130	34	31
3	Heating of remaining pools, starts in parallel with 1&2	320	84	60	190	50	137
4	Evaporation of 4 m of water from the pools	160	346	444			
5	Evaporation to separated pools	25	54	516			
6	Evaporation in separate pool, until first fuel uncovery	29	62	689			
A	*Extra time if water is injected*	*105*	*227*	*330*			

Table 8-2: Sequence B summary

From this is can be concluded that in case power is available during the first 2 hours of the events and a limited amount of heat can be dissipated in the primary system:

- Boil-off will start after approx. 31 hours for the most conservative case and after 60 hours for the best estimate.
- At the 72-hour mark, water has been boiling off for about (72-31=)41 hours in the most conservative estimate, and for (72-60=)12 hours under the well-mixed assumption. The amount of water boiled off at 72 hours is 17 m3 for the conservative estimate, resulting in a water level drop of about 0.4 m.
- The -4 m level will be reached after 444 hours. This is nearly one day shorter than the former situation (12 hours heat up of the primary system)
- Fuel uncovery occurs after 689 hours in the spent fuel pool; approx. one day later followed by core uncovery. This is due to the fact that at that time the heat output of both sources are almost equal, while the remaining amount of water in the reactor pool is larger.
- Compared to the former situation (12 hours heat up of the primary system) it can be indicated that the long run events occur about one day earlier in this situation.
- Again 330 hours of additional time is gained by supply of extra water that is already stored on-site.

9.2.3 No power available for primary system pumps

Assessment

In these sequences, complete station black out is assumed. This means that no significant heat can be dissipated into the primary system; so only heat up and evaporation of the water of the pools will be taken into account.

Both sequences are illustrated by Figure 8-11 to Figure 8-13. These figures show the same sequence as Figure 8-5 to Figure 8-7, but without heating of the primary system. In this scenario, all events are reached around 30 hours earlier, compared to the scenario with power available.

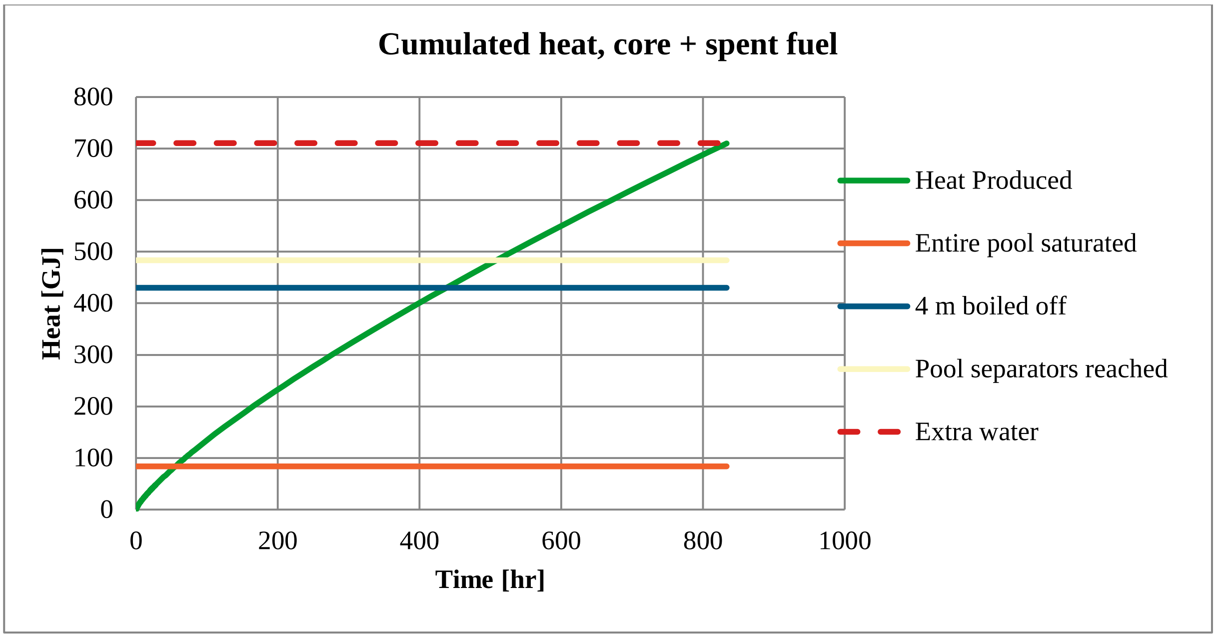


Figure 8-11: Total heat production, for the mixing situation.

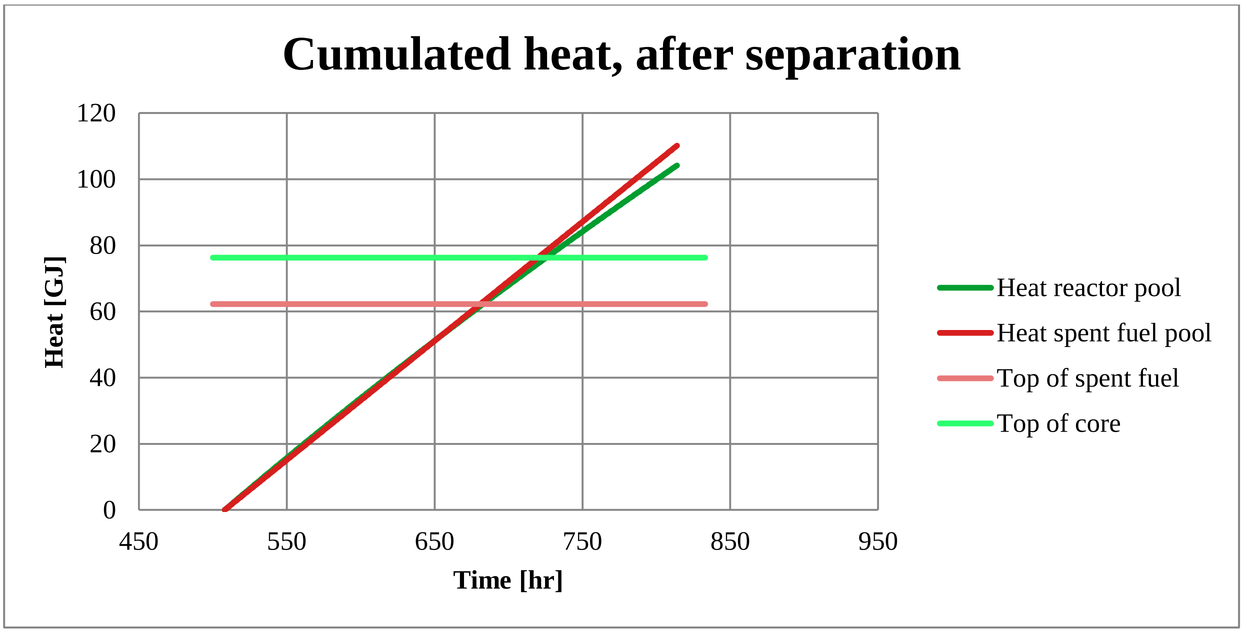


Figure 8-12: Heat production after separation of pools.

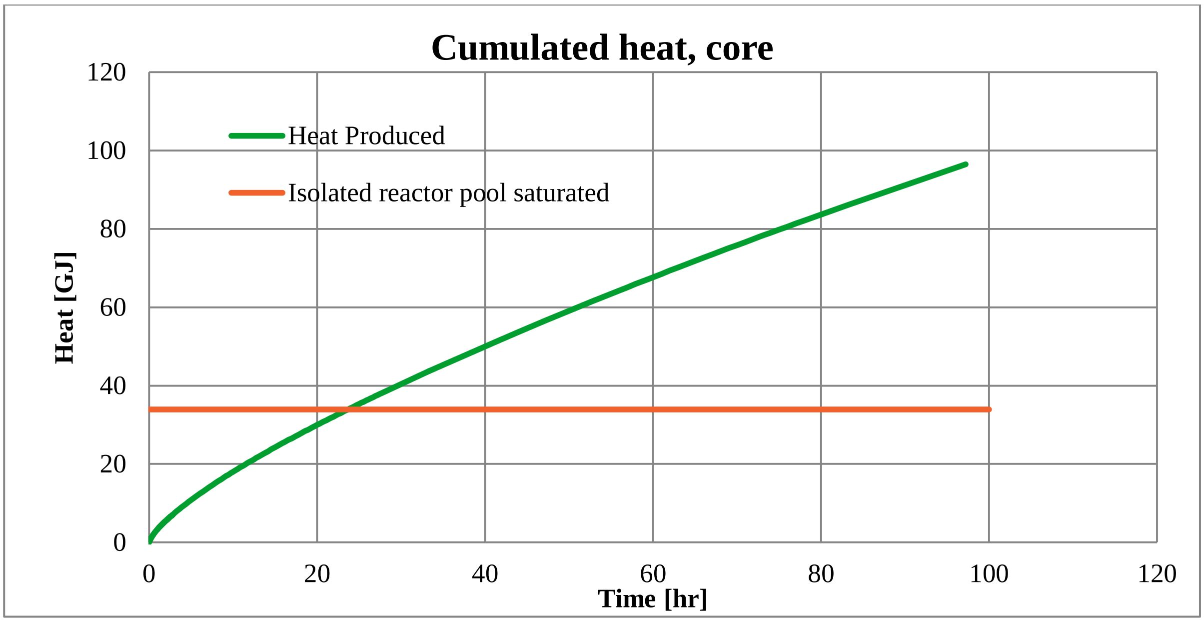


Figure 8-13: Core heat production, for the isolated situation.

In the isolated situation the reactor pool boils off first after 24 hours. The sequence is illustrated in Figure 8-13.

Conclusion

These sequences and the results that can be derived from Figure 8-11 to Figure 8-13 are summarized in Table 8-3.

Step	**Sequence**	**Mixing situation**			**Isolation situation**		
		Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)	Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)
1	Heating up reactor pool water from 37 ˚C to 100 ˚C	320	84	56	130	34	24
2	Heating of remaining pools, starts in parallel with 1	320	84	56	190	50	137
3	Evaporation of 4 m of water from the pools	160	346	437			
4	Evaporation to separated pools	25	54	508			
5	Evaporation in separate pool, until first fuel uncovery	29	62	681			
A	*Extra time if water is injected*	*105*	*227*	*330*			

Table 8-3: Sequence C summary

From this is can be concluded that in case no power is available, so no heat is dissipated in the primary system:

- Boil-off will start after approx. 24 hours for the most conservative case and after 56 hours for the well-mixed estimate.
- At the 72-hour mark, water has been boiling off for two days (72-24 hours) in the most conservative estimate, and for 16 (72-56) hours under the well-mixed assumption. The amount of water boiled off at 72 hours is 20 m3 in the conservative estimate, resulting in a water level drop of about 0.5 m.
- The -4 m level will be reached after 437 hours ; 2 hours of extra cooling by the primary system (sequence B) provides eventually a delay of events by about 7 hours.
- Fuel uncovery occurs after 681 hours in the spent fuel pool; more than one day later followed by core uncovery. This is due to the fact that at that time the heat output of both sources are almost equal, while the remaining amount of water in the reactor pool is larger. Compared to the situation that electrical power is available, it can be indicated that for this long run this rough estimate shows a difference of over one day.
- Again 330 hours additional time is gained by supply of extra water that already is stored on-site.

9.2.4 Fuel of core and spent fuel together in spent fuel basin, 3 hours after reactor shut down

Assessment

This sequence refers to the situation that the core is unloaded and stored at the earliest convenience in the storage pool. This means that, according to current practice, it is assumed that three hours after reactor shut down the core is placed in the storage pool. In addition to this it is assumed that the storage pool is loaded to its maximum capacity by other radioactive materials of which spent fuel is the larger part; all in conformity with the LCO conditions.

Figure 8-14 illustrates this sequence. Since all heat production in this sequence takes place in one pool, this single figure illustrates the same events as the sets of three graphs do in each of the previous sequences.

In particular, the graph shows both the moment that that the entire pool would be saturated under the assumption of complete mixing, and the moment that the spent fuel pool would be saturated under the assumption of isolation.

Conclusion

These sequences and the results that can be derived from the Figure 8-14 are summarized in Table 8-4.

From this is can be concluded that when only heat is produced in the storage pool:

- The spent fuel pool itself would be saturated after 13 hours, if no mixing with the other pools occurs. Saturation of the entire pool would occur after 59 hours.
- At the 72-hour mark, water has therefore been boiling off for about 60 hours in the most conservative estimate, and for 13 hours under the well-mixed assumption. The amount of water boiled off at 72 hours is 33 m3 for the conservative estimate, resulting in a water level drop of about 0.8 m.
- The 4 meter level occurs after 445 hours, and separation into separate pools after 517 hours. These are similar times as for the scenarios with the core still in the reactor.
- Since all heat production takes place in the spent fuel pool, the water level after separation will fall faster in this pool than in scenarios where the core is still in the reactor. After 603 hours, the top of the spent fuel will be reached. This is a difference of amply 3 days less than the sequence C situation when complete station black out is assumed.

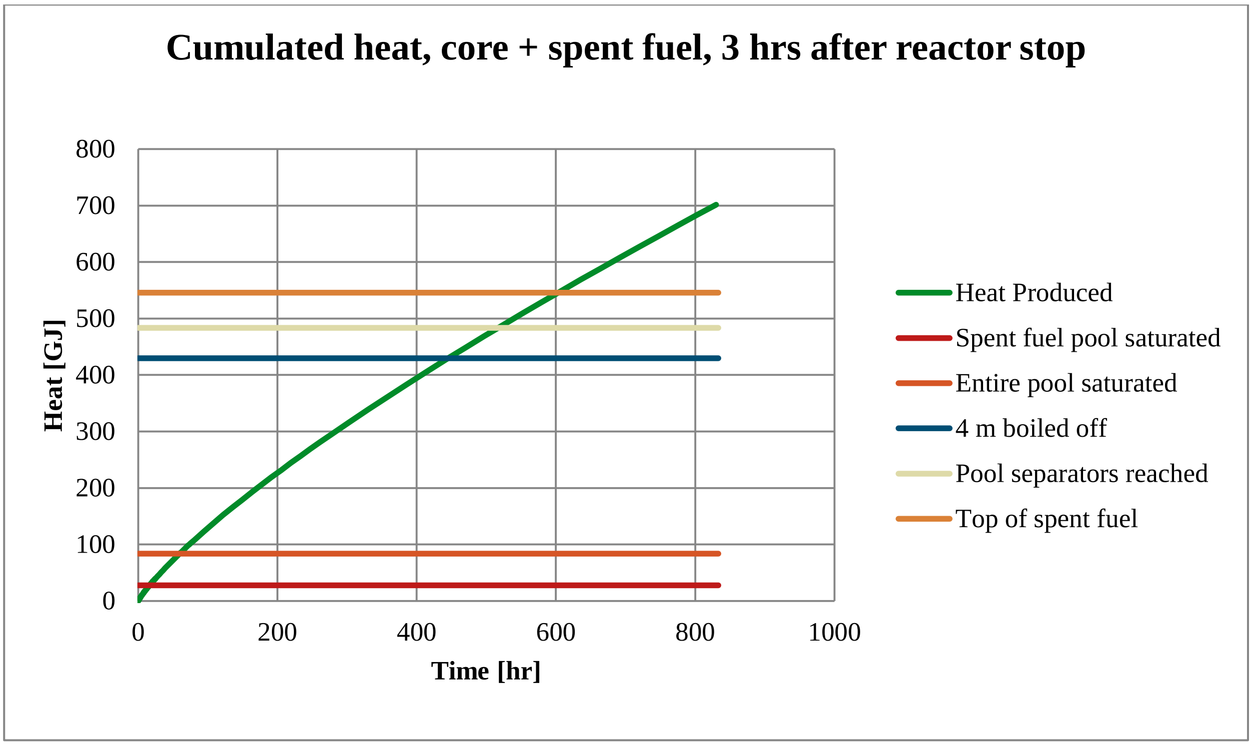


Figure 8-14: Heat production by stored core and spent fuel, 3 hours after reactor shut down.

Step	**Sequence**	**Mixing situation**[52]			**Isolated situation**		
		Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)	Volume involved (m^3)	Energy involved (GJ)	End time (hrs. since shutdown)
1	T rise of fuel pool from 37 to 100 ˚C	320	84	59	106	27.7	13
2	T rise of remaining pools from 37 to 100 ˚C	320	84	59	214	55.8	59
3	Evaporation of 4 m of water from the pools	160	346	445			
4	Evaporation to separated pools	25	54	517			
5	Evaporation in separate pool, until first fuel uncovery	29	62	603			
A	*Extra time if water is injected*	*105*	*227*	*330 extra*			

Table 8-4: Old core and spent fuel sequence summary

[52] For the mixing situation step 1 and 2 are merged

Heat removal by recirculation

Assessment

One possibility to remove heat from the core and spent fuel is the application of a procedure to discharge hot water (heated up by core and spent fuel) to the DWT facility before the onset of boiling. Replenishment of that amount of water can be performed by supply from several systems. Repetition of these steps might make it possible to alternatively cool down the reactor and spent fuel in case loss of ultimate heat sink occurs.

This paragraph assesses the amount of water needed to meet the requirement that boiling will be prevented.

For this assessment it is assumed that water of 37 ˚C is inserted and that water of 90 ˚C is discharged. Figure 8-15 shows the water circulation that is needed to keep the water temperature at 90 ˚C.

These flows are based on a similar calculation as for the boiling cases, for the indicated sequences A, B and C. Energy is absorbed by the entire pool until it reaches 90 ˚C. After that point, every m^3 of replaced water removes 0.21 GJ of energy from the system. At the rates shown, the replacement flow exactly balances the heat produced by core and spent fuel together.

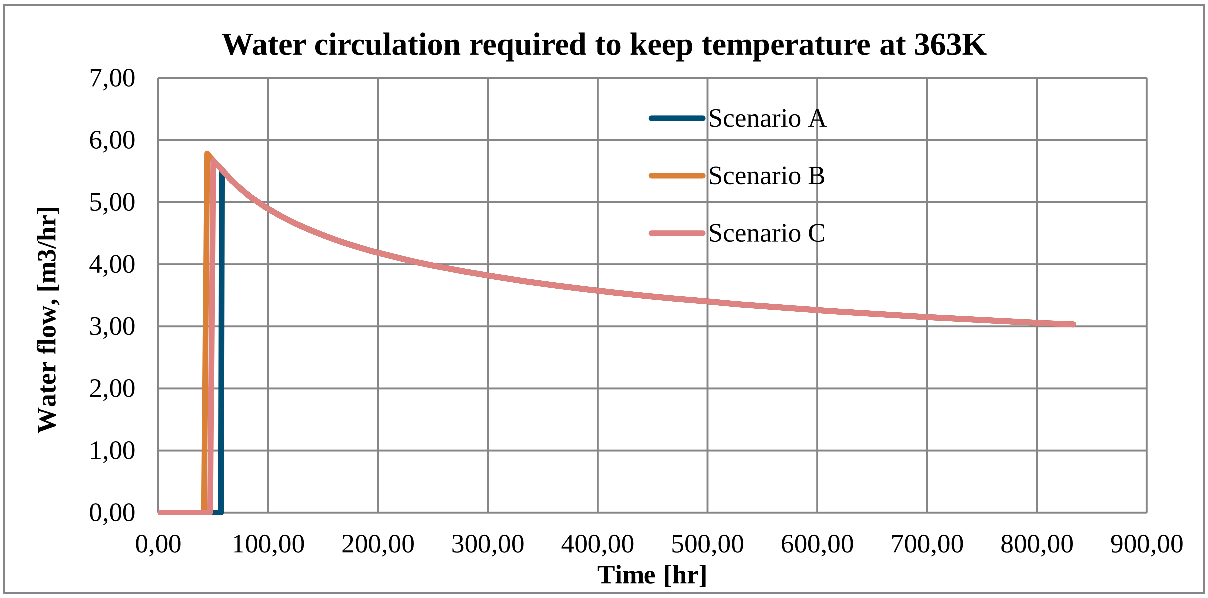


Figure 8-15: Minimal water flows.

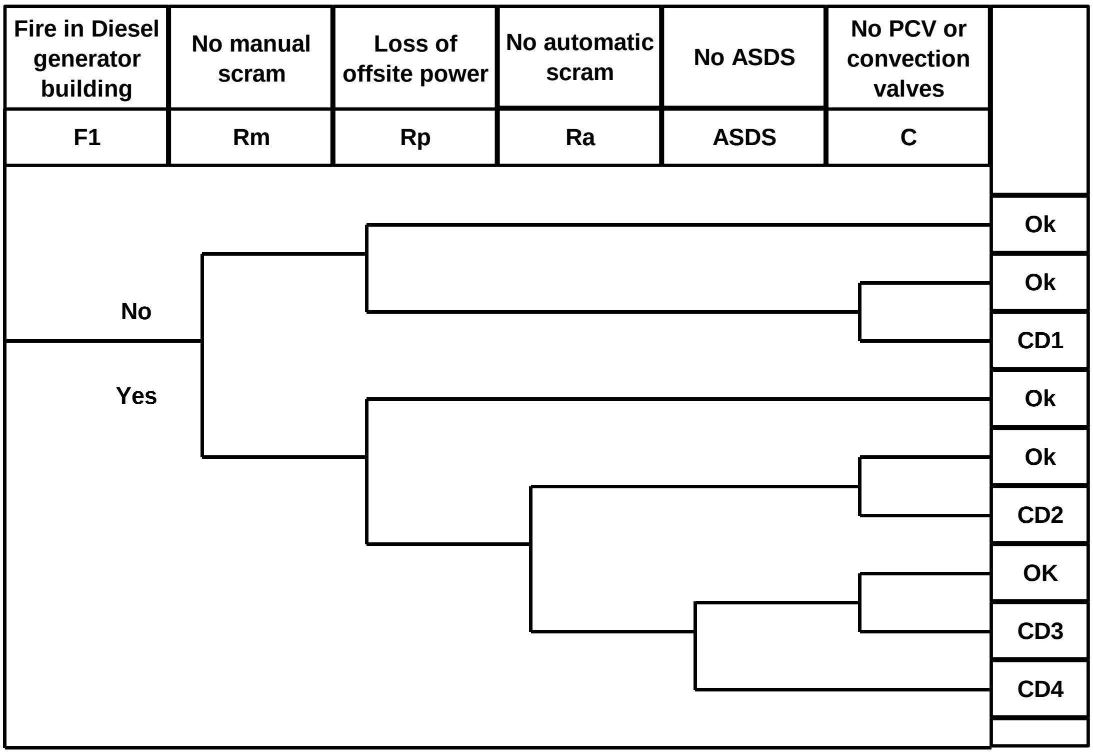
Conclusion

From this figure it can be concluded that replacement of water should start after approximately 60 hours. To keep to temperature at 90 ˚C the initial flow has to be about 6 m^3/h, which can be reduced to 3 m^3/h in the very long run.

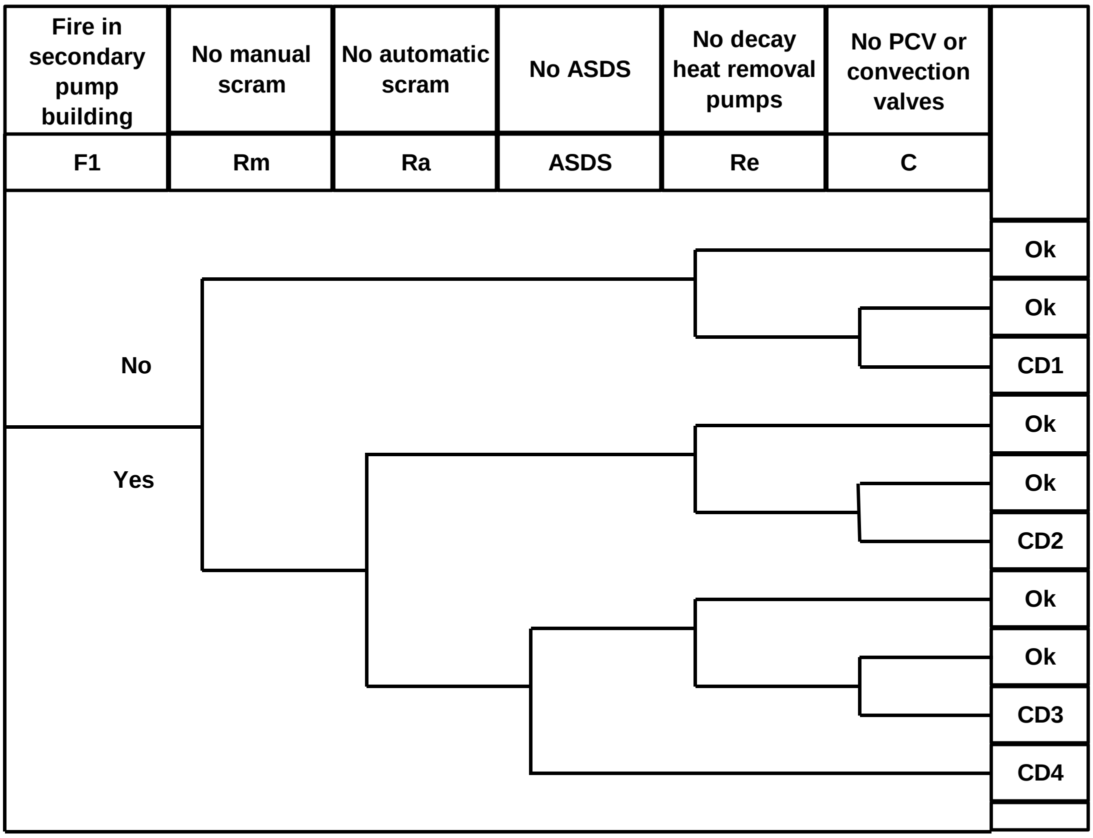
A.10 Fire events leading to HFR core damage

From the event trees derived in the fire analysis on the HFR, the combinations of fires and failures that lead to core damage can be derived:

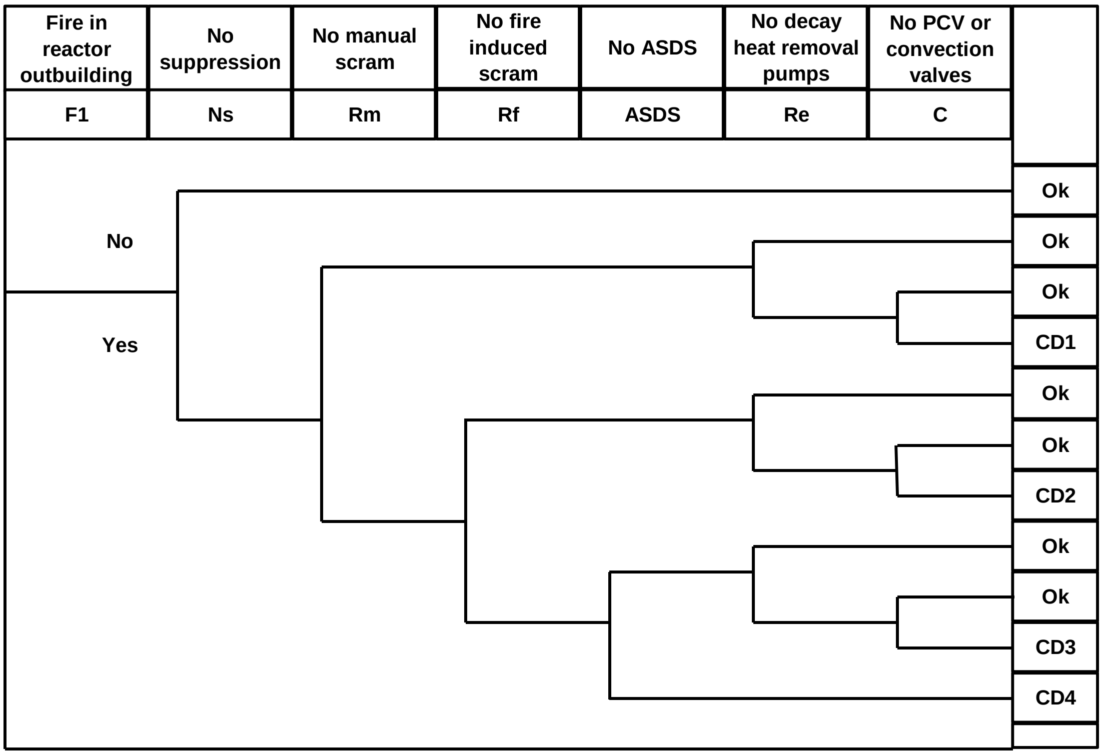
Fire in the Emergency Power Building or Air Treatment Building:



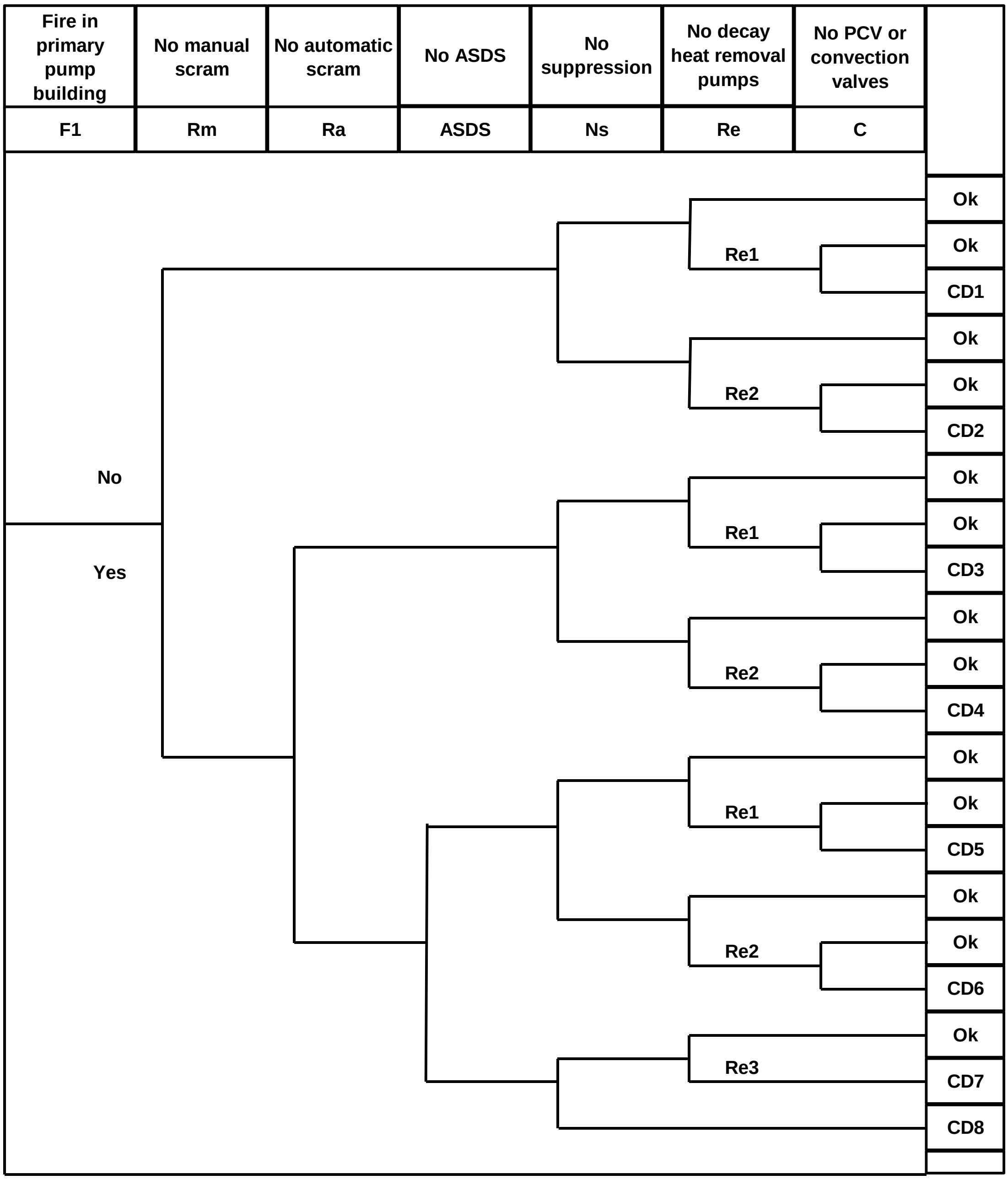
Fire in the Secondary Pump building:



Fire in the Reactor Outbuilding:



Fire in the Primary Pump Building:

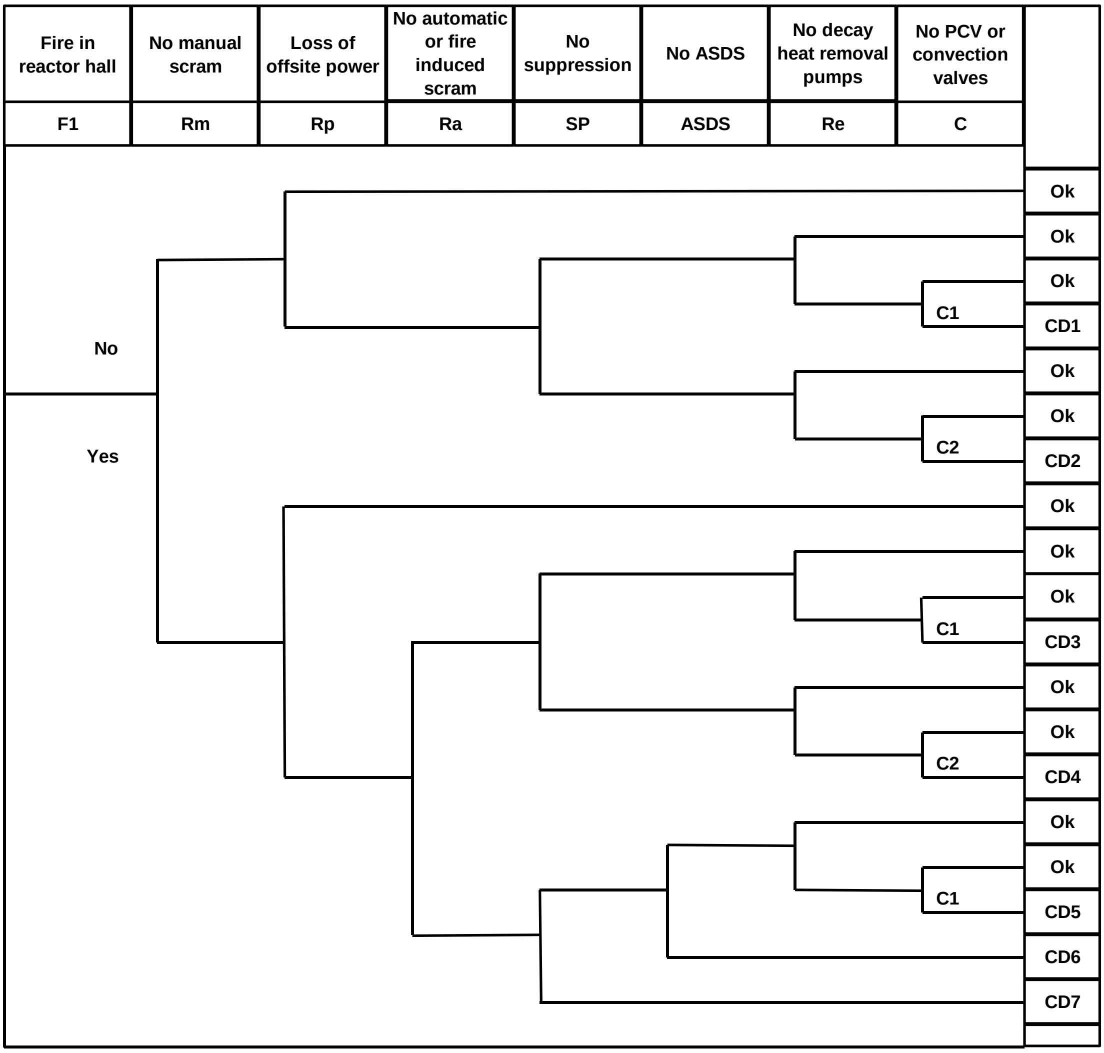


Re1 = Current situation

Re2 = Both decay heat removal pumps separated from G01 area

Re3 = Primary pumps continue running after suppression.

Fire in the reactor hall:



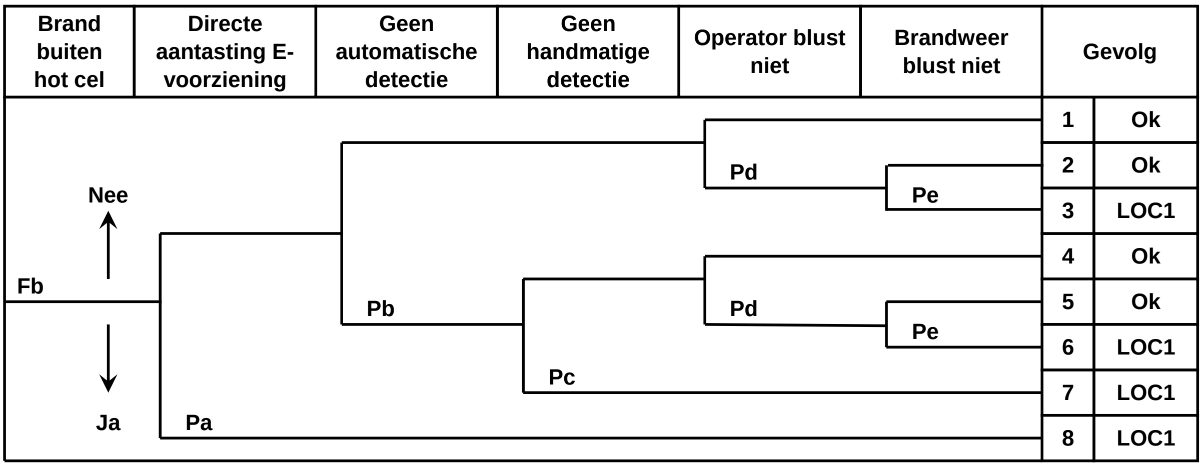
C1 = Current fire protection hydraulic actuation system pool water injection valves

C2 = Fire resistant actuation system pool water injection valves.

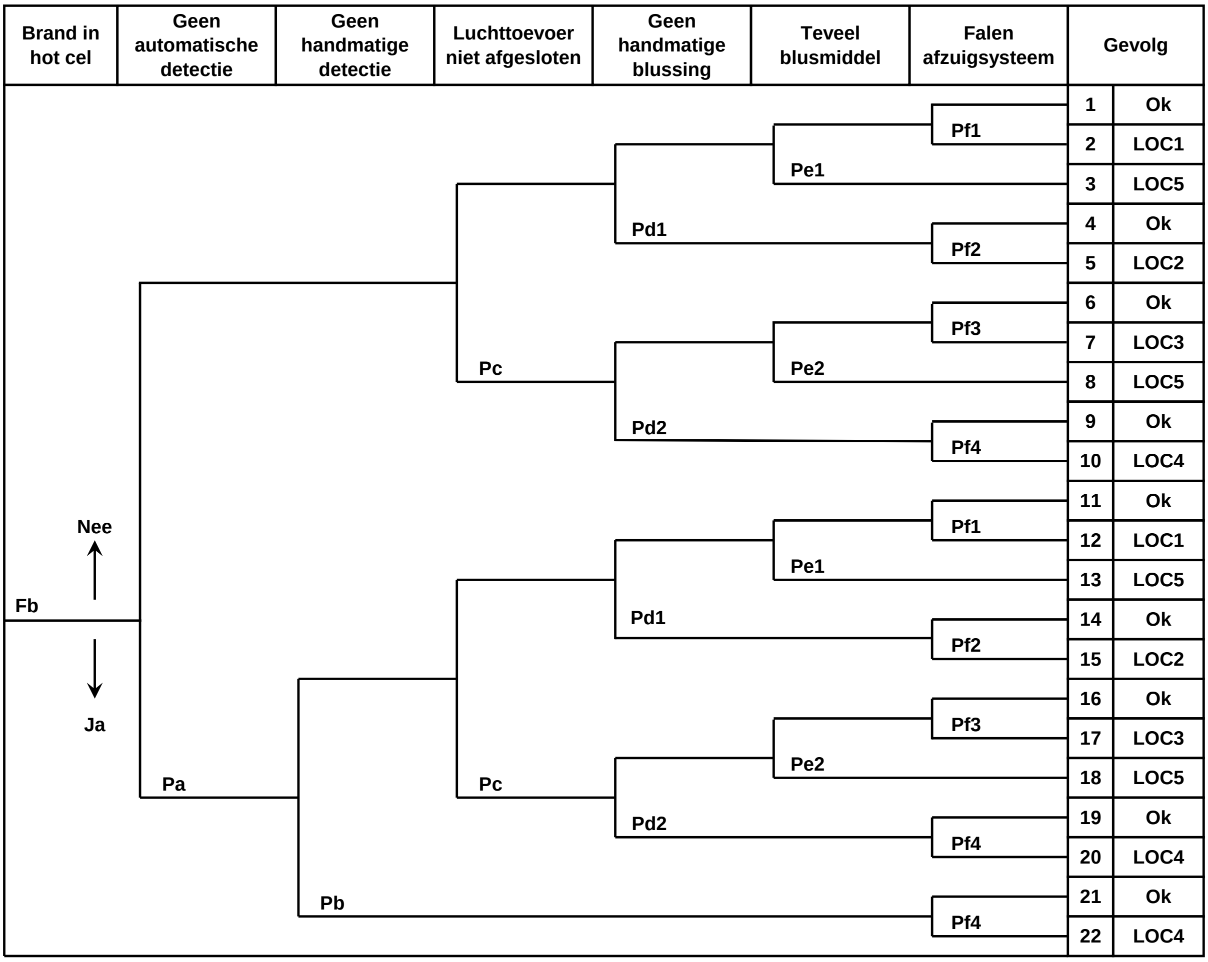
A.11 Fire events leading to HCL loss of containment

From the event trees derived in the fire analysis on the HCL, the combinations of fires and failures that lead to loss of containment can be derived:

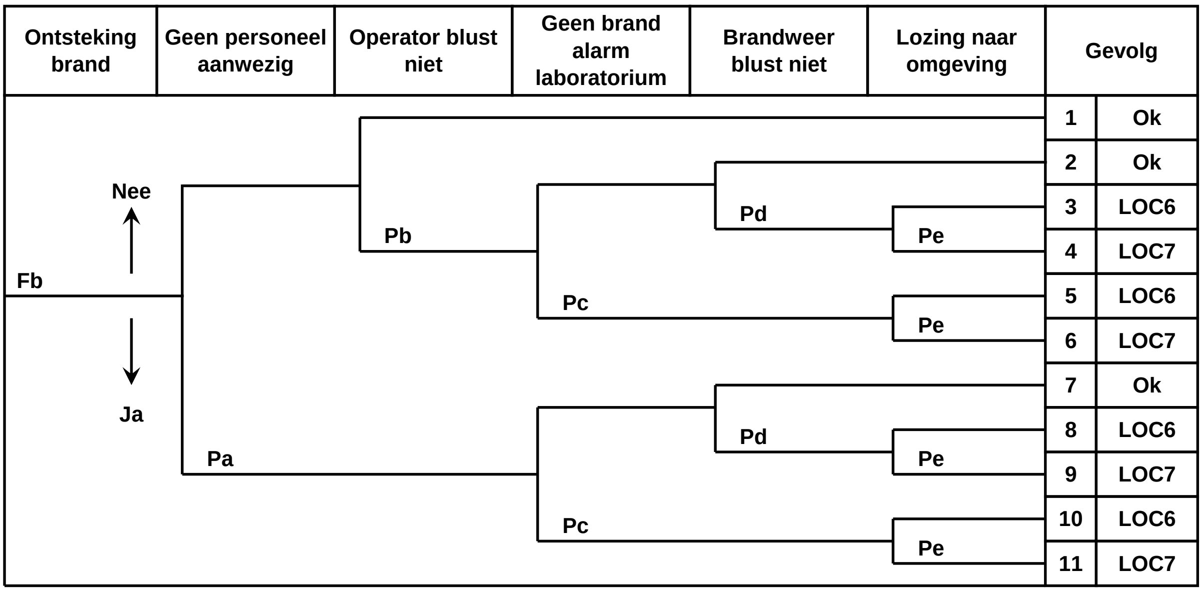
For a fire outside the hot cells:



For a fire in a hot cell (both NRG and Covidien):



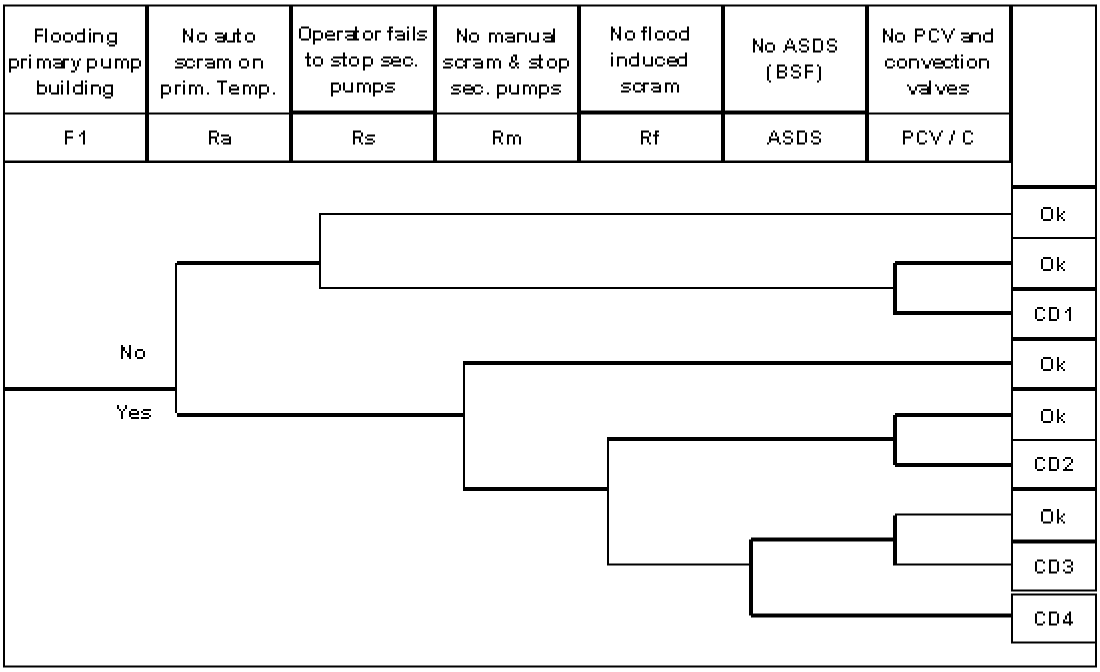
For a fire in the actinide lab:



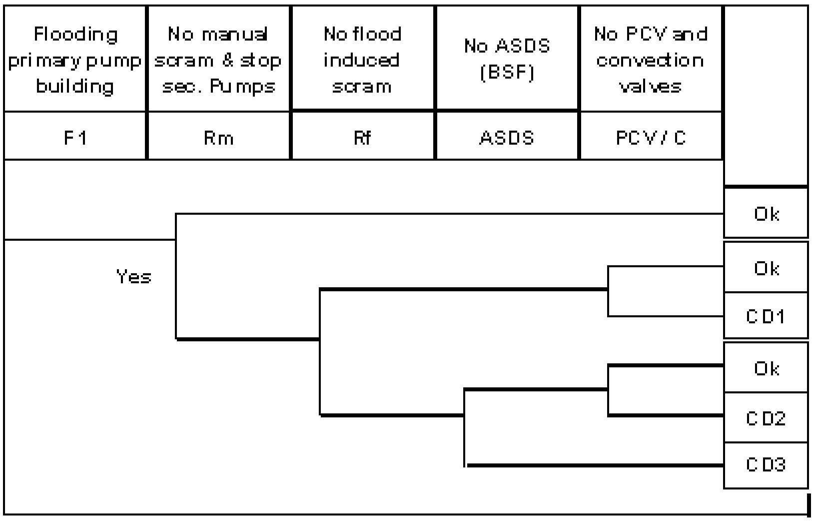
A.12 Flooding events leading to HFR core damage

From the event trees derived in the HFR Internal Flooding Analysis, the combinations of flooding and failures that lead to core damage can be derived:

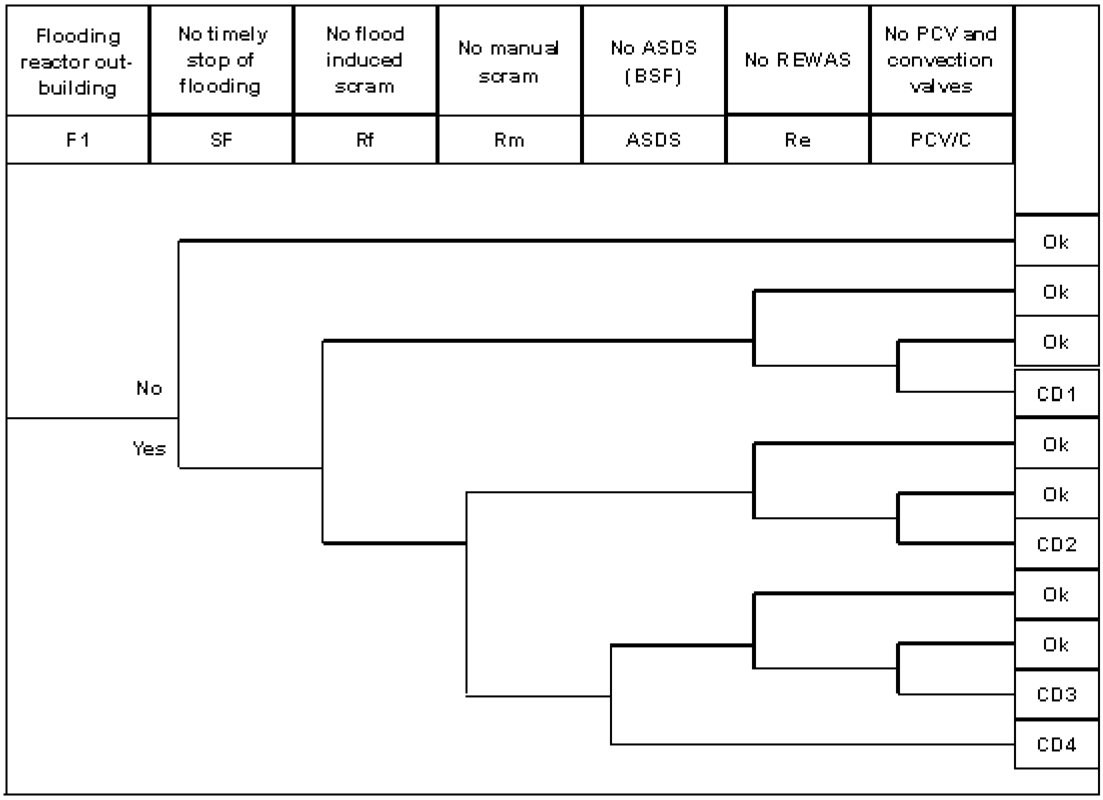
Flooding in the Primary Pump Building, before the heat exchangers:



Flooding in the Primary Pump Building, after the heat exchangers:



Flooding in the Reactor Outbuilding:



A.13 Radiation protection - legal framework

Legal Framework

Compliant to Chapter 9 of the Decree on Radiation Protection, intervention by personnel during emergency situations is restricted by justification and ALARA principles (Article 112) as well as dose limitations (Article 118), see Table A13.5 below. In addition NRG has the obligation to plan these interventions, to execute these interventions in accordance with written and trained procedures that are approved by the Competent Authority (Article 115, 119). NRG is responsible for the registration of the consequences and effectiveness of these interventions (Article 117). Emergency personnel involved in these interventions which are not employees of NRG (such as municipal fire brigade, police, etc.) are bound to similar restrictions, see Article 113.

Table A13.5: Dose restrictions to intervention during emergency operations for employees and emergency workers as presented in Article 118.

Type of intervention	Restriction to effective dose (mSv)
Saving lives	750*
Saving important materially interest	250
Support of monitoring measurements and implementations of emergency response measures such as evacuation, iodine-prophylaxis, public discipline and public safety	100

*) This value may only be superseded when urgent saving of live or saving of very important material interests are at stake. The person will participate in this intervention on voluntary base and will be briefed on the risks during the intervention.

Dose limitation of interventions by non-NRG staff emergency workers is also regulated in Chapter 9 of the Decree on Radiation Protection.

Fire brigade personnel is allowed to intervene without specific dose planning, if the dose rate at the locations is below 25 µSv/hr or the accumulated dose is less than 2 mSv (see Radiologisch handboek hulpverleners, 2004. In case the expected dose accumulated during intervention is larger than 2 mSv, preparation of the intervention, dose planning, registration and the limits of Article 118 of the Decree are applicable.

General approach of fire brigade and other public emergency workers is described. The facilities of NRG, excluding HFR and LFR are according to NPK, so-called B-objects for which the general approach during intervention is specifically described.

The task of dose planning during interventions in a NRG facility is a task of the local radiation protection expert of the facility concerned. Dose planning and other measures as described in Chapter 9 of the Decree on Radiation protection are tasks of the General Radiation Protection supervisor of NRG (verantwoordelijk stralingsdeskundige). Responsibilities, competences and tasks of these officials are described in part 3 of the NRG safety report. The procedures are described in the NRG general management system.

A.14 Assessment of communication systems' vulnerability

Availability of CAS and ERO rooms

The HFR guard building, housing the CAS room is not built to resist an extreme earthquake. Flooding and extreme weather conditions may impair the communication equipment of the CAS (see below). It seems unlikely that a cyber-attack may impair the availability of the CAS room (although the latter may affect communication equipment, as discussed below), but it cannot be excluded without further assessment. In case of localised events (aircraft crash, explosion, fire[53]), the availability of the CAS room can potentially be impaired (note that currently there is no backup CAS room available).

The availability of the ERO room is judged to be less vulnerable: due to its redundancy at two spatially separated buildings, localised events (aircraft crash, explosion, fires of neighbouring buildings or forest outside the OLP) can be accommodated.

Internal & external wired telephone infrastructure (KPN)

The wired telephone lines that connect the OLP with the outside world are judged not be a reliable resource in case of flooding or earthquake. It is assumed that the connections to the external KPN network can potentially be impaired in case of severe lightning strikes or cyber-attacks. It is assumed that localised events (aircraft crash, explosions, fires) can impair (connection to) the external KPN network only partially[54]. For the internal network, it is judged that localised events (aircraft crash, explosions, fires, severe lightning strikes), earthquakes and cyber-attacks can impair its function. It is currently unclear whether a flooding event can impair the telephone server situated in the basement of the Office Building. Furthermore, it must be noted that in case of LOOP the provision of emergency power by diesel generators is limited to about three days.

Internal & external wireless telephone infrastructure (GSM)

The reliability of the GSM network (internal and external) is difficult to assess. The GSM repeater on one of the JRC buildings is not equipped with ups/emergency power, and it is not unlikely that the GSM network will get overloaded in case of a severe accident [55]. The wireless telephone connection on-side and

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[53] i.e., a fire in the CAS room

[54] the KPN network enters the OLP at different locations/buildings

[55] Although it is assumed that in case of emergency the employees not involved in rescue actions or countermeasures are ordered not to use the GSM network, this regulation seems not to be known to the personnel. Moreover, employees might not even be aware of an accident or critical situation.

off-side the OLP by GSM are therefore judged not be a reliable resource in case of flooding or earthquake and localised events as fires, explosions, lightning strike and aircraft crash that damage the local repeater. It is judged to be unlikely that the GSM network can be impaired by a (on-site) cyber-attack.

National wired emergency telephone infrastructure (Noodnet)

Since the infrastructural conditions are quite similar, for the availability of the Noodnet similar considerations apply as for the KPN network. However, it should be noted that because the Noodnet does not require on-site power supply, it can assumed to be functional also in case of a long-lasting LOOP. Because contact via the Noodnet is foreseen via the CAS, a fire in the CAS presents an additional vulnerability (see paragraph Availability of CAS room above).

National wireless emergency communication infrastructure P2000 (Semafoon) & C2000 (Portofoon)

The P2000 and C2000 communication structures that connect the Research Location Petten (OLP) to external emergency organisations are judged not be a reliable resource in case of flooding. Insufficient information was available in order to assess whether the P2000/C2000 system would still be available in case one of the antenna towers in Petten or in the surrounding area (Callantsoog, Hargen, Schagen, Anna Paulowna) would fail due to regional (earthquake, lightning strike) or localised events (fire, aircraft crash, explosion). The P2000 and C2000 system are judged to be potentially vulnerable to cyber-attack, i.e. interference by external parties that suppress, alter or fake communication content.

Local wireless emergency network (Mobilofoon)

Insufficient information was available in order to assess whether the mobilofoon system would still be available in case of failure of parts of the system due to localised events. Lightning strike might disable the local repeater, but there is a backup system present at the OLP.

Pager (ERO representatives, BHV)

The pager system signal is provided by an external supplier (KPN). Insufficient information was available in order to assess whether the pager system would still be available in case of failure of parts of the KPN transmitter system due to regional or localised events. However, a large scale flooding is assumed to impair the usability of the pager system.

Intercom/direct line CAS - HFR/direct line CAS - fire department

Although a detailed assessment of the topology of these systems was not performed, it is assumed that localised events (aircraft crash, explosions, fires, lightning strike) and an earthquake could potentially impair their functioning. A failure of these systems in case of the other events is assumed to be less likely, but cannot be excluded without further assessment.

Fax (ERO room)

Since the infrastructural aspects related to the vulnerability of the system are comparable, for the availability of fax communication in the ERO room it is assumed that it depends on the availability of both the internal KPN network and access to the external KPN network.

Internal & external computer network infrastructure

Since the infrastructural aspects related to the vulnerability of the system are comparable, the vulnerability of computer network communication is assumed to be comparable to that of the KPN telephone network. Note that although the computer network is a strong communication tool, it has currently no (documented) role in the SAM organisation.

Synthesis

The analysis above shows that the efficacy of the SAM organisation can be impaired by the effect of external events on the communication tools. Flooding as regional event can affect both public communication networks (KPN, GSM) and emergency networks (C2000, P2000, Noodnet) at the same time resulting in a relevant challenge for the SAM organisation. The central role of the CAS in emergency communication (see also Section **6.1.2.4**) makes it also vulnerable in case of technical failure: in case both the KPN and GSM networks fail, it will be difficult to reach the CAS[56]. In the current internal emergency plans, no alternative communication route is described in case the CAS room will not be functioning for any reason. Thus it is unclear, how the NRG/OLP emergency response organisation will communicate in case the CAS room will not be available. The same applies to paging of Head ERO and further ERO crisis team members: the CAS is given as the only location where information on the ERO crisis team members and further responsible persons in charge is available (alarm roles).

A.15 Potential effects from nearby installations

Fire hazard and releases of airborne substances from facilities on the OLP

Important for fire hazard is the amount of flammable material present in each facility on the OLP. With respect to fire hazard and the amount of released (potentially toxic) substances, the central storage facility

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[56] note that the exact location of the CAS is not mentioned in any of the cited documents

of ECN is considered a relevant location. A fire in this facility is one of the three reference scenarios of the OLP disaster response plan (RBP) and could potentially have effect up to 1.5 km downwind. Other facilities with larger amounts of flammable materials are the production facilities of Covidien, and the diesel tanks spread over the site.

Table A15.6 presents the potential impacts of smoke and other airborne toxic substances as result of a fire at one facility on other facilities at the OLP, dependent on the wind direction. The potential impact is estimated on the basis of distance and potential contents of the fire source.

As Table A15.6 shows, the HFR might be affected by smoke from a fire at Covidien as well as from a fire at the central storage of ECN in case of south-east to south-westerly winds. In case of north to north-easterly winds, a fire at the JRC complex might affect the HFR. The HCL-MPF might be affected by smoke from fire at the Covidien building or at the central storage of ECN in case of south-west to north-west winds, while fire at the HFR will have a limited impact on the HCL-MPF in case of north to north-west winds.

Relevant for the safe shutdown of the HFR and the HCL-MPF are availability of emergency power and cooling water supply. If the emergency diesel generators are affected by smoke or fire, the supply of emergency power would be impaired. This is described in Chapter 5.

As these are supplied in merely closed systems, it is very unlikely that the drinking water and cooling water supply would be affected by smoke or fire in other buildings.

The facilities of the SAM organisations have quite some redundancy:

- In case of a severe accident the SAM organisation needs an ERO room. When one of the ERO rooms is affected by smoke of fire from neighbouring facilities, the other ERO room will be used.
- When the fire station is affected by fire or smoke, the fire engines will be moved away from the station and remain operative without availability of the building.
- When the ICT building is affected by smoke of a fire from a neighbouring building this might impair the means of communication with off-site organisations. Loss off communication systems is discussed in Section **6.1.3.2**.
- The DWT can be used for decontamination of objects and people. An alternative location has to be envisioned if the DWT cannot be used. This is described in Section **6.1.1.4**.

effects on OLP facility / source s at OLP facility	JRC	HFR	emergency diesel generator	HCL	Covidien	GBD Building	fire station	FORUM	Office Building	Emergency diesel generator ICT	central storage ECN	DWT	WSF	JGL
ERO crisis room (Forum)	③	③↑	①↑	①↑	①↑	①↑	①↑	①↑	①↑	①↑	②↑	①↑	①↑	①↑
HFR and CAS	②↓	③	③↑	②↑	③↑	①↑	①↑	①↑	①↑	①↑	③↑	①↑	①↑	②↑
emergency diesel generator	②↓	③↓	③	③↑	③↑	①↑	①↑	①↑	①↑	①↑	①↑	①↑	①↑	②↑
HCL	①	②↓	①↓	③	③ →	①↑	①↑	①↑	①↑	①↑	③ →	① →	① →	③↓
GBD building	①↓	①↓	①↓	①↓	② →	③	①↑	①↑	①↑	① →	②↓	①↓	①↓	①↓
Fire station	①↓	①↓	①↓	①↓	①↓	①↓	③	② →	①↑	① →	①↓	①↓	①↓	①↓
ERO crisis room (JRC premises)	①↓	①↓	①↓	①↓	①↓	①↓	① ←	③	②↓	①↓	②↓	①↓	①↓	①↓
ICT	①↓	①↓	①↓	①↓	①↓	①↓	① ←	①↑	③	② ←	①↓	①↓	①↓	①↓
Emergency diesel generator ICT	①↓	①↓	①↓	①↓	①↓	①↓	① ←	①↑	② →	③	①↓	①↓	①↓	①↓
DWT	①↓	②↓	①↓	①↓	①↓	①↓	①↑	①↑	①↑	①↑	②↑	③	③↓	① ←
Drinking water supply building	① →	① →	① →	①↑	② →	① ↑	① ↑	① ↑	① →	①↑	① →	① →	① →	①↑
Cooling water inlet building	① →	① →	① →	① ↑	② →	① ↑	① ↑	① ↑	① →	①↑	① →	① →	① →	①↑

Table A15.6 Mutual impacts of fire in the OLP facilities such as smoke. ↑: wind direction;
①: negligible to minor impact; ②: minor to medium impact; ③: medium to major impact

The CAS has no backup. As the CAS is located on the HFR site, influences at the HFR pertain for the CAS as well. Consequently besides fires at the HFR itself, also fires at Covidien or the central storage of ECN (southerly winds) and the JRC complex (northerly winds) might impair its functionality.

Releases of airborne radioactive material at nuclear facilities on the OLP

Potential impacts of releases of airborne radioactive substances on safe operation of the OLP nuclear facilities are presented in Table A15.7. Unmanned facility buildings are assumed not to be at risk and are discarded in Table A15.7. The most relevant potential sources of accidental releases are the HFR and HCL. The explanation of the relevance of radioactive releases on the facilities is similar to that of the effects of smoke on the facilities.

effects on OLP facility	source s at OLP facility	JRC	HFR	HCL	Covidien	GBD Building	DWT	WSF	JGL
ERO crisis room (JRC premsis)		①	③↑	①↑	①↑	①↑	①↑	①↑	①↑
HFR and CAS		①↓	③	③↑	③↑	①↑	①↑	①↑	②↑
HCL		①↓	②↓	③	③ →	①↑	① →	① →	③↓
GBD Building		①↓	①↓	② ↓	② →	③	① ↓	①↓	①↓
Fire station		①↓	①↓	①↓	①↓	①↓	①↓	①↓	①↓
ERO crisis room (Forum)		①↓	①↓	①↓	①↓	①↓	①↓	①↓	①↓

Table A15.7:Impacts of radioactive releases on OLP facilities. ↑: wind direction;
①: negligible to minor impact; ②: minor to medium impact; ③: medium to major impact

Effects of direct exposure to radioactivity contained in the HFR reactor building

In case of a severe accident at the HFR leading to an internal release of radioactivity into the containment of the HFR reactor building – e.g. due to melt of an in-core experiment or even part of the core itself – to limit their external exposure also personnel present in neighbouring facilities might need to be evacuated temporarily, depending on the extent of the release. For the severe accident considered in Section **6.4.2.5** – core melt leading to a release of all gaseous and volatile radionuclides into the containment of the HFR reactor building – also the external radiation effects have been assessed. The radionuclides involved in are for the most part short-lived, causing dose rates to drop by several orders of magnitude within days. Hence, possible access restrictions for neighbouring facilities would be of short duration.

effects on OLP facility	Times after reactor scram	3 hours	10 hours	25 hours	100 hours	1000 hours
ERO crisis room (JRC premises)		③	②	①	①	①
HFR		③	③	③	③	②
CAS		③	③	③	②	①
NDO / RMS		①	①	①	①	①
HCL		③	③	②	①	①
GBD Building		③	②	①	①	①
Fire station		②	①	①	①	①
ERO crisis room (Forum)		②	①	①	①	①

Table A15.8: Impacts of core melt at the HFR on habitability of OLP facilities.
①: less than 0.1 mSv/hr; ②: between 0.1 and 1 mSv/hr; ③: over 1 mSv/hr

The assessment did show that without containment venting unshielded dose rates would initially be around 10 Sv/hr in the direct vicinity of the HFR. The level of 1 mSv/hr would approximately be reached at 600 m after 3 hours, at 400 m distance after 10 hours, at 200 m after 25 hours, at 100 m after 100 hours and in the direct vicinity of the HFR after about one month. The level of 0.1 mSv/hr would approximately be reached at 950 m after 3 hours, at 700 m distance after 10 hours, at 350 m after 25 hours, at 200 m after 100 hours and at 50 m of the HFR after about one month.

Because of its thick shielding walls, the NDO building – housing the RMS, operational from mid-2012 on – will always be habitable in these circumstances.

A.16 Strategies for High Flux Reactor (HFR)

A.16.1 Strategies to restore sub-criticality

A. <u>From existing procedures</u>

1. Reactor stop by control rods.
2. Reactor emergency stop by control rods.
3. Anticipated Transient Without Scram situation. In the control room the following can be tried to reduce reactor power:
    a. Switch off magnet power supply
    b. Turn off master switch control rod drives
    c. Try different commands (rods down, AVV, …).
4. Anticipated Transient Without Scram situation. The following can be done to make and maintain the reactor sub-critical:
    a. Remove molybdenum production facilities (in-core and out-core molybdenum facilities)
    b. Place the Cadmium plate at the west side of the reactor vessel wall
    c. Increase pool temperature or temperature of the primary system
    d. Remove fuel elements out of the core.

Explanation:

- With the first two strategies the reactor is made sub-critical by use of normal functioning control rods. A reactor shutdown is caused when the safety systems settings are exceeded. When not all control rods are dropped, the Scram button must be pushed.
- The third strategy is used when the reactor has not scrammed after strategies 1 and 2. In the control room several possibilities with normal available features or means exist to reduce reactor power.
- The fourth strategy is used when the control rods are mechanically blocked. The first step is to remove all in-core and out-core facilities. This will have an effect of 15 pcm per gram ^{235}U. A

further decrease of reactivity can be achieved by placing the Cd plate at the west side of the reactor vessel wall, which reduces the reactivity with 300 pcm. These two actions make the reactor sub-critical for some 45 hours (a reactivity reduction of about 300 pcm should be enough to shut down the reactor, so placing the Cd plate is in principle sufficient). To minimise reactivity effects (effect: ca. -100 pcm per 10 °C increase of core cooling water) the pool temperature must be increased by reducing or stopping the secondary cooling of the pool heat exchanger. To increase the temperature of the primary system, the secondary cooling of one or more primary heat exchangers must be reduced or stopped. The last step is to maintain sub-criticality after 45 hours. Because of the increasing xenon concentration after shutdown, the reactivity of the core decreases rapidly due to neutron capture by xenon. After 10 hours the xenon concentration decreases again due to decay, resulting in a slow increase in reactivity in the core. Without strategic actions some 45 hours after shutdown the reactor will become critical again. The loss of 3300 pcm reactivity due to xenon decay must be compensated within that period of time. To that end the fuel elements might be removed, preferably those having the largest contribution to the core reactivity. To remove fuel elements several operations have to be carried out:

- Removal of all molybdenum production facilities with reactor vessel head penetrations
- Removal of the reactor vessel head lid
- Opening and removal of the grid bars.

It must be noticed that in a situation where the control rod drive mechanism failed due to mechanical defects or blockades, it could be possible that at least one of the three mentioned actions cannot be carried out. When this is the case the fuel rods cannot be removed.

Remark 1:

In procedure B-24 Cd plugs are used to maintain sub-criticality. At this moment the Cd plugs are not available at the HFR. So sections in procedure B-24 in which Cd plugs are used for this purpose do not apply yet.

Remark 2:

Before implementation of a specific accident management measure possible negative impacts will be identified and, if necessary, evaluated. Dependent on the results of the evaluation the accident management measure(s) will be implemented or not. In the last steps of strategy 4, for removing the fuel elements from the core the pool water level has to be lowered to enable opening of the reactor vessel. However, lowering the water level in the reactor pool may put many hours of cooling time at risk. This means that such an action will only be performed if replenishment of the pool is assured, or if the water level in the pool is low anyway.

B. *Possible within the HFR areas, but currently not mentioned in a procedure*

(None)

A.16.2 Strategies for maintaining sufficient coolant inventory

Note that for the function "maintaining sufficient coolant inventory" distinction is made between the following strategies:

- strategies for maintaining sufficient coolant inventory for heat removal of the core, presented in Appendix A.16.2.1
- strategies for maintaining sufficient coolant inventory for heat removal of the spent fuel, presented in Appendix A.16.2.2
- strategies for maintaining sufficient coolant inventory for heat removal of molybdenum production facilities, presented in Appendix A.16.2.3

A.16.2.1 Strategies for maintaining sufficient coolant inventory for heat removal of the core

A. From existing procedures

1. The primary system. The water content of the primary system is controlled by the level control of the expansion vessel.
2. Water from pool 1, pool 2 and the reactor pool. Pool water is available by opening of the pool water injection valve (Bassinwater Noodsuppletie Systeem Reactorkern, BNSR).
3. Water from pool 1, pool 2 and the reactor pool. Pool water is available by opening of the two convection valves.
4. Demin water storage tanks. The reactor pool (and primary system) can be filled with water from several storage tanks. There is no procedure for filling the primary system from the storage tanks.
5. *Hydrofoor* (PWN) drinking water. The reactor pool can be filled by opening valve HY-116.
6. Supply of pool water. Refill the pool by pumping leaked primary water from the basement of the reactor building to the reactor pool by use of the warm drain pumps.

Explanation:

- The first strategy is used during normal operation.
- The second strategy has priority over strategies 3-5. The pools have the following water contents:
  - Reactor pool: 151 m^3
  - Pool 1: 106 m^3
  - Pool 2: 84 m^3

The pools are separated by two stacked doors. The pools are connected by a slit in the middle of the two doors. During normal operation the top door of the reactor pool between the reactor pool and pool 1 is removed. For an accident situation with large leakage in the primary system the available amount of water for cooling the core is about half the amount of pool 1, half the amount of pool 2 and more than half the amount of the reactor pool contents. For the latter this depends on the emergency valve used. The height of the BNSR valve is a little below the top of the lowest pool door (height lowest pool door: 4.3 m), the height of the highest convection valve is 2.4 m above the pool bottom, the lowest convection valve is 0.3 m above the pool bottom.

For an accident situation without leakage of the primary system the water contents of the pools can be used for heat removal (however, the effectively available water contents depend on the grade of mixing of hot water from the reactor pool with the water in pools 1 and 2).

- The two 82.5 m^3 storage tanks (storage for primary and pool water) in the basement of the PPG have a guaranteed minimum content of 30 m^3 each, so a total water volume of 60 m^3 can be pumped to the reactor pool. For storage of produced demin water two additional storage tanks (Duper tanks) are used. They have a volume of 30 m^3 and 15 m^3 and can also be used to pump water into the pool. Only one tank is filled with demin water from the demin water production system. So a minimum water content of 15 m^3 is available. These tanks are located outside the PPG.

  The water is pumped into the reactor pool and flows to pools 1 and 2 as long as the level of the reactor pool is above the lowest pool door. The height of the lowest pool door is 4.3 m.

  Demin water can be pumped into the primary system directly via the 82.5 m^3 storage tanks. Water from the additional tanks (15 m^3 and 30 m^3) can be pumped to the primary system only via the 82.5 m^3 storage tanks.

- Strategy 4 can be used to maintain a sufficient high pool level to cool the core by use of the CV or BNSR valves. This strategy is typically used in large (pool) LOCA situations.
- When the above strategy is not sufficient, the reactor pool can be filled with hydrofoor water (PWN). The valve HY-116 is located at the second platform in the RB. The hydrofoor water used by the HFR is made in a separate water make up building (Reinwaterkelder) located at the OLP area. In this make up building drinking water is collected in 4 separate tanks 120 m^3 each, from which the facilities at the OLP are served. The make-up flow for these tanks is 26 m^3/hr drinking water, delivered by the regional utility (PWN). 70% of one tank is reserved for the HFR, resulting in a stock of about 80 m^3 of water. The water can be transported to the HFR either by the drinking

water pipe (5 bar) or by use of a second pipe system, the gravity pipe (valleiding). So for filling the reactor pool a minimum of 80 m^3 water is available.

- Strategy 6 can be used to maintain a sufficient high pool level to cool the core by use of the BNSR valve. This strategy is typically used for in-containment ex-pool LOCA situations.

B. *Possible within the HFR area, but currently not mentioned in a procedure*

1. Water from the storage tanks (2 x 82.5 m^3) can be pumped directly to the pool (with the fill & drain pump) or to the primary system (with the fill & drain pump or the demin pump) if electrical power is available. Refill of the storage tanks by hoses is possible by opening the hatch at the top. Water sources and measures for refill are:
   a. Duper tanks (15 and 30 m^3, minimum of 15 m^3 available; they are connected to the storage tanks, so only valves must be opened and present pumps must be activated)
   b. Reinwaterkelder (4 tanks of 120 m^3 each, with a minimum of 70% of one tank reserved for the HFR). A flexible hose (length indication: 40-50 m) can be connected from the hydrofoor system (storz connection) to the storage tanks. The water is pumped by use of the hydrofoor pumps or mobile pumps
   c. Lake in front of the HFR area by deployment of the fire brigade. Via a flexible hose and mobile pump the water is transported
   d. Water from the fire brigade truck. Via a flexible hose and mobile pump the water is transported
   e. Water from the channel between the SPG and PPG. This water is pumped by use of a mobile pump via a flexible hose placed in the inspection sump, close to the PPG

2. Pumping water via the storz connection on the pool heat exchanger (located: PPG cell 8) into the pool or into the primary system. For pumping into the primary system, the normal feed pumps (located also in cell 8 of the PPG) are used. Water sources and measures for refill are:
   a. The storage tanks (2 x 82.5m^3, a minimum of 2x30 m^3 is available). Via the hatch at the top of storage the tanks, a flexible hose and a mobile pump the water is transported
   b. The Duper tanks (15 and 30 m^3, minimum of 15 m^3 available). Via a storz connection, flexible hose and mobile pump the water is transported
   c. Reinwaterkelder (4 tanks of 120 m^3 each, with a minimum of 70% of one tank reserved for the HFR). Via a storz connection at the hydrofoor system, flexible hose and hydrofoor pump or mobile pump, the water is transported
   d. Hot and warm drain tanks (resp. size: 2 x 2 m^3 tanks 1&2, size: 2 x 3.5 m^3 tanks 3&4. The tanks are empty when the content is just drained to the decontamination facility). Via a storz connection, flexible hose and mobile pump the water is transported

e. Lake in front of the HFR area by deployment of the fire brigade. Via a flexible hose and mobile pump the water is transported
f. Water from the fire brigade truck. Via a flexible hose and mobile pump the water is transported
g. Water from the channel between the SPG and PPG. This water is pumped by a mobile pump via a flexible hose placed in the inspection sump, close to the PPG

Explanation:

- The first strategy uses the fixed connections of the storage tanks with the pool and primary system. Electrical power is needed for the use of the fill & drain pump and/or the demin pump.
- The second strategy uses the connection of the pool heat exchanger with the pool and primary system. For pumping directly into the primary system, electrical power is needed for the normal feed pumps.

A.16.2.2 Strategies for maintaining sufficient coolant inventory for heat removal of spent fuel

A. <u>From existing procedures</u>

1. Demin water storage tanks. The reactor pool can be filled with water from several storage tanks.
2. Hydrofoor (PWN) drinking water. The reactor pool can be filled by opening valve HY-116.
3. Supply of pool water. Refill the pool by pumping leaked primary water from the basement of the reactor building to the reactor pool by use of the warm drain pumps.

Explanation:

- The spent fuel is cooled by the pool water and stored at the bottom of pool 1 and/or pool 2. The height of the top of the spent fuel is about 1 m; during normal operation the water level of the pools is 8.7 m.

  The two 82.5 m^3 storage tanks (storage for primary and pool water) in the basement of the PPG have a guaranteed minimum content of 30 m^3 each, so a total water volume of 60 m^3 can be pumped to the reactor pool. For storage of produced demin water two additional storage tanks (Duper tanks) are used. They have a volume of 30 m^3 and 15 m^3 and can also be used to pump water into the pool. Only one tank is filled with demin water from the demin water production system. So a minimum water content of 15 m^3 is available. These tanks are located outside the primary pump building (PPG).

The water is pumped into the reactor pool and flows to pool 1 and 2 as long as the level of the reactor pool is above the lowest pool door. The height of the lowest pool door is 4.3 m.

- When the above strategies are not sufficient, the reactor pool can be filled with hydrofoor water (PWN). The valve HY-116 is located at the second platform in the reactor building. The hydrofoor water used by the HFR is made in a separate water make up building (Reinwaterkelder) located at the OLP area. In this make up building drinking water is collected in 4 separate tanks 120 m^3 each, from which the facilities at the OLP are served. The make-up flow for these tanks is 26 m^3/hr drinking water, delivered by the regional utility (PWN). 70% of one tank is reserved for the HFR, resulting in about 80 m^3 of water. The water can be transported to the HFR either by the drinking water pipe (5 bar) or by use of a second pipe system, the gravity pipe (valleiding).
  So for filling the reactor pool a minimum of 80 m^3 water is available.
- Strategy 3 is typically used in pool leakage situations. The amount of leaked pool water should not exceed the capacity of the warm drain pumps too much

B. *Possible within the HFR area, but currently not mentioned in a procedure*

See appendix A.16.2.1

A.16.2.3 Strategies for maintaining sufficient coolant inventory for heat removal of the molybdenum production facilities

A. *From existing procedures*

1. Demin water storage tanks. The reactor pool can be filled with water from several storage tanks.
2. Hydrofoor (PWN) drinking water. The reactor pool can be filled by opening valve HY-116.
3. Supply of pool water. Refill the pool by pumping leaked primary water from the basement of the reactor building to the reactor pool by use of the warm drain pumps.

Explanation:

- The cooling systems of the molybdenum production facilities use water (by suction) from the reactor pool (U-BEKWS and HD-BEKWS) and pool 1 (BEKWS). The suction pipe for the cooling systems is located at about 3.5 m above the pool bottom, so below this level the cooling systems will not operate.

The two 82.5 m^3 storage tanks (storage for primary- and pool water) in the basement of the PPG have a guaranteed minimum content of 30 m^3 each, so a total water volume of 60 m^3 can be pumped to the reactor pool. For storage of produced demin water two additional storage tanks (Duper tanks) are used. They have a volume of 30 m^3 and 15 m^3 and can also be used to pump water into the pool. Only one tank is filled with demin water from the demin water production system. So a minimum water content of 15 m^3 is available. These tanks are located outside the primary pump building (PPG).

The water is pumped into the reactor pool and flows to pool 1 and pool 2 as long as the level of the reactor pool is above the lowest pool door. The height of the lowest pool door is 4.3 m.

- When the above strategies are not sufficient, the reactor pool can be filled with hydrofoor water (PWN). The valve HY-116 is located at the second platform in the RB. The hydrofoor water used by the HFR is made in a separate water make up building (Reinwaterkelder) located at the OLP area. In this make up building drinking water is collected in 4 separate tanks 120 m^3 each, from which the facilities at the OLP are served. The make-up flow for these tanks is 26 m^3/hr drinking water, delivered by the regional utility (PWN). 70% of one tank is reserved for the HFR, resulting in about 80 m^3 of water. The water can be transported to the HFR either by the drinking water pipe (5 bar) or by use of a second pipe system, the gravity pipe (valleiding).
  So for filling the reactor pool a minimum of 80 m^3 water is available.
- Strategy 3 is typically used in pool leakage situations. The amount of leaked pool water should not exceed the capacity of the warm drain pumps too much.

B. *<u>Possible within the HFR area, but currently not mentioned in a procedure</u>*

See appendix A.16.2.1

A.16.3 Strategies for primary heat removal

For the function “primary heat removal” distinction is made between the following strategies:

- strategies for cooling of the core, presented in Appendix A.16.3.1
- strategies for spent fuel cooling, presented in Appendix A.16.3.2
- strategies for cooling of molybdenum production facilities, presented in Appendix A.16.3.3

A.16.3.1 Strategies for cooling of the core

A. From existing procedures

1. Cooling with the primary system. The primary system is a closed system that removes heat via three parallel heat exchangers and forces circulation with three parallel primary pumps P1, P2 and P3.
2. Cooling with the primary system by use of the (emergency) decay heat removal pumps P4 or P5.
3. Cooling with pool water. Opening of convection valves (CV).
4. Cooling with pool water. Opening of pool water injection valve (Bassinwater Noodsuppletie Systeem Reactorkern, BNSR).

Explanation:

- The first two strategies have priority over the other strategies. Strategy 1 (normal operating system) is only possible when off-site power is available. The emergency pumps P4 and P5 start automatically (manually is also possible) when the primary pumps stop. Pumps P4 and P5 are independent and powered by the emergency power supply.
- With strategy 3 and 4 the core is cooled by direct use of pool water. Strategy 3 is used typically for complete station black out situations (LOOP-SBO2). Convection valves can be opened, after local unlocking of the valves, from the control room as well as manually (key procedure, to be included in operating procedure B-20. Strategy 4 is typically used for LOCA situations. Pool water enters the primary system and removes just enough heat to cool the core. The BNSR valves can be opened from the control room as well as manually (key procedure, to be included in operating procedure B-20.

B. Possible within the HFR area, but currently not mentioned in a procedure

1. Removal of in-core molybdenum production facilities and/or removal/lifting of the reactor vessel head
2. Removal of observation windows (kijkglaasjes*)* in the reactor vessel head. Two clamps have to be removed by use of specific tools that are (always) present at the third platform in the reactor building

Explanation:

- If strategies 1 and 2 from the existing procedures have failed it is crucial for emergency cooling of the core that there is a possibility to inject pool water into the primary system. When the valves of the BNSR or CV fail to open, no cooling by use of pool water is possible. When the previous mentioned strategies are not possible, the remaining option is to create/force an entrance to the primary system from the pool to cool the core by pool water.

A.16.3.2 Strategies for spent fuel cooling

A. <u>From existing procedures</u>

1. Cooling with the normal pool cooling system (Bassin Koelwatersysteem, BKWS and secondary heat removal system (cooling by convection).

Explanation:

- The BKWS removes the heat to the secondary system. This is the normal procedure.

B. <u>Possible within the HFR area, but currently not mentioned in a procedure</u>

1. Cooling by heat exchanging (water to water). Connection of a hose at a fire hose connection point (*hydrant*, not used anymore for fire fighting purposes) just outside the reactor building at the HFR area. The hose can be connected to the secondary side of the heat exchanger (nr 4). The heated water (drinking water from the Reinwaterkelder) is pumped into the secondary heat removal channel (to the North Sea).
2. Cooling by heat exchanging (water to air). Pool water can be pumped to the two storage tanks (82.5 m^3 each) located in the basement of the primary pump building. The surface of the tanks and connecting pipe-work function as a heat exchanger to the air. Ventilation of the storage room is realised by opening the door (connection to the open air).
3. Cooling by evaporation of water from the pools. Refill the pools with water from the storage tanks (see also appendix A.16.2).
4. Cooling by evaporation of water from the pools. Refill the reactor pool with PWN drinking water by opening valve HY-116.

Explanation:

- The first strategy is only possible when the pool cooling system BKWS functions (see strategy A1) and the secondary heat removal system has failed.
- The second strategy is also used during reactor stops when pool water temperature increases too much. The valves of the storage tanks are operated manually in the basement of the reactor building (RB).
- For a situation where cooling by the second strategy is not possible the last two strategies 3 and 4 can be applied. First the pools heat up to boiling temperature followed by evaporation. Because of the evaporation the level in the pools will decrease. Refill is necessary with strategy 3 and 4.

A.16.3.3 Strategies for cooling of molybdenum production facilities

A. From existing procedures

1. At the HFR three systems are used for cooling the molybdenum production facilities:
   a. BEKWS, normal cooling system.
   b. U-BEKWS, extended cooling system.
   c. HD-BEKWS, high pressure cooling system for Incomodo and Tycomo irradiations.

Explanation:

- The molybdenum production facilities are located either in the core or outside the reactor vessel (ex-core) and are cooled with a dedicated cooling system, using cooling water (by suction) from pool 1 (BEKWS) or the reactor pool (U-BEKWS and HD-BEKWS). The pump systems are located at the second platform in the reactor building. The BEKWS and U-BEKWS systems are connected with each other but isolated during normal conditions. When pressure drops in the U-BEKWS system, the isolation is automatically broken whereby the BEKWS system takes over the cooling from the U-BEKWS (one of the pumps of the BEKWS is connected to emergency power). When the pressure drops in the press header of the HD-BEKWS system, two emergency pumps are automatically started (both connected to emergency power) to cool the system.

B. Possible within the Reactor Building, but currently not mentioned in a procedure

1. In case of damage of experiment cooling hoses, the damaged (open) part of the hoses must be fully submerged in the pool water to force water into the cooling system in order to facilitate gravity driven convection cooling. An alternative is to cut the inlet hose at the top of the irradiation facility and keep the hose in the pool (submerged).

2. When the cooling system(s) of the molybdenum production facilities fail(s), remove sample holders from the facilities and store them at the isotope tables.

Explanation:

- If forced flow is lost it is important to keep the molybdenum production facilities completely filled with water in order to make cooling by natural circulation possible. If a damaged hose is under water, the possibility of preventing target failure is increased.
- The samples holders from the molybdenum production facilities on the isotope tables are covered and cooled by the pool water.

A.16.4 Strategies for secondary side heat removal

A. *From existing procedures*

(None)

Explanation:

- The secondary system uses water from the Noordhollands Kanaal for heat exchange. The heated water is pumped into the North Sea. The system consists of the following:
    - Inlet (Noordhollands Kanaal) with coarse screen and rotating scrape screen
    - Filter pools, belt filters, pump pool and pumps located in the secondary pump building (SPG)
    - First high point
    - Heat exchangers in the primary pump building (PPG)
    - Vacuum tank/system
    - Second high point connected to vacuum system
    - Sea side block valve and outlet (North Sea).

When secondary flow stops or reduces, at least one of the following main reasons causes the stop:

I. No/not enough water available (e.g. no water in the SPG pools due to low water level of the Noordhollands Kanaal, damage to piping)

II. Screens or Filters blocked

III. Pumps not available (e.g. system failure or spurious control signals).

In the control room alarms (for instance the sub alarms from alarm nr. 35) are present for the different sub-systems of the secondary heat removal system. In case of an alarm of one of these sub-systems, the sub-system must be checked (according to a list given in the alarm description) at the specific location within or outside the OLP area.

Remark:

The inlet and outlet of the secondary system are outside the OLP area. The HFR operators have experience with removal of blockages of the inlet at the Noordhollands Kanaal. There is no experience with removal of blockages of the outlet (outlet has never been blocked in the past).

B. *Possible within the HFR area, but currently not mentioned in a procedure*

1. Connection of a hose at a fire hose connection point (*hydrant*, not used anymore for fire fighting purposes) just outside the reactor building at the HFR area. The hose(s) can be connected to the secondary side of the heat exchangers. The heated water (drinking water from the Reinwaterkelder) is pumped into the secondary heat removal channel (to the North Sea).

Remark:

Instead of drinking water also water from the little lake in front of the HFR, outside the HFR area, can be used. A mobile pump is needed for transportation to the storz connection on the secondary heat exchanger.

2. Injection of (sea)water in the secondary cooling water system via the first de-aerating point.

Explanation:

- When the secondary heat removal systems fails the first strategy is only possible when the primary system and pool cooling system BKWS function.
- When water from the Noordhollands Kanaal is not available, (sea)water can be pumped in the secondary cooling water system (after the SPG and before the PPG). When sea water is used, much flexible hose length is necessary and probably several mobile pumps.
-

A.16.5 Strategies for HFR containment function

A. <u>From existing procedures</u>

1. Status Reactor Building: *low activity*
   - Ventilation system:
     i. Reactor Building isolation valves close
     ii. By-pass valves open ► inlet air from the reactor building directly to the air treatment building (LBG) and stack.
   - Off-gas system:
     i. Switches to clean filter
     ii. Off-gas to the air treatment building (LBG).
   - Hot cell system:
     i. Switches to recirculation mode (connected to off-gas system).

2. Status Ventilation Outlet: *high activity*

- Ventilation system:
  i. Reactor Building isolation valves close
  ii. By-pass valves open ► inlet air directly to the air treatment building (LBG) and stack.
- Off-gas system:
  i. Switches to recirculation mode ► no off-gas to the air treatment building (LBG) or stack.
- Hot cell system:
  i. Switches off.
- Supply systems with containment penetration
  i. N_2(l), Pneum. isotope transport, Reactor Building pressure measure balance,
  PPG Off-gas system: valves close
  ii. Pressed air: valve closes when $P \leq 5.5$ bar.

3. Status Ventilation Outlet: *year dose*

- Ventilation system:
  i. Ventilation system shuts off
  ii. Reactor Building isolation valves close
  iii. By-pass valves closed ► no air to the air treatment building (LBG) or stack
  iv. Closure of stack valves.
- Off-gas system:
  i. Switches to recirculation mode ► no off-gas to the air treatment building (LBG) or stack.
- Hot cell system:
  i. Switches off.
- Supply systems with containment penetration
  i. N_2(l), Pneum. isotope transport, Reactor Building pressure measure balance,
  PPG Off-gas system: valves close
  ii. Pressed air: valve closes when $P \leq 5.5$ bar.

4. In the situation of a leaking heat exchanger. Isolation of the leaking heat exchanger, which can be the:
   a. Primary heat exchanger or the
   b. Pool heat exchanger.

Explanation:

- The first strategy is typically used when an activity concentration of > 370 kBq/m^3 is measured on gas monitor 1 (GM1) on a 2 out of 3 set. GM1 measures the activity concentration from the

Reactor Building air outlet (breathing air). The GM1 signal results in a RSA. This confinement status ensures that during accident conditions where activity is released, effective removal of e.g. Iodine isotopes is realized (high radiotoxicity). This is achieved by forcing the Reactor Building air into the off-gas system in which several filters are operating. The pressure in the Reactor Building is controlled with an isolation valve (V-071) and a pressure control valve (V-072).

- The second strategy is typically used when the concentration > 18.5 GBq/m^3. The concentration is measured on gas monitor 2 (GM2) on a 2 out of 3 set. The GM2 signal results in a RSA. This measurement is done in the ventilation outlet (retardation room of the air treatment building, LBG) in which the ventilation and off-gas system come together. In off-gas recirculation mode Reactor Building air is forced into the different inlets and filtered at several locations in the off-gas system, due to which aerosols and Iodine are removed from the Reactor Building air.
- The third strategy is typically used when the *total dose* of 222 TBq is measured by the integrator of GM2. The air treatment building (LBG) is isolated, so there is no air removal to the stack. The filters of the off-gas system remove radioactive products in the recirculation mode.
- Strategy 4 is typically performed when activity is found in the secondary water of the heat exchanger outlets. If one heat exchanger is isolated from this system, still two heat exchangers are for cooling the primary system. For the pool only one heat exchanger is available, so pool cooling stops when its heat exchanger is isolated.

Remark:

A water lock protects the reactor containment building against an excessive over- and under-pressure. The pressures at which the water lock opens are:

- in case of overpressure: 5.0 m Wk (water height), corresponds to about 0.5 bar
- in case of under-pressure: 65 cm Wk.

The water lock is placed in the basement of the reactor outbuilding and consists of a cylindrical vessel, an open pipe that connects the water lock to the containment and a standpipe to the outside (environment).

No procedures or recovery possibilities are available for recovery of the water lock function.

B. Possible within the HFR area, but currently not mentioned in a procedure

1. Reduce containment pressure and/or mitigate fission product release by venting via absolute filters and active carbon filters in the air treatment building (LBG) to the environment.
2. Inundation of the Swan Lake (Zwanenmeer) when the U-bend piping in the Swan Lake is damaged.

Explanation:

- The gasses in the Reactor Building flow via absolute filters and active carbon filters to the stack of the air treatment building (LBG). Pressure is reduced in the Reactor Building and the emitted gasses are purified (by adsorption and particle filtration). This venting can be done until the year dose limit has been reached. Venting above the year dose limit is only permitted with permission of the national authorities.
- Piping of the primary and pool connections are routed via the Swan Lake (U-bend) which closes the water system by a water-lock (passive lock). These connections are the two primary pipes (hot and cold leg) and pipe systems like the pool cooling system (Bassin Koelwatersysteem, BKWS), the pool demineralization system (Bassin Demineralisatiesysteem, BDS), storage tanks (Opslagtanksysteem, OTS) etc. If the water-lock is damaged during accident conditions an open connection between the Reactor Building or damaged core and the environment (PPG is not leak tight) exists. By inundation of the Swan Lake an alternate water-lock can been created.

A.17 Strategies for Hot Cell Laboratories (HCL)

A.17.1 Strategies for HCL-Research Laboratory (RL) containment function

A. <u>From existing procedures</u>

(None)

Explanation:

Brief system description

The ventilation system of the HCL-RL consists of two main parts, namely:

- Air Supply System
- Air Removal System.

The air supply consists of one system (supply air fan) for the entire building. The inlet air is filtered and heated by a heating coil. To prevent excessive under-pressure in the maintenance hall in case the air supply system stops, the valve in the channel behind the heating coil is fixed in the fully open position.

The air removal consists of several systems that take care of different pressure zones in the building. In the areas with the highest risk of contamination the highest under-pressure exists. The hierarchy of air suction is such that air flows from relatively clean areas to relatively less clean areas. The air removal system consists of three separate parts:

- Cell ventilation
- Building ventilation
- Off-gas ventilation.

Cell ventilation

This system provides ventilation through an absolute filter of the concrete cells (cell A t/m E), the lead cells, the actinides laboratory and the creep laboratory. The air is extracted by two single centrifugal fans. The flow of exhaust air is controlled via inlet valves. If cell ventilation stops or when there is insufficient under-pressure in the suction channel, the entire (cell and building) ventilation system will stop (the off-gas system will still be running).

When activity is measured in the cell air, the air is led through an active carbon filter (automatic switchover).

Building ventilation

From the maintenance hall, loading locks and the pipe-works on the first floor the building exhaust air is carried through a pre-filter and an absolute filter. The building ventilation system also consists of two single centrifugal fans, both in permanent operation. The flow is controlled by inlet valves. To protect the under-pressure in the building, the building ventilation can operate only if either one of the cell ventilation fans is operating. When the building ventilation stops, the supply fan (as well as the off-gas fans) will stop automatically.

The supply and exhaust air channel of the Actinides lab are provided with fire dampers, the exhaust air also with a spark arrester.

Off-gas ventilation

This system provides continuous air extraction through an absolute filter of the extraction boxes in the actinides lab and the G1 cell. This system has limited capacity and is intended to remove toxic fumes. The off-gas ventilation consists of 3 fans, of which at least one is in operation. The fans are not automatically connected. If a failure occurs, there is no automatic transfer. The inlet and outlet valves of the fans are manually operated. The exhaust air channel of the glove boxes is provided with a fire damper and spark arrester.

Loss of power

When a voltage dip or power failure occurs, the emergency power generator will be operating within 10 seconds. The cell, building and off-gas fans are connected to this generator. One ventilation set (1 building exhaust fan and 1 cell exhaust fan) will be activated, the off-gas will keep on running. The remaining set can be manually operated. So there is always one ventilation set active during emergency operation. Batteries (UPS) supply power for relevant control systems during the first 10 seconds of power loss. When no emergency power (generator) is available, batteries will supply power for the control system for about 10 minutes.

B. <u>Possible within the HCL-RL Building, but currently not mentioned in a procedure</u>

1. Closure of the supply fan opening at the inlet (connection to environment) by manual closing the sealing doors (kneveldeuren).
2. The fail-safe (spring closed) motor-operated isolation valves of the building and cell fans at the outlet (connection to the environment) can also be operated manually in a loss of power situation. So the outlet can also be locked.
3. The off-gas isolation valves must be manually closed when no power is available.

4. For the cell and building ventilation fans, the pressure side of the common outlet of the two fans can also be closed, especially if closing of the motor operated isolation valves at the building outlet fails. The valve can be put manually in de locked position.
5. When off-gas fails, cell ventilation fans can take over the off-gas ventilation (cell G1 to G6) by switching valves.
6. When cell ventilation fails, the off-gas can take over cell ventilation by switching valves. Because of less capacity of the off-gas system, under-pressure will not reach normal set point. The supply line of the cells has to be closed.
7. Overpressure situation due to failure of cell fans. Switch off supply fan manually when automatic switch off action fails (and if necessary closing the inlet and outlet, see above).
8. Recovery of cell ventilation in case of loss of the original power cable. A cable with plug is present in the technical room (location of the air removal fans). This cable can be connected to the cell fan motor in case of damage of the original power cable (and therefore loss of cell ventilation). The socket to be used is connected to emergency power.

Explanation:

- The first three strategies are typical for LOOP-SBO2 situations (no power at all). After performance of these actions the cell system and building are disconnected from the environment. The under-pressure in the cells will not be maintained.
- Strategies 5 to 8 are applied for various accident situations.

A.17.2 Strategies for HCL-Molybdenum Production Facility (MPF) containment function

A. <u>From existing procedures</u>

(None)

Explanation:

Brief system description

The HCL-MPF ventilation system has the same operation principles as the HCL-RL ventilation system. However, the processes in the cells of the HCL-MPF differ and the ventilation system is much more up to date. Features of the HCL-MPF ventilation system are given below.

Redundancy

Regarding the need for safety systems to be always available (with full or limited capacity), they are redundant. These systems are:

- The exhaust fans of the cells (2 x 100%)
- The exhaust fans of the building (2 x 100%)
- The supply fans (2 x 50%)
- The active carbon filter of the cells (ranging from single to six-fold redundancy)
- The pre-filters, iodine filters and absolute filters of the building ventilation system (2 x 100%).

Safety

- Fire dampers are installed in the air supply diffusers of the separate rooms. When fire is detected, the valves close automatically and the room is isolated. The fire dampers are electro-pneumatic operated, fail-safe and connected to emergency power.
- For every cell a fire damper is installed at the supply line. The fire damper closes automatically when fire is detected (can also be operated by pushing the button).
- Some cells (01 and 11) are connected to a nitrogen supply line to purge with nitrogen during processing (a nitrogen back up tank is present for emergency situations).
- In case of a large leakage in the process system, cell air can be transported through an emergency filter. An operator has to push a button to align the necessary valves.
- For every cell a filter pack is installed at the supply line in case of back flow of cell air in special accident conditions.

B. <u>Possible within the HCL-MPF Building, but currently not mentioned in a procedure</u>

1. Closure of the supply fan opening at the inlet (connection to environment) by manual closing the motor-operated isolation valve in case the fail-safe mode (=valve closed) has not been reached.
2. The fail-safe motor-operated isolation valves of the outlet, located before and after the building and cell fans (connection to the environment), can also be operated manually in a loss of power situation. In the safe mode the valves are closed.
3. Overpressure protection. Switch off the supply fan manually when the automatic action fails.

Explanation:

- Strategies 1 and 2 are typical for SBO situations (no power at all). After performance of these actions the cell system and building are isolated from the environment. Pressure in the cells will increase slowly in time.

- Strategy 3 is applied for various accident situations.

A.18 Strategies for Jaap Goedkoop Laboratory containment function

A. *<u>From existing procedures</u>*

(None)

Comment:

As it is a so called B Lab which nuclear inventory is limited, the Jaap Goedkoop Laboratory (JGL) deals with much less nuclear material as the facilities of the Hot Cell Laboratories (HCL). The ventilation system is up-to-date (4 years old) and provided with less (different) filters compared to the HCL ventilation systems.

B. *<u>Possible within the JGL Building, but currently not mentioned in a procedure</u>*

1. Closure of the supply fan opening at the inlet (connection to environment) by manual closing the motor-operated isolation valve in case the fail-safe mode (=valve closed) has not been reached.
2. The fail-safe motor-operated isolation valves of the outlet, located after the building and cell fans (exhaust, connection to the environment), can also be operated manually in a loss of power situation. In the safe mode the valves are closed.
3. Manually close the isolation valves located before the building and cell fans.

Explanation:

- The first two mentioned strategies are typical for SBO situations (no power at all). After performance of these actions the cell system and building are isolated from the environment.
- The isolation valves located before the building and cell fans are operated manually only.

A.19 Criticality studies

A.19.1 Approach and assumptions

A.19.1.1 Approach

Criticality safety is not an issue in case the total mass of fissile material in a neutronically isolated system is lower than the so-called safe mass. This safe mass is lower than the minimum critical mass. In case of uranium (up to 100% enrichment in ^{235}U) mixed with water, the minimum critical mass is 800 g, assuming full reflection by water or concrete. In case of refection by heavy water, graphite, lead or beryllium, the minimum critical mass will be significantly lower. However, this is nowhere the case at the Research Location Petten (OLP), except for the waste storage tanks in the basement of the Molybdenum Production Facility(MPF) of the Hot Cell laboratories (HCL) and the converter plate on the irradiation trolley of the Low Flux Reactor (LFR).

Generally the safe mass is set at ≤ 45% of the lowest critical mass. If the possibility of inadvertent double batching (i.e. the presence of two batches of fissile material in a facility, where only one is allowed) can be excluded, the safe mass can be set at ≤ 90% of the minimum critical mass. For the research laboratories and the STEK hall the safe mass has been set at 600 g ^{235}U. For the Research Laboratory (RL) of the HCL the safe mass has been set at 700 g ^{235}U. For other fissile isotopes conversion factors can be defined that convert the amount into an equivalent ^{235}U mass. This conversion factor can be derived from the minimum critical mass for these isotopes determined under the same conditions as for ^{235}U.

For each installation in which the maximum possible amount (i.e. either physically possible or permitted by procedures and working instructions) of fissile material present is *not* significantly smaller than the critical limit, the consequences of the postulated accidents (e.g. earthquake, flooding) are compared with the underlying assumptions and boundary conditions of the currently applicable criticality safety assessment(s), if available. This comparison may yield two possible outcomes:

- The consequences of the postulated accident are fully covered by the assumptions and boundary conditions of the criticality assessment(s). In that case sub-criticality is fully guaranteed under postulated conditions.
- The consequences of the accident are not fully covered by the assumptions and boundary conditions of the existing criticality assessment(s). In that case sub-criticality is not fully guaranteed. In such case supplemental criticality safety assessments may be justified, dependent on the feasibility or probability of the identified situation.

The underlying assumption is that the as-built/as-is configuration and the operation of the installations are in compliance with the applicable criticality safety assessments and relevant license conditions. Plant walk-downs have been performed in HCL-RL, HCL-MPF and Waste Storage Facility (WSF), in order to check this assumption explicitly. For the High Flux Reactor (HFR) and for the spent fuel storage in the HFR pool the situation is also known to be according to documentation, due to acceptance procedures carried out in the past.

A.19.1.2 Assumptions

Criticality safety assessments generally consider the presence of all possible amounts and densities of water. This is evaluated more specifically for each of the installations under consideration in subsequent sections of this appendix. Furthermore, in criticality safety assessments no credit is taken for availability of (electrical) power. Finally, it must be noted that it is implicitly assumed that fuel will not, by whatever initiating event, be relocated outside its original location (building, storage rack, vault, etc.).

A.19.2 Criticality analysis per installation

A.19.2.1 HFR: reactor core

Reactivity control of the reactor is generally achieved by 6 control elements. The HFR may only be operated if the core configuration complies with the requirements stated in Section 4 of the HFR VTS. These requirements ensure that the reactor can be made and kept sub-critical under all conceivable operational conditions by insertion of the control elements. More specifically this ensures that:

- The reactor is made sub-critical even when the two most reactive control rods are fully extracted (in absence of ^{135}Xe);
- The reactor is still sub-critical if all control rods are extracted over a distance corresponding with 50% of their respective reactivity worths (in absence of ^{135}Xe).

In case of a power failure the control rods are designed to fall into the core automatically by gravity as well as by drag force of the primary coolant, provided no change or deformation of the reactor structure has occurred, which prevents the control rods from being inserted.

In case of failure of the shutdown mechanism (ATWS – Anticipated Transient Without Scram), an alternative shutdown system (ASS) is available in the form of a cadmium plate, that has to be placed on a special Pool Storage Facility (PSF) rig, and subsequently moved towards the PSF (west) wall. Though, presently, it is unclear how much time would be needed to execute this procedure, nor whether all facilities that are currently located in the PSF can be moved out or allow for enough space to execute this procedure.

When the Cd plate is placed next to the vessel wall with a critical reactor, the reactor will become about 300 pcm (more) sub-critical. This will cause an increase in the concentration of reactor poisons (mainly ^{135}Xe and ^{149}Sm) in the fuel, which will drive the reactor even more sub-critical for a while. However, after approximately 45 hours, the reactivity level will have returned to the level at the time of the shutdown, due to the decay of ^{135}Xe. Before that moment, measures must have been taken to keep the reactor sub-critical in order to prevent a significant power excursion. So, in order to ensure long-term sub-criticality, it is planned to develop cadmium plugs that can be placed in the reactor manually.

In the above it is assumed that no change or deformation of the reactor or PSF structure has occurred which prevents the placement of the Cd plate and/or the Cd plugs.

It is recommended that the placement of the Cd plate next to the HFR vessel be tested in the current situation, and that it be appraised whether the time needed to execute this procedure is acceptable. It is also recommended that the Cd plugs for placement in the HFR core (or an equivalent system) be designed and built, or procured, and subsequently tested.

A.19.2.2 HFR: Fresh fuel storage

Fresh fuel is stored at several locations:

- K1 vault (reactor building basement): bulk fresh HFR fuel elements;
- K3 vault (reactor building basement): bulk fresh control rods.

A criticality assessment has been made for the storage vaults for fresh HFR LEU fuel and control elements (vaults K1 and K3 in the basement of the containment building, respectively). In this assessment it was conservatively assumed that the fuel element storage vault is filled with 60 LEU fuel elements and the control element storage vault with 30 LEU control elements. The modelling was conservative in the sense that several criticality-reducing features have been omitted, and some criticality-increasing features have been added (e.g. periodic boundary conditions). The possible presence of a transport cart with 6 fuel elements in front of the open vault has also been taken into account, as well as full and partial inundation.

Sub-criticality is ensured under all operational conditions as well as in case of flooding. Deformation of the vault and/or the fuel or control elements has not been taken into account.

A.19.2.3 HFR: Molybdenum production target storage

Fresh, non-irradiated targets are stored at several locations:

- K2 vault (reactor building basement): bulk irradiation targets;
- K4 vault (reactor building basement): small target storage (for direct use);
- LMH vault (Lage Montagehal): fresh experimental fissile material, cycle target load.

The handling of irradiated targets in the HFR pool and in transport to the Molybdenum Production facility (MPF) of the Hot Cell Laboratories (HCL) only concern amounts of fuel significantly less than the critical mass limit. Therefore this is not treated separately.

Fresh HEU targets currently in use for Molybdenum production (tubular targets for IRE and rectangular plates for Covidien) can be stored in three locations:

- Vault K2 in the basement of the containment building. That vault is being used to store the bulk supply of the targets.
- The vault of the LMH (Lage Montagehal). That vault is used for intermediate storage of targets required in one reactor cycle. The rectangular plate type targets for Covidien are stored in the LISTA cupboard whereas the tubular targets for IRE are stored in the side rack (zijrek).
- Shortly before irradiation targets of both types may be stored in the “target vault” (vault K4 in the basement of the containment building).

A criticality assessment has been performed of the storage facility for bulk fresh IRE and Covidien targets for Molybdenum production (i.e. vault K2 in the basement of the containment building). In that assessment it was conservatively assumed that all targets contain the maximum amount of ^{235}U, and that all containers in the storage location contained the maximum, physically possible amount of targets. Also the possible presence of a trolley with HFR LEU elements (without cadmium wires) at a variable distance was taken into account. In the analysis also the water density was varied between 0.0 (dry) and 1.0 (fully inundated) g/cm^3 to cover all possible states of inundation and, consequently, moderator-to-fuel ratios.

Sub-criticality is ensured under all operational conditions of the storage of HEU targets in the K2 vault, as well as for flooding. Deformation or relocation of the vault and/or the storage containers has not been taken into account.

The amount of fissile material in vault K4 is limited to thirty targets by organisational measures, which means that the fissile mass in vault K4 is well below the safe mass limit (see Section A.19.1.1).

The LISTA cupboard can contain 4 x 128 targets of approx. 4.8 g ^{235}U each. This amounts to approx. 2.5 kg. For the intermediate storage of targets in the LMH (LISTA cupboard and side rack) a criticality safety analysis has shown that sub-criticality is assured in all feasible operational conditions.

In the analyses to date, the four vaults K1 – K4 are treated as independent fissile material containing zones. Implicitly it has been assumed that the vaults are neutronically isolated. Given their mutual distances and configuration, neutronic coupling of these four vaults would most probably be very weak if at all existent. Therefore, no assessment of the possible effect of neutronic coupling of these vaults is deemed necessary.

A.19.2.4 HFR: Spent fuel storage racks

The standard racks in the HFR pool contain (partially) irradiated HFR fuel elements and control elements.

A criticality assessment has been performed of the storage racks for spent HFR LEU fuel and control elements in the HFR storage pool. In this assessment it was conservatively assumed that the racks were entirely filled with fresh fuel elements. Consequently also the control elements, containing a lower amount of fuel, are covered. Also cadmium wires in the fuel elements were conservatively neglected. Furthermore, in the calculations several different sets of neutronic boundary conditions (including periodic) have been taken into account. Finally, it was shown that the storage racks are neutronically decoupled, so criticality safety is also given for any number of storage racks in the storage pools.

In the analysis the storage pool was considered to be totally filled with water. Generally the multiplication factor for a totally dry storage facility would be lower. However, states with varying water densities from 0.0 g/cm^3 upward have not been specifically analysed, so that states with partial inundation may not be covered. Furthermore, a geometrical deformation of any of the structural parts is excluded from the analyses. It is recommended to analyse safety criticality of the HFR fuel storage pool for these deviating conditions.

A.19.2.5 HFR: MTR 2/33 container basket with UCW in HFR pool

When uranium containing waste (UCW) from the molybdenum production process is prepared for transport, the inner basket of an MTR 2/33 container is placed at the bottom of the HFR pool. One by one the UWC filter containers are transported from Hot Cell Laboratories (HCL) to the HFR pool and placed in the basket. When the basket is fully loaded, an MTR 2/33 container with another (empty) basket inside is hoisted into the HFR pool using the crane, whereby the hoisting route is designed to minimize the chance of contact with other objects in the pool. One by one the UCW filter containers are taken out of the basket at the bottom of the pool, and placed inside the MTR 2/33 container suspended from the crane.

A criticality safety analysis has been performed for the situation with the basket at the bottom of the HFR pool, next to the spent fuel storage rack. In that analysis, a fully loaded basket was considered at variable distance from the spent fuel racks. Also, the presence of an extra HFR fuel element next to, or on top of, the basket was taken into account as an accident scenario. That analysis covered all situations with normal water levels, without water, and all situations in between. Geometrical deformation of any of the structural parts was excluded from the analyses. The case of two partially filled MTR 2/33 container baskets was not considered. Criticality safety has been proven under these assumptions.

A.19.2.6 Low Flux Reactor (LFR): Reactor

The LFR has been permanently shut down since 10 December 2010, and the fuel has been unloaded from the reactor core and stored in the storage pits (pluggennest). Consequently, criticality is not an issue for the reactor core.

However, the reactor structure contains an irradiation facility built into the irradiation trolley. This facility includes a converter plate that contains 18 standard LFR fuel platelets (360 g of ^{235}U in total) contained in a lead box. The irradiation trolley can be moved towards and away from the reactor core. When the irradiation trolley is next to the reactor core, the lead box containing the converter plate is located next to a 4 inch lead wall and the graphite surrounding the reactor core. As the fuel elements have been removed from the LFR core and the converter plate is in fixed shielded position on the irradiation trolley, in the present situation it is certain that no other fissile material will be in close proximity of the converter plate.

This converter plate has not been assessed as part of the criticality safety analysis for the LFR fuel storage (see next section). For this specific configuration the safe mass is not easily established. According to, the minimum critical mass for various systems containing ^{235}U, water, lead and graphite, is 352 g U_{total} (for relevant enrichments). Thus, mass considerations alone are not sufficient to prove criticality safety in all feasible conditions including inundation. However, the uranium in the converter plate is distributed over widely spaced platelets firmly fixed in one plane, a configuration far apart of the dense almost spherical configurations necessary to achieve criticality with such small amounts of uranium. Therefore, excluding extreme deformations, criticality of the converter plate can be excluded.

A.19.2.7 Low Flux Reactor (LFR): Fuel storage

A criticality assessment has been made of the LFR fuel storage. In that analysis it was conservatively assumed that 29 of the 30 storepits in the storage facility (pluggennest) are filled with LFR fuel elements containing 9 fuel platelets each, and 1 with 14 LFR fuel platelets. All platelets were conservatively assumed to be fresh, containing the maximum mass of ^{235}U. That analysis covered all situations during and after inundation with water. Geometrical deformation of any of the structural parts was excluded from the analyses. Criticality safety has been proven under these assumptions.

A.19.2.8 HCL-Molybdenum Production Facility (MPF): MPF production lines and waste storage tanks

For the facilities and installations in the HCL that are not covered by a specific, dedicated criticality assessment, in accordance with Section 3.1 of the VTS of the HCL criticality can be excluded if the amounts of fuel are less than the safe masses (^{235}U equivalent) indicated in Table 1 of the VTS. These safe masses are in agreement with the minimum critical mass values for ^{235}U, as indicated in, under the assumption of full reflection by either water or concrete and exclusion of inadvertent double batching (see Section A.19.1.1).

A criticality safety assessment has been performed for the molybdenum production lines of the HCL-MPF, also taking into account the possible inundation of the cells of the production line. In that assessment it was assumed that:

- Fissile material is in solid state during storage;
- No lead is present in the cell or in the fissile material;
- Double batching is excluded by a strict administrative procedure which controls all movements of fissile material;
- No handling of fissile material is carried out in case of inundation;
- During normal storage or inundation movement of fissile material outside the zone is physically impossible;
- Cells (= zones) are neutronically decoupled (concrete wall > 40 cm);
- The separation between uranium targets in two adjacent cells is larger than 60 cm by physical means. This is ensured by the fact that fissile material is only present in cells 01 and 11, each of which is part of one of both molybdenum production lines.

Under these assumptions sub-criticality is ensured if the amount of fuel in a zone is less than the safe mass of 700 g ^{235}U equivalent. This is ensured by organisational measures.

A criticality safety assessment has been performed for the waste tanks in the basement of the HCL-MPF. In that assessment it was conservatively assumed that the waste tanks are spherical and that all tanks are surrounded by a lead reflector of 1 m thickness. This assumption also covers the case of inundation of the basement. Furthermore, all absorbing materials have conservatively been omitted.

It was shown that the safe mass is at least 395 g ^{235}U equivalent per half of the basement. Procedures are in place to monitor the amount of uranium in the waste tanks. This is reported to the manager HCL, Reactor Safety Committee (RSC) and KFD. Extrapolation of the data from the period 2009-2010 indicates that the safe mass will not be reached before 1 June 2013.

A.19.2.9 HCL-Research Laboratories (RL): storage pool with filter waste containers

A criticality safety assessment has been performed for the storage pool in the HCL transport hall. In that analysis the presence of two transport racks for UCW filter waste containers was considered, in combination with four HFR HEU fuel elements and a row of 42 fuel pins. Several accident conditions were explicitly modelled. Conservative assumptions included the amount of ^{235}U per UCW filter container (300 g), and per HFR HEU element (500 g), as well as the simultaneous occurrence of several accident conditions. Also a scenario without any water in the pool was taken into account, but the situation that the pool is partly filled with water was not studied. Geometrical deformation of any of the structural parts was excluded from the analyses. Sub-criticality was proven under these conditions. It is recommended to analyse criticality safety of the HCL-RL storage pool for these deviating conditions.

A.19.2.10 HCL- Research Laboratories (RL): Storage pipes

For storage of fissile material in the HCL-RL storage pipes a criticality safety analysis is described. The four corner storage pipes are allowed to be loaded with UCW filter containers, similar to the ones loaded into the Waste Storage Facility (WSF) pipe storage West. The analysis covered all situations during and after inundation with water. Geometrical deformation of any of the structural parts was excluded from the analyses. A procedure is in place to assure that fissile material will not be stored in locations that are not designated to contain fissile material. Sub-criticality was proven under these conditions.

A.19.2.11 Waste Storage facility (WSF): Trenches

There are two zones with fissile material in the WSF, viz. the trenches and the storage pipes. These zones have been treated independently in several criticality safety analyses, neutronic coupling between both zones has not been assessed. Given the actual configuration, such coupling of both zones would most probably be very weak if at all existent. Therefore, presently no assessment of the possible effect of neutronic coupling of these zones is deemed necessary.

A criticality safety assessment has been performed for the WSF trenches no. 1, 2, 4 and 7 under normal conditions. This analysis took account of the actual content of these trenches at the end of 2010. Flooding/inundation and deformation of structural parts were excluded. Under these assumptions criticality safety under operational conditions was proven. A supplemental criticality safety assessment was performed for the as-is situation of 31 August 2011. The main difference with the previous situation was that meanwhile all depleted and natural uranium compounds have been removed. Now, flooding was incorporated in the analysis. Sub-criticality was proven under these conditions. Again it was assumed that the fissile material remains in place, which is not assured in all accident conditions. Also deformation of structural parts was excluded in the analyses. It is recommended that the fissile material in the WSF trenches be stored in such a way that the location of the fissile material is assured under accident conditions.

A.19.2.12 Waste Storage facility (WSF): Storage pipes

Of the WSF storage pipes West, 33 of the 99 storage pipes may contain fissile material. The other storage pipes are not allowed to contain fissile material. Criticality safety analyses exist for the storage of fissile material in the 33 pipes. These cover the storage of UCW filter containers as well as storage of fissile material from other sources. The combination of the two analyses covers any possible combination of storage of the said materials in the 33 storage pipes. The analyses cover all situations during and after inundation with water. Geometrical deformation of any of the structural parts is excluded from the analyses. Criticality safety has been proven under these assumptions.

A procedure is in place to safeguard that fissile material is not placed into the 66 storage pipes of WSF storage pipes West not designated to contain fissile material. However, there are indications that these 66 'empty' storage pipes might already contain small amounts of fissile material, up to 50 g natural uranium per storage pipe, as a result of operations from long before the procedure was established. As the total amount of ^{235}U equivalent in these 66 pipes is certainly less than 25 g, which is less than the procedural safety margin taken into account for each of the loaded pipes, this small possible deviation does not warrant a supplemental criticality safety analysis of WSF storage pipes West.

In the WSF storage pipes North and South, some vessels contain α-emitting radio-isotopes. It cannot be excluded that these vessels might contain small amounts of fissile material as well. However, the amounts of fissile material must be very small to end-up unrecorded in these vessels, small enough to make a criticality safety assessment superfluous.

A.19.2.13 STEK hall

In the STEK hall the only fissile material is natural uranium as uranyl nitrate solution, containing $\leq$ 1100 g natural U (7.62 g ^{235}U). Furthermore, there will be no other fissile material stored in the STEK hall in the future. Therefore criticality is not an issue.

A.19.2.14 Jaap Goedkoop Laboratories (JGL)

In the JGL there is no fissile material. Therefore criticality is not an issue.

A.20 HFR instrumentation

A.20.1 HFR containment instrumentation

Parameter	Tag. No.:	Unit	Readout CR	Readout Local	Range (nominal)	Range (max.)	DBA qualified	Harsh environment qualified	Remarks
Under-pressure containment	PITXA-V9	mm H_2O	y	n	-50..+50	-50..+50	y	y	
Pressure containment	PIA-V20	m H_2O	DAS	RMS [1]	-0.08..+6	-0.08..+6	y	y	
Containment temperature	TE-V9	^oC	n	LBG1	-35..+70	0-100	y	y	
Containment temperature	TE-V9	^oC	n	LBG1	-35..+70	0-100	y	y	
Containment activity	RIRXSA-GM1	$Bq.m^{-3}$	y	LBG2 [2]	3*10E3	4*10E8	y	y [3]	also in CRM
Containment activity (via off-gas)	RIRXSA-GM3	$Bq.m^{-3}$	y	LBG2 [2]	1*10E4	9*10E8	y	y [3]	also in CRM
Containment activity via TM-1 monitor	RIRA-GM4	$Bq.m^{-3}$	DAS	RMS [1] [2]	1.2*10E3	5*10E8	y	y[3]	also in CRM
Containment radiation	RE-101	$Sv.h^{-1}$	CRM	CRM	1*E-6	10	y	y	
Containment radiation	RE-102	$Sv.h^{-1}$	CRM	CRM	1*E-6	10	y	y	
Containment radiation	RE-103	$Sv.h^{-1}$	CRM	CRM	1*E-6	10	y	y	
Containment radiation	RE-104	$Sv.h^{-1}$	CRM	RMS	1*E-6	10	y	y	
LBG1: Ventilation building 2nd floor									
LBG2: Ventilation building ground floor									
[1] Operation planned in 2012									
[2] Detector in LBG2 [3] Except seismic qualification									

A.20.2 HFR pool instrumentation

Parameter	Tag. No.:	Unit	Readout CR	Readout Local	Range (nominal)	Range (max.)	DBA qualified	Harsh environment qualified	Remarks
Level reactor pool	LITSA-B3	m	y	VS			y	y	
Level reactor pool	LITSA-B12	m	y	VS-CBL			y	n	
Level reactor pool	LITSA-B15	m	y	VS-CBL			y	n	
Level pool 1	LITA-B4	m	y	VS			y	y	
Level pool 2	LITA-B5	m	y	VS			y	y	
Level pool 1	LITRSA-B13	m	y	CBL			y	n	
Level pool 2	LITRSA-B14	m	y	CBL			y	n	
Pool temperature (inlet core box)	TIRTA-B1	^{o}C	y	n	0-100	-50..600	y	y	
Pool temperature (outlet core box)	TIT-B4	^{o}C	y	n	0-100	-50..600	y	y	
Pool activity	RUITA-B1	$c.s^{-1}$	y	n	0.1-10E4	0-10E4	y	y	
Radiation around pool	RE-1021	$\mu Sv.h^{-1}$	y	n	0.5-1000	0.5-1000	y	n	alarm at > 50
Radiation around pool	RE-2018	$\mu Sv.h^{-1}$	y	n	0.5-1000	0.5-1000	y	n	alarm at > 100
Camera at poolside	CAM 2 [1]		y	n			y	n [2]	
Camera at poolside	CAM 3 [1]		y	n			y	n [2]	
Camera at poolside	CAM 4 [1]		y	n			y	n [2]	
PBP: Process Monitoring Panel RBG									
VS: Valve station basement containment									
CBL: Cabinet pool level 2nd platform									
[1] pan-tilt-zoom capability									
[2] guaranteed ≤60 ºC; probably 70 ºC									

A.21 HCL-RL, HCL-MPF and JGL instrumentation

A.21.1 Instrumentation of the building pressure

Parameter	Tag. No.:	Unit	Readout GBS	Readout Local	Normal value	Alarm at	DBA qualified	Harsh environment qualified	Remarks
HCL-RL:									
Under-pressure central hall (OT)	-	Pa	y	y [1]	-80	-25	y	n [2]	
Under-pressure central cell exhaust	-	Pa	y	y [1]	-1500	-900	y	n [2]	
HCL-MPF:									
Under-pressure transport hall	-	Pa	y	y [1]	-80	-25	y	n [2]	
JGL:									
Under-pressure central hall	-	Pa	y	y [1]	-200	-150	y	n [2]	
Under-pressure laboratories	-	Pa	y	y [1]	-30	-10	y	n [2]	
Remarks:									
[1] at central panel and in plant									
[2] only local readout qualified									

A.21.2 Instrumentation of the building activity

Parameter	Tag. No.:	Unit	Alarm GBS	Readout CRM	Range	Alarm at	DBA qualified	Harsh environment qualified	Remarks
HCL-RL:									
Noble gases building		$Bq.m^{-3}$	y	y	4E6	4E5	y	n	stack
Off-gas noble gases		$Bq.m^{-3}$	y	y	9.4E7	1.4E7	y	n	stack
Aerosols breathing air		$Bq.m^{-3}$	y	y	-	α's: 2*MDA	y	n	
						β's: 100	y	n	
HCL-MPF:									
Noble gases building		$Bq.m^{-3}$	y	y	4E6	2E6	y	n	
Iodine building		$Bq.m^{-3}$	y	y	1E5	10	y	n	
JGL:									
Noble gases building		$Bq.m^{-3}$	y	y	4E6	4E5	y	n	stack [1]
[1] aerosols, I and ^{3}H are monitored off-line by grab-sampling									