The Safety of Borssele Nuclear Power Station

First report of the Borssele Benchmark Committee

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Executive Summary and Conclusions

The task of the Borssele Benchmark Committee is to determine whether the Elektriciteits Produktiemaatschappij Zuid-Nederland (EPZ) ensures that its

> *"Borssele nuclear power plant (Kerncentrale Borssele- KCB) continues to be among the twenty-five percent safest water-cooled and water-moderated power reactors in the European Union, the United States of America and Canada. As far as possible, safety shall be assessed on the basis of quantified performance indicators. If quantitative comparison is not possible for the design, operation, maintenance, ageing and safety management, the comparison shall be made on the basis of a qualitative assessment by the Committee."*

This condition is part of an agreement not to close down the plant in 2013 - as was politically intended - but to allow it, in principle, to continue operation until 31 December 2033. This agreement was formalised in a covenant, which also included the installation of the Borssele Benchmark Committee to evaluate if KCB meets this condition. The Committee had to report the results of its first evaluation in 2013.

To establish an expert opinion on the safety level of the KCB, as compared with the approximate 250 water-cooled and water-moderated reactors in the EU, US and Canada, the Committee had to develop its own methodology. There are no internationally harmonised evaluations available for all

safety aspects of a nuclear plant on the basis of which the safety can be expressed in one well-defined number. Requirements for nuclear safety are basically the responsibility of national regulatory authorities, which implies that the importance attached to various safety aspects is not necessarily uniform. The efforts of organisations, like IAEA (International Atomic Energy Agency), to harmonize these requirements withstanding national differences remain. Furthermore, opinions about what is important for nuclear safety evolve over time as a result of operating experience including root cause analyses of incidents. The speed and possibilities to adapt plants to new requirements differ. Lastly, it is debatable if the safety of a plant can be expressed in one well-defined number. Work is going on to combine all relevant safety aspects of design and operations in one model. Advanced Probabilistic Safety Analysis (PSA) would make this, in principle, possible. However, to use such a model for benchmarking the safety of the KCB, it would have to be implemented in a highly standardised way. Even when this would theoretically be possible, it would require an enormous effort, and be hindered by the unavailability of standardised plant specific information and data.

Ranking plant safety is subsequently a complicated, if not impossible, task with a time dependant outcome. Nevertheless, the Committee is convinced that it developed a meaningful methodology on the basis of all available information that could be used to compare the safety of the 250 plants the Committee had to assess. Schematically the Committee opted for the approach as shown in the figure on page 6. This methodology contains a separate safety assessment of:

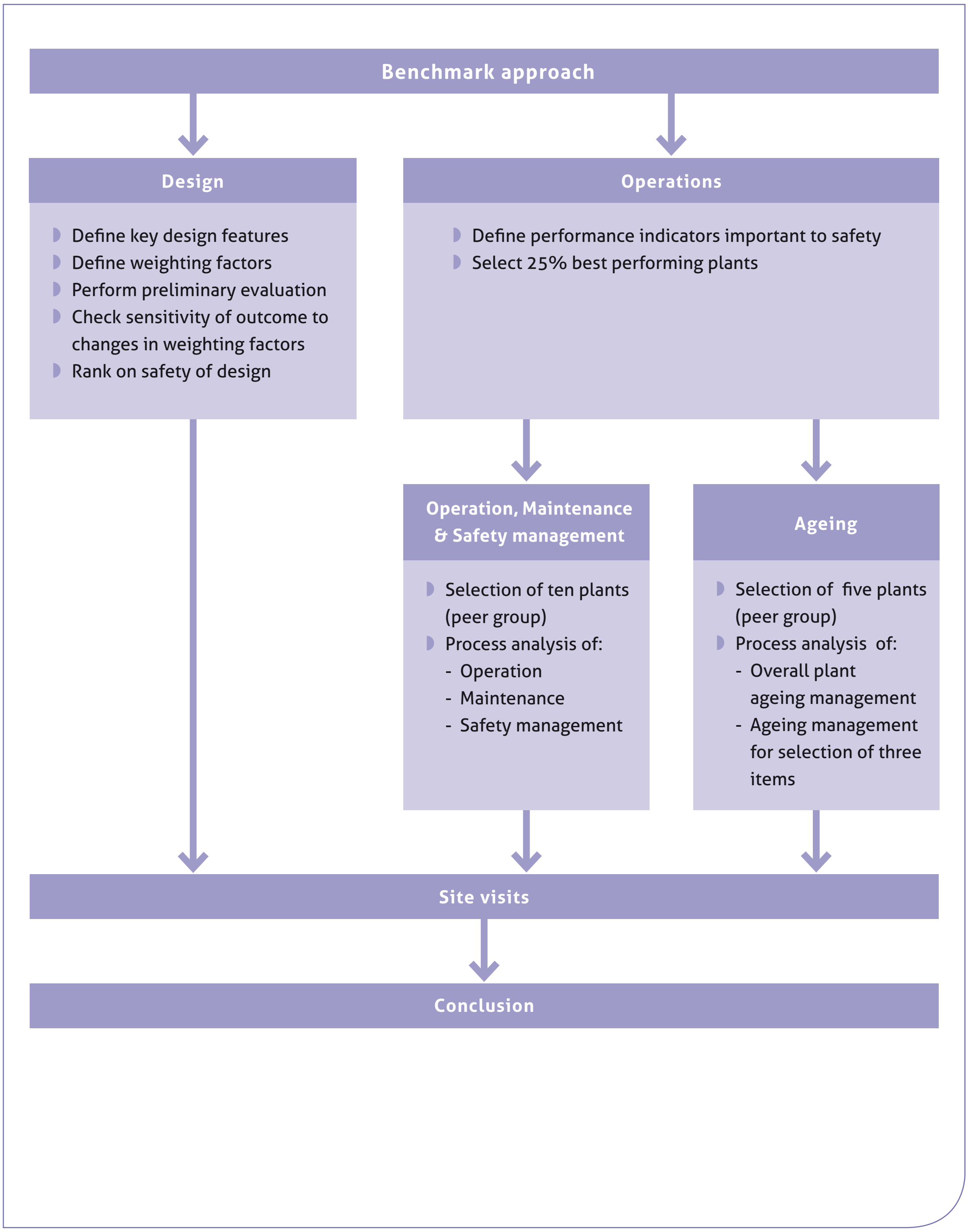
- Reactor design
- Reactor operations (covering operation, maintenance, safety management and ageing).

Using the developed methodology the committee compared the safety of the approximate 250 plants. From this assessment the committee unanimously concluded that both in design and operations the KCB is well within the top 25% safest water-cooled and water-moderated reactors in the EU, US and Canada.

So the plant meets, at this moment, the condition in the covenant regarding its safety to continue operation.

To properly explain the background of this conclusion this summary contains a fairly complete overview of the report.

Schematic approach for the benchmark



Design safety

For nuclear plant safety it is essential to assure under all circumstances:

1) Reactivity control
2) Heat removal
3) Confinement of radioactivity

The Committee discussed the contribution of design to achieve these goals, in particular regarding the plants capabilities for accident prevention, accident mitigation and containing radioactive substances within the plants' interior to reduce hazards for the environment. It decided that from this perspective the most relevant design features are:

- Redundancy and diversity of safety systems
- Design of containment
- Availability of bunkered systems
- Severe accident management

For each of these design features, design solutions were identified and scoring criteria attributed depending on their impact on safety for each type of plant.

Considering the complexity of the evaluation method, the scoring scheme was tested which lead to some improvements in the benchmarking method. In the end it gave the Committee the comfort that the methodology it developed was appropriate for the evaluation of the safety of design. All the approximate 250 plants in the benchmark were evaluated with this methodology using the large amount of available data on the design of each of those plants. This required a considerable effort. The outcome was subsequently used to identify the 25% safest plants from the design point of view.

From the results the Committee concluded that:

- The plants considered have scores in the entire range possible, with a denser population in the middle of the scale and scarcer towards both ends
- Both Pressurised Water Reactors and Boiling Water Reactors rank in the top 25% and below
- Both older and newer plants have high scores as well as low ones
- The results do not depend exclusively on one of the design features, nor are they insensitive to any of them

The scoring scheme utilised in this design benchmark places all the four features of design safety at the same level of importance. To analyze the sensitivity of the results for this assumption the Committee performed a sensitivity analysis changing the relative importance of the four features. In this analysis the importance of each of the design features was in turn reduced to half or doubled, while maintaining the same score for the other three.

This sensitivity analysis showed that:

- The group of plants in the top 25% did not change in any of the studied cases
- A few plants change their position within the top 25% but none are leaving the top

Conclusion for design

After carefully studying the results the Committee is convinced that it has developed a relevant way of ranking the safety of design for the purpose of this benchmark.
Using this method the Committee concluded that in design the KCB is well within the top 25% safest water-cooled and water-moderated reactors in the EU, US and Canada.

In the opinion of the Committee the KCB's favorable score in the design review is the result of prudent original design, but even more of continuous safety improvement programs which have taken place since 1986 as a result of periodic safety reviews.

Safety in operations

For evaluating the safety of the way plants are operated the Committee opted for a two-step approach. In the first step it decided to select the top 25% of best performing plants on the basis of performance indicators. These indicators reflect performance in the past but do not assure the same performance in the future. To cover this, the Committee concluded that it is equally important to assess, in the second step, whether safety performance is the result of well-defined and well-managed processes directed by the plant's management. Considering the amount of information needed for a detailed process analysis it is only feasible for a sample of the plants concerned. However, to determine whether KCB's performance in the management of operations is similar to that of the 25% of best performing plants in operations, it is sufficient to compare KCB in a detailed analysis with a properly selected sample.

The first step: selecting the 25% best performing plants in operations

To improve the quality of performance, the nuclear industry has instituted an internal reporting system to monitor operations on the basis of a number of performance indicators of which most are also relevant for evaluating safety. The reliability of the reporting is regularly checked in peer reviews.

On a confidential basis the Committee had access to these performance indicators and used them in its first step to select the 25% nuclear power plants with the best safety performance in plant operations of the approximate 250 nuclear power plants the Committee had to consider. To do so the performance indicators had to be combined into a composite number using weighting factors expressing their relevance for plant safety.

The first time the Committee did an evaluation of the safety performance of plant management was in 2008. Given the fact that scores in such type of monitoring systems can be substantially affected by one-off items the Committee decided to use multi-year averages.

Following the initial evaluation in 2008 it was also decided to look for trends in the outcome of the ranking by annually repeating these evaluations. From this trend analysis, it could be concluded that the top 25% on operational safety is a fairly stable population.

In all evaluations KCB was well within the top part of the 25% plants that had the best scores on the basis of these indicators.

The second step: evaluation of the plant internal processes

To evaluate if safety performance is the result of well-defined and well-managed processes directed by the plant's management, requires a lot of information about the way plants are operated. The Committee concluded that for operations, maintenance and safety management the reports from the Operational Safety Review Team (OSART) programme of IAEA would be the only appropriate available source of information for such an analysis. Ageing management is not addressed in the OSART reviews and an equivalent source of information is lacking. Considering also the fact that for a meaningful evaluation of KCB's performance in this area a different peer group would be needed the Committee decided to do process evaluation of ageing management separately.

For the process evaluation of *operations, maintenance and safety management*, a peer group of 10 plants was selected for which sufficiently recent OSART reports were available. The Committee developed a scoring system to combine the outcome of the very extensive OSART missions in a composite number indicating to what extent safety performance is the result of well-controlled processes.

The results show that the score of KCB is in the middle of the range (fifth position) of the scores obtained by the peer group. The sensitivity analysis indicates that this outcome is not affected by the choice of the evaluation methodology, or by specific "weighting" elements. Even though the input data contain a number of uncertainties related to the moment in time when the OSART mission was conducted, as well as to the uniformity and subjectivism of the evaluation, the approach taken assures that the results are robust. The middle score obtained by KCB supports the conclusion that the safety performance in plant operations, maintenance, and safety management of the KCB compares well to that of the 25% best performing plant in operations.

For the analysis of *ageing management* the Committee had to do its own reviews using Ageing Management Criteria based on IAEA guidelines. It decided to do so by comparing ageing management of KCB with that of a peer group of five plants from the 25% best performing plant in operations having an age relevant for the Borssele benchmark.

The ageing reviews was based on data supplied by the plants about:

- The plants policy, organisation, and methodology for ageing management
- Its ageing management programmes for specific components
- The plants activities for long-term operation

Following the reviews, the outcome for KCB regarding ageing management was compared with that of each of the other plants.

The evaluation lead to the conclusion that the main difference between KCB and the plants reviewed is KCB's lacking the implementation and documentation of a proper overall ageing management strategy, ageing management organisational arrangements, and methodology. KCB indicated that this will be solved at the end of 2013. If this is the case and assuming that it will fulfil the Ageing Management Criteria (based on IAEA guidelines), ageing management of KCB and of the other five plants can be considered comparable.

Conclusion for operations

The results from both the first and second step evaluation of the safety in operations indicate that overall the KCB is well within the group of 25 % best performing plants. However, its ageing management governance should be improved. According to KCB the implementation and documentation of proper ageing management governance will be completed at the end of 2013, which is also a requirement of the regulator.

Site visits

To complement its analyses the Committee visited a number of plants. Mainly to check whether the conclusions reached during the desktop reviews were supported by the impression on-site. In particular, whether the strengths and weaknesses, as compared with Borssele, that were identified in the review process are in line with the impressions obtained during the plant visit. Furthermore, to understand how specific operational safety aspects are managed at each of the visited plants compared with the way they are managed at KCB.

In total five plants from the top 25% in design and operations were visited apart from KCB. In selecting these plants attention was given to a proper geographical spread.

The site visits confirmed that, although there are differences in the way plants are managed, the operational performance of these plants is definitely the result of strictly specified and controlled processes.

A lot of attention in all plants visited is given to further improving safety performance. On the one hand by improving safety awareness and safety culture and on the other hand by incorporating the insights of probabilistic safety analyses in the management of the plants. The Committee noted that KCB is quite active in both.

The overall conclusion of the site visits was that the impressions of the Committee were in line with the expectations from the desktop reviews.

Acknowledgement

The Committee likes to express its appreciation to nuclear power plants participating in the benchmark for the support and collaboration it got for its activities. This was in particular the case during the site visits. Improving safety is very much a common goal in the industry. To achieve this there is a lot of information exchange within the industry and it is also very open for initiatives that can lead to safety improvements.

Abbreviations

AMC	Ageing Management Criteria
AMP	Ageing Management Programme
BBC	Borssele Benchmark Committee
BWR	Boiling Water Reactor
ENSREG	European Nuclear Safety Regulatory Group
EPZ	N.V. Elektriciteits Produktiemaatschappij Zuid-Nederland
EQ	Environmental qualification
EU	European Union
FAC	Flow accelerated corrosion
IAEA	International Atomic Energy Agency
KCB	Borssele Nuclear Power Plant (Kerncentrale Borssele)
LTO	Long Term Operation
NPP	Nuclear Power Plant
OSART	Operational Safety Review Team
PHWR	Pressurised Heavy Water Reactor
PSA	Probabilistic Safety Analysis
PSR	Periodical Safety Review
PWR	Pressurised Water Reactor
R&D	Research and Development
SALTO	Safety Aspects of Long Term Operation (IAEA)
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
Scram	Fast reactor shutdown
SSC	System, structure (including structural elements) or component
TLAA	Time Limited Ageing Analysis
TMI	Three Mile Island
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators Association

1

Introduction

In June 2006 the Dutch Government and the owner of the Borssele nuclear power plant (N.V. Elektriciteits Produktiemaatschappij Zuid-Nederland – EPZ) and its shareholders (N.V. Essent and N.V. Delta) agreed to terminate the operating life of Borssele nuclear power plant no later than 31 December 2033 under a number of conditions. This agreement was formalised in the "Covenant Kerncentrale Borssele"[1].

One of the conditions in the covenant (see art. 4) is that:

> *"EPZ shall ensure that Borssele nuclear power plant (Kerncentrale Borssele- KCB) continues to be among the twenty-five percent safest water-cooled and water-moderated power reactors in the European Union*[2]*, the United States of America and Canada. As far as possible, safety shall be assessed on the basis of quantified performance indicators. If quantitative comparison is not possible for the design, operation, maintenance, ageing and safety management, the comparison shall be made on the basis of a qualitative assessment..."*

This condition is usually referred to as the "safety benchmark condition".

According to the Covenant, a Committee of five independent experts, established by the covenant parties, shall assess whether or not this condition is met. The opinion of the Committee shall be reported to the covenant parties every five years, starting in 2013 and will be made public.

The Committee was established in 2008 with the following composition:

- W.K. Wiechers, former CEO of Essent (chairman), The Netherlands
- J. Pachner, principal advisor of Pachner Associates, Ottawa, Canada
- R. Stück, head Reactor Safety Analysis Division, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Köln, Germany
- B. Tomic, principal consultant at ENCO, Vienna, Austria
- A.M. Versteegh, former managing director at Nuclear Research and consultancy Group, Petten, The Netherlands

The Committee's main duties are:

- To determine whether the KCB meets the above mentioned 25% criterion specified in the Covenant.
- To assess safety in relation to design, operation, maintenance, ageing, and safety management.
- To assess safety as far as possible by reference to quantified indicators.
- In so far as quantitative comparison is not possible, to make the comparison on the basis of expert qualitative assessment.
- To carry out its duties objectively, independently of the interests of industry, civil society organisations, politics, and current government policy.

[1] Covenant Kerncentrale Borssele – juni 2006

[2] Although Switzerland is not a member of the European Union it largely follows on a voluntary basis the European regulations on nuclear safety and actively participates to European initiatives on nuclear safety. Swiss power plants were therefore included in the benchmark.

To be able to carry out its duties, the Committee needed and obtained full cooperation of the KCB and access to all documents related to safety of the KCB. In order to do this, it was assured that the confidentiality of such documents would be respected and safeguarded where needed.

This report contains the results of the first assessment of the Committee and its unanimous opinion on the basis of these results. Before going into these results, it should be emphasized that:

- The task of the Committee is not to give an absolute opinion on the safety of the KCB, but to compare its safety with that of the water-cooled and water-moderated power reactors in the European Union, the United States of America, and Canada. On the basis of that comparison, the Committee will state whether in its opinion the safety benchmark condition of the covenant is met.
- Much of the information the Committee needed could only be obtained if strict confidentiality would be ensured. For this reason the information in this report is anonymised to the level needed to ensure confidentiality.
- Considering its task, the Committee focuses only on safety aspects that are relevant for the protection of the environment surrounding the plant. Safety aspects relevant only for the consequences inside the plant are not taken into account. These consequences are considered a (economic) risk for the plant owners.

In the following chapters, the used methodology is described (chapter 2), the steps in the evaluation are explained in more detail and the results are provided (chapter 3 and 4). The findings of the site visits are described in chapter 5 and the final chapter (chapter 6) of the report presents the relation between the findings of the recent European post-Fukushima stress test and the findings of the Committee.

2

Methodology

The benchmark study covered approximately 250[2] nuclear power plants, divided into three basic types: Pressurised Water Reactors (PWR), Pressurised Heavy Water Reactors (PHWR), and Boiling Water Reactors (BWR). Figure 2-1 shows the relative fractions of these types in the total population of the plants covered by the benchmark study. A geographical distribution of the reactors is shown in Figure 2-2.

To fully quantify a safety ranking of so many different nuclear power plants is an impossible task. The safety of a plant cannot be expressed in one well-defined number. Some would argue that the Probabilistic Safety Analysis (PSA), combining both the design and operations in one model, manages to translate all into one significant plant risk level. To do this, the PSA model combines data on reliability of plant components and of the operators to determine the probability of a core meltdown, or in case of a so called Level 2 PSA, the probability of release of radioactivity into the environment.

While PSA is a recognised and highly useful tool for assessing and improving plant safety, their results are not suitable for comparison. This is mainly due to the differences in assumptions used in the model, the level of details of the model, but also the methodology and the data selected. To enable the comparison of several plants, the PSA would need to be developed and quantified in a highly standardised way, which is not the case today. Re-doing PSA in a standardized way for the purpose of the assessment within the Borssele benchmark would require an enormous effort, and even then be hindered by the availability of plant specific information and data.

Furthermore, ranking plant safety is a dynamic process with a time dependant outcome. Requirements for nuclear safety are basically the responsibility of national regulatory authorities, which implies that the importance attached to various safety aspects is not necessarily uniform. The efforts

Figure 2-1 | *Distribution of the reactor types in the benchmark population*

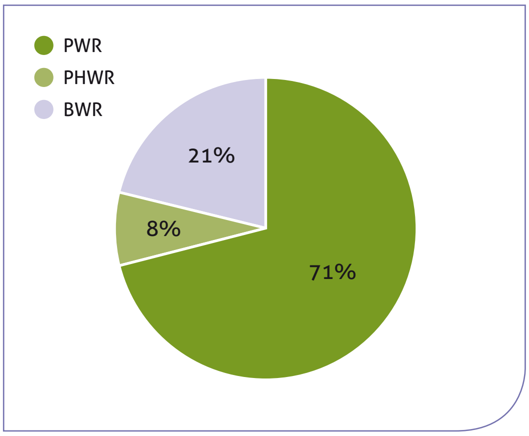
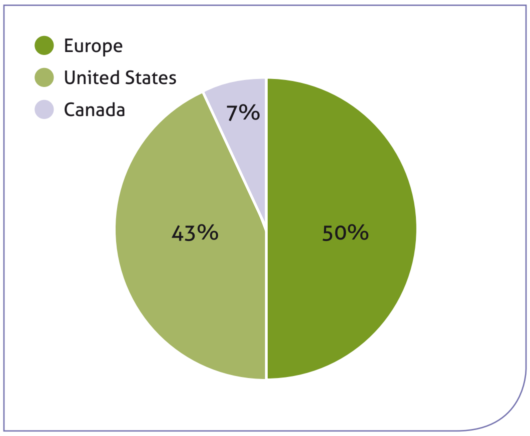


Figure 2-2 | *Geographical distribution of the benchmark population*



[2] The exact number changes over time due to the commissioning of new plants and the shutdown of existing plants, and therefore cannot be given

of organisations, like IAEA, to harmonize these requirements withstanding national differences remain. Also, opinions about what is important for nuclear safety evolve over time as a result of operating experience including root cause analyses of incidents. The speed and possibilities to adapt plants to new requirements differ too.

Taking these considerations into account, the Committee had to develop its own methodology to establish an expert opinion on the safety of the KCB, using available information on the different elements of plant's safety that could be meaningfully compared among the approximate 250 plants. It should be emphasised however that even a simplified approach is challenged by the fact that although there is a lot of information available on power plants internationally, much of it is not comparable enough for a numerical ranking of the approximate 250 plants to consider. Nevertheless, after looking into numerous reports, assessments and comparisons undertaken, which sometimes encompassed hundreds of pages of documentation, the Committee feels confident that by smart use of the available data in combination with expert qualitative assessment it is possible to determine, with sufficient confidence, whether KCB is among the safest 25% of water-cooled and water-moderated nuclear power stations in Europe, the USA, and Canada.

The committee is convinced that it developed a meaningful methodology on the basis of all available information that could be used to compare the safety of the KCB to that of the other 250 plants considered. This methodology contains, because of their different nature, a separate safety assessment of:

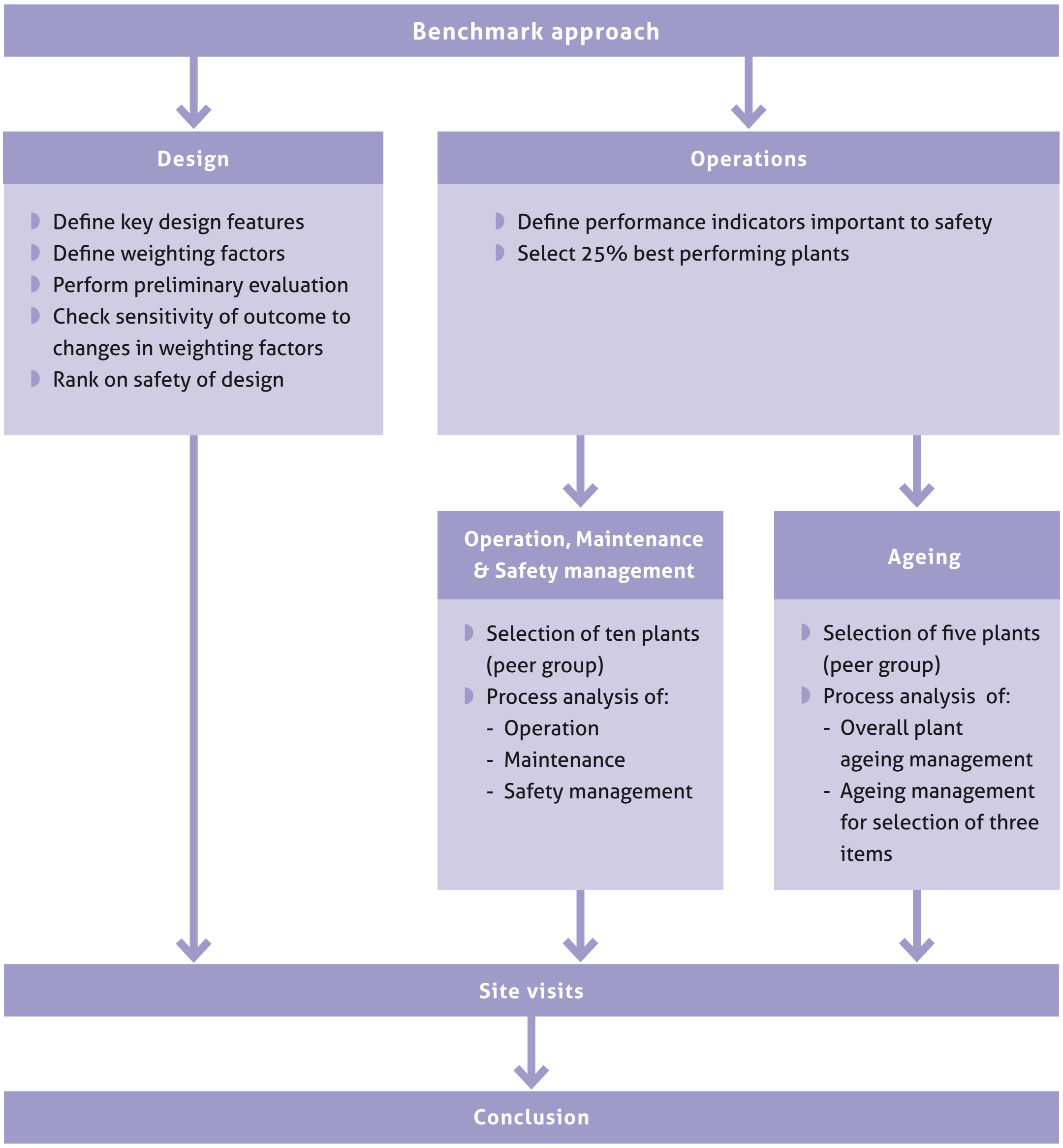
- reactor design
- reactor operations (covering operation, maintenance, safety management, and ageing)

Schematically the Committee opted for the approach as shown in Figure 2-3.

Separating the safety characteristics of design from the safety characteristics of plant operations was necessary because of their different nature. The assessment of the design was carried out for all approximate 250 nuclear power plants, based on specified key design features. For evaluating the safety of the way plants are operated the Committee opted for a two-step approach. In the first step it decided to select the top 25% of best performing plants on the basis of performance indicators. These indicators reflect performance in the past but do not assure the same performance in the future. To cover this, the Committee concluded that it is equally important to assess, in the second step, whether safety performance is the result of well-defined and well-managed processes directed by the plant's management. Considering the amount of information needed for a detailed process analysis it is only feasible for a sample of the plants concerned. However, to determine if KCB's performance in the management of operations is similar to that of the 25% of best performing plants in operations from that group, it is sufficient to compare KCB in a detailed analysis with a properly selected sample.

The results of the assessments of plant design and operations were complemented by a number of site visits.

Figure 2-3 | *Schematic approach for the benchmark*



3 Evaluation of Design Safety

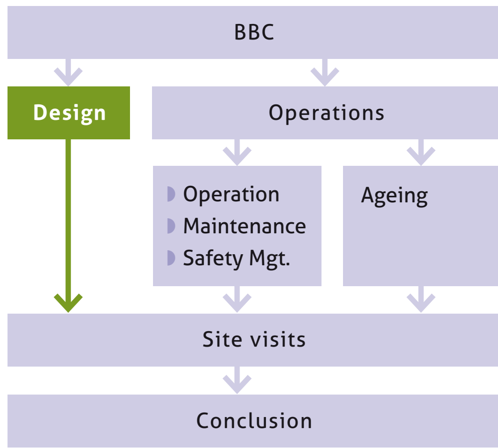
3.1 Introduction

To enable a desk top evaluation of the safety level of the plant's design, the Committee needed to identify a set of prominent features and quantify their relevance to the potential hazard of the plant for the surrounding environment. This methodology resulted in a scoring scheme.

Further implementation of this approach included:

- A pilot study on 15 plants, upon which the scoring scheme for the categorisation was tested in practice and adjusted to assure a realistic representation of the design safety level.
- The collection of design information on the operating nuclear power plants considered for the benchmark.
- The evaluation and ranking of the entire group of nuclear power plants within the scope of the benchmark, according to the refined scoring scheme.

The pilot study was performed on 15 plants of different designs and ages. These were ranked according to the initially proposed ranking scheme. This pilot study was also intended to verify the availability and obtainability of plant specific data needed for the assessment.
The plants for the pilot study were chosen to cover the entire time span during which nuclear power plants operating today in Europe and North America started operation, which means between 1969 and 2004. Furthermore, the plants were chosen in a way to include all reactor types. Plants with the same reactor type but with different suppliers were included in the pilot study too.



3.2 Definition of key design features and categories

Ranking plant design safety requires the definition of key design features and determination of their expected relevance/impact for the potential external radiological impact of the plant.

All currently operating nuclear power plants belong to the so-called "second generation". They include three basic reactor types, which are the subject of this evaluation:

- Light water-moderated reactor:
  - Pressurised Water Reactor - PWR
  - Boiling Water Reactor - BWR
- Heavy water-moderated reactor:
  - Pressurised Heavy Water Reactor - PHWR

Regardless of being developed by a number of vendor countries (USA, Germany, France, Canada, USSR), the initial safety concepts and requirements of the three above mentioned reactor types were originally designed to a more or less similar level, though in some cases (i.e. German design) advanced safety features were introduced earlier than by some other vendors. With accumulated and shared operating experience and new safety concerns (e.g. lessons from the TMI accident in the USA in 1979) both regulators and industry increased their safety demands and requirements. This resulted in diverging solutions addressing the same cause and thus in different features being added to the designs in order to enhance the safety level.

Worldwide efforts in enhancing levels of safety were intensified in the last decade, mainly by harmonization of design requirements. Through Periodic Safety Reviews (in Europe) or the Regulatory Compliance Programme (in the USA), plant characteristics were periodically checked against new safety insights and requirements. In many cases, adaptation of nuclear plants (backfitting) was required.

To assure plant safety, three fundamental safety functions need to be assured under all circumstances:
1) Reactivity control
2) Heat removal
3) Confinement of radioactivity

These fundamental safety functions remain the same for all types of light or heavy water reactors.

The starting point for the assessment reflects the most relevant design concept to assure nuclear safety, the "defence-in-depth" concept (see Table 3-1). Defence-in-depth encompasses all safety elements of a nuclear power plant, whether organisational, behavioural, or hardware related. It assures that there are overlapping or backstopping provisions, so that if a failure were to occur it would be detected, compensated for or corrected by appropriate measures. The application of the concept of defence-in-depth throughout the design and operation provides a graded protection against a wide variety of anticipated operational occurrences, design basis accidents, and severe accidents, including those resulting from equipment failure or human action within the plant and hazards that originate outside the plant.

Levels 1 and 2 of the defence-in-depth are mainly addressed by prudent design and appropriate safe operation. Both are verified by the regulatory body; design during the initial licencing process and safety in operation through regulatory inspections (oversight) and periodic safety review or other mandated regulatory checks. Within the Committees evaluation, the Level 2 defence-in-depth was assessed as a separate element; operational safety. The design-related evaluation within the Committees framework addressed Level 1 of the defence-in-depth by looking at the redundancy and diversity. Nevertheless, the focus of the Committees assessment was on safety aspects that are relevant for the environment surrounding the plant, as this, due to higher level protections, is where today's nuclear power plants differentiate between safe and very safe. To adequately capture those aspects, the assessment focussed on enhanced capabilities for accident control and accident mitigation and for containing radioactive substances within the plants' interior. These are

Table 3-1 | *Defence in depth concept (ref. IAEA INSAG-10)*

Levels of defence in depth	Objective	Essential means
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems, and other surveillance features
Level 3	Control of accidents within the design basis	Engineered safety features and accident procedures
Level 4	Control of severe plant conditions, including prevention of accident progression, and mitigation of the consequences of severe accidents	Complementary measures and accident management
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency response

mainly the elements of the level 3 and level 4 of the defence-in-depth.

The key engineered design features for control and mitigation of accidents have the following objectives:

- Control accidents to remain below the severity level postulated in the design basis.
- Control severe plant conditions and mitigation of consequences, including confinement protection.

Given this background, the Committee identified four relevant design features, which determine the safety level of the plant from the perspective of potential impact on the environment:

- Redundancy and diversity of safety systems
- Containment
- Availability of bunkered systems
- Severe accident management

3.2.1 Redundancy and Diversity

Redundancy and diversity are the major design features that have a dominant impact on the probability of events/accidents that may lead to overheating of the reactor core (e.g. due to lack of cooling) and thus so called "core damage" or "core melt".

Redundancy is referring to the multiplication of critical components or systems with the intention of increasing the reliability of the system (e.g. two, three or even four parallel pumps or systems where only one or two would be needed to fulfil the required safety function).

Diversity is referring to having different kinds of equipment to do the job to improve the availability (partly) of a given function under all circumstances e.g. electric, steam, or diesel driven pumps.

Redundancy and diversity principles are employed in all nuclear power station designs. With the same goal of assuring safety, practical solutions employed by different designers vary greatly, making their comparison relevant only from the point of view of their contribution to safety. Whether providing redundancy and diversity at the function level (different systems employing diverse operating principles capable of performing the same safety function) or at the system level (one system only with highly redundant and sometimes diverse components performing a specific safety function), the final goal of ensuring safe operation is considered to be achievable in all cases. In particular, the way redundancy and diversity is realized, differs greatly between PWRs/PHWRs and BWRs, due to the principle differences between these reactor types.

To achieve a high level of safety, redundancy and diversity have to be deployed in the design of the systems and components important for safety, as well as the associated support systems. New insights into the adequacy of redundancy and diversity need to be incorporated in already operating plants through modifications of the existing equipment (backfitting).

The Committee realises that its approach results in a rather high-level comparison. However, a more detailed analysis would require a large amount of information (which is not available and would be nearly impossible to obtain for all reactors involved), and an extremely large amount of data related to detailed systems and components and their functional relation. Nevertheless, the Committee is of the opinion that the added value of the collation of a large volume of data would not be justified by a limited added precision of the assessment.

The assessment by the Committee resulted in the following matrix (Table 3-2) for evaluating and ranking of redundancy and diversity. Although the selection and categorisation reflect international practices and many years of experience available to the Committee, there is a certain level of subjectivity in the selection of the categories above. The Committee is aware that the areas selected for consideration of redundancy and diversity could be chosen differently and that the degrees of redundancy and diversity could be defined differently.
In order to assess what influences different categorisation might have on the outcome, a sensitivity analysis was undertaken. The final conclusion was that with the methodology chosen, this effect was limited and not relevant for the final categorisation.

3.2.2 Containment

The confinement of radioactive material in a nuclear power plant, including the control of discharges and the minimization of releases, is a fundamental safety function to be ensured during normal operational modes, anticipated operational occurrences, design basis accidents and, to the extent possible, severe accidents.

In accordance with the concept of defence in depth, this fundamental safety function is achieved by means of multiple barriers and levels of defence. The containment – a strong structure enveloping the nuclear reactor – is a major factor in achieving the objectives of the third and fourth levels of defence. The containment structure also serves as protection of the reactor against external hazards.

Table 3-2 | *Scoring criteria for redundancy and diversity*

REDUNDANCY AND DIVERSITY		
	PWR / PHWR	**BWR**
1	2 x 100% [1] or less redundancy in emergency core cooling system; No diversity in auxiliary feedwater system	No redundancy in high pressure coolant injection; 2 x 100% or 3 x 50% in low pressure coolant injection; 1 x 100% in core spray
2	More than 2 x 100% redundancy in emergency core cooling system; No diversity in auxiliary feedwater system OR 2 x 100% redundancy in emergency core cooling system; Diversity in auxiliary feedwater system	Redundancy, no diversity in high pressure coolant injection; 4 x 50% or 3 x 100% in low pressure coolant injection; 1 x 100% in core spray OR No redundancy in high pressure coolant injection; 4 x 50% or 3 x 100% in low pressure coolant injection; 2 x 100% in core spray
3	More than 2 x 100% redundancy in emergency core cooling system; Diversity in auxiliary feedwater system	Redundancy and diversity in high pressure coolant injection; 4 x 50% or 3 x 100% in low pressure coolant injection; 2 x 100% in core spray

1) 2 x 100% redundancy implies that one of two system parts is sufficient to fulfil the required function.
4 x 50% redundancy implies that two of the four available system parts are sufficient to fulfil the required function.

Several containment designs are employed in the currently operating nuclear power plants:

- **Pressure suppression containments** where pressure suppression is obtained by one of the following means:
  - suppression pool
  - bubbling condenser
  - ice condenser

  Pressure suppression is reached by directing the high-pressure steam–air mixture that is generated during a pipe rupture accident into water pools (first two types), or through vent doors into chambers containing ice. The cold water or ice condenses the steam leading to decrease of the containment pressure.

- **Full pressure dry single containment** - steel shell or concrete building (cylindrical or spherical) with or without a steel liner designed to withstand the increase in pressure and temperature that occurs in the event of any design basis accident, especially a Loss of Coolant accident; the atmospheric pressure in the containment envelope is usually maintained at a negative gauge pressure during normal operations.

- **Full pressure double wall containment** - a steel or concrete shell, basically cylindrical or spherical in shape (the containment, with or without a liner) surrounded by a concrete shell (the secondary confinement); leakage from the containment is captured in the annulus between the two shells.

All the designs mentioned above provide a broadly equivalent level of protection in the case of design basis accidents. For severe accidents, it is commonly accepted that a full-pressure double wall containment would be more favourable and limit the releases in cases of accidents involving damage of the core. Further to this, double wall containments additionally designed to withstand major external impact like the crash of an aircraft would be able both to protect the core from this external impact and prevent the release of radioactivity. Such containments were considered to provide the highest safety level in this evaluation. The containment function was finally grouped into three levels, as shown in Table 3-3.

3.2.3 Bunkered systems

Hazards of internal or external origin such as explosions or fires, flooding, earthquakes, and malevolent acts have the potential to initiate events that would simultaneously affect or breach more than one safety barrier and adversely affect design features provided for mitigating the consequences. Specially designed bunkers that contain some of the key systems (like power supplies, heat removal, and basic controls) were not included in the original design of most nuclear power plants. Those were added later on to increase plant safety by assuring protection of safety systems from internal and external hazards. The bunkered systems also resulted in increase of redundancies and the solution of deficiencies (e.g. inadequate spatial separation, one of the most important protective features for internal and external hazards), if not provided for by the original design.

Initially, bunkered systems were seen as an additional redundancy, sometimes relying on the same supporting function, e.g. the water supply. Lately, more and more sophisticated systems were constructed often having multiple trains and completely autonomous power and water supply.

Table 3-3 | *Scoring criteria for the containment*

CONTAINMENT	
1	Pressure suppression containment (all types) or full pressure dry single containment
2	Full pressure double wall containment
3	Full pressure double wall containment capable to withstand a crash of an aircraft

Natural hazards were, to a different extent, considered in the initial design of nuclear power plants. Where experience or analyses later showed that additional hazards needed to be considered, safety improvements were made. Manmade hazards were also considered in some designs, like external explosions caused by nearby industrial facilities or impact of aircrafts.

In the last decade, more severe external hazards of human origin, e.g. big commercial aircraft crashes, have called for attention. Although no bunkered systems were recently added specifically for the purpose of improved safety in such cases, some plants benefit from robust design improvements implemented in the past.

Some of the older operating plants that were backfitted with bunkered systems attain a safety level equivalent to newer plants. The safety analyses performed in the last decade show that, even though not designed specifically for this purpose, some bunkered systems offer comparable protection against modern threats such as the deliberate impact of a modern long-range commercial airplane. In such conditions, the existence of properly designed safety systems capable of preventing a core melt can be equally important than the containment function.

The level to which bunkered systems can withstand conventional and modern threats is in the Committee's opinion the most important factor in defining their value to plant safety. On this basis, three safety levels were identified to be used in ranking the safety of design, as shown in Table 3-4.

3.2.4 Severe accident management

Severe accidents are generally considered to be events beyond the design basis for the generation of nuclear power plants that is in operation now. In those beyond design basis events, it is assumed that multiple failures of safety-related systems would occur, thus compromising the capability to maintain adequate cooling of the fuel and resulting in significant damage to the fuel (core melt), and possibly compromising the containment. Under certain circumstances, the containment may also be postulated to fail or to be bypassed, potentially resulting in a major radioactive release to the environment.

To enhance the protection against those beyond design basis events, nuclear power plants are

Table 3-4 | *Scoring criteria for the bunkered systems*

	BUNKERED SYSTEMS
1	Bunkered systems withstanding conventional hazards of natural and human origin.
2	Bunkered systems withstanding conventional hazards and with limited resistance against modern threats.
3	Bunkered systems withstanding both conventional and modern threats.

developing and adopting an approach called Severe Accident Management (SAM), usually represented in a form of guidelines (SAMG) to be used by operators. SAM encompasses both the equipment and the actions taken during the course of a severe accident by the plant operating staff to:

- Prevent core damage
- Restore failed equipment, or use any other available equipment to prevent or minimise the consequences of the accident
- Maintain containment integrity for as long as possible
- Minimize offsite releases

Main concerns related to severe accidents include:

- **Reactor vessel integrity**
  Severe accidents may involve core damage that progresses to the stage at which core geometry is compromised, and molten core material may relocate into the bottom of the reactor pressure vessel. The molten mass may then cause degradation of the vessel. Strategies may be employed to depressurize the reactor pressure vessel and to provide adequate cooling of the debris, from either inside or outside the reactor pressure vessel, so as to maintain vessel integrity.

- **Hydrogen control**
  Generation of hydrogen during an accident involving significant core damage is a concern for water-cooled and water-moderated reactors. Release of hydrogen into the containment can pose a threat of hydrogen deflagration or detonation. Such events could cause increase in containment pressure that could affect the containment integrity for certain containment designs.

- **Containment integrity**
  The containment building is the last barrier against the release of radioactive material to the environment in the event of a severe accident. Consequently, it is essential that the containment remains intact for a substantial period during severe accidents. Radioactive material can also be released by bypassing the containment even if the building itself remains intact. Strategies may therefore be employed to make the containment more robust against failure modes and to limit potential bypass pathways to the environment.

The severe accident management approach systematically identifies non-safety-related systems that could be employed in a severe accident, and any actions necessary to realign or reconfigure those systems in such an event, and determines the best ways in which to employ the systems' capabilities. The initial element of severe accident management is enhanced instrumentation allowing the monitoring of plant parameters and accident evolution. Existing instrumentation can be used provided it is qualified for operation under the harsh conditions of a severe accident.

Dedicated hardware and modification of existing equipment for severe accident mitigation include means like:

- Connection of the station grid to adjacent stations to provide for additional functionality of safety systems when the power is lost.
- "Feed and bleed" capability to provide for depressurisation and injection of coolant, thus compensating for loss of designated cooling systems.

- Cavity flooding to provide for cooling the reactor vessel from outside, thus preventing the failure of the vessel.
- Passive autocatalytic recombiners to provide for removal of hydrogen from the containment, thus preventing the deflagration that could compromise containment's integrity.
- Filtered venting of the containment to prevent the failure of the containment with minimal impact on the environment.

Some or all of these features were considered and prepared for functioning in nuclear power plants as part of the implementation of severe accident management concepts.

The level to which plants have equipped themselves for severe accident management, modified existing hardware or installed dedicated hardware for this purpose was the basis for defining scores for this aspect of design. The safety levels the Committee considered relevant are indicated in Table 3-5.

3.2.5 Final scoring table

The final evaluation of the safety of design, taking into account the considerations for each key feature as presented in the subsections above, was based on the Table 3-6. The scoring system using this table resulted in scores that could range between a maximum of 12 and a minimum of 4, a higher score corresponding to a higher safety level of the design.

Table 3-5 | *Scoring criteria for Severe Accident Management*

SEVERE ACCIDENT MANAGEMENT	
1	Monitoring, use of existing means, no plant specific SAMGs.
2	Monitoring, use of existing means following plant specific SAMGs.
3	SAMGs and dedicated hardware or modified existing hardware.

Table 3-6 | *Final scoring criteria for the of the safety of design*

REDUNDANCY AND DIVERSITY		
	PWR / PHWR	BWR
1	2 x 100% or less redundancy in emergency core cooling system; No diversity in auxiliary feedwater system	No redundancy in high pressure coolant injection; 2 x 100% or 3 x 50% in low pressure coolant injection; 1 x 100% in core spray
2	More than 2 x 100% redundancy in emergency core cooling system; No diversity in auxiliary feedwater system OR 2x100% redundancy in emergency core cooling system; Diversity in auxiliary feedwater system	Redundancy, no diversity in high pressure coolant injection; 4 x 50% or 3 x 100% in low pressure coolant injection; 1 x 100% in core spray OR No redundancy in high pressure coolant injection; 4 x 50% or 3 x 100% in low pressure coolant injection; 2 x 100% in core spray
3	More than 2 x 100% redundancy in emergency core cooling system; Diversity in auxiliary feedwater system	Redundancy and diversity in high pressure coolant injection; 4 x 50% or 3 x 100% in low pressure coolant injection; 2 x 100% in core spray
CONTAINMENT		
1	Pressure suppression containment (all types) or full pressure dry single containment	
2	Full pressure double wall containment	
3	Full pressure double wall containment capable to withstand a crash of an aircraft	
BUNKERED SYSTEMS		
1	Bunkered systems withstanding conventional hazards of natural and human origin	
2	Bunkered systems withstanding conventional hazards and a certain limited resistance against modern threats	
3	Bunkered systems withstanding both conventional and modern threats	
SEVERE ACCIDENT MANAGEMENT		
1	Monitoring, use of existing means, no plant specific SAMGs	
2	Monitoring, use of existing means following plant specific SAMGs	
3	SAMGs and dedicated hardware or modified existing hardware	

3.3 Pilot study results

A pilot study was used to test and adjust the scoring scheme for the categorisation in practice in order to assure a realistic representation of the design safety level. The pilot study led to some improvements in the ranking method, and in the end it gave the Committee comfort that the methodology it developed was appropriate for the evaluation of the safety of design. From the pilot study the Committee concluded that:

- The plants selected have scores in the entire range possible, with a denser population in the middle of the scale and scarcer towards both ends.
- The scoring system is not sensitive to the reactor type; both Pressuruized Water Reactors and Boiling Water Reactors rank in the top 25% and below.
- The scoring system is not sensitive to plant age; both older and newer plants have high scores as well as low ones.
- The results do not depend exclusively on one of the design features, nor are they insensitive to any of them.

These conclusions were reconfirmed in the final design benchmark.

3.4 Final results

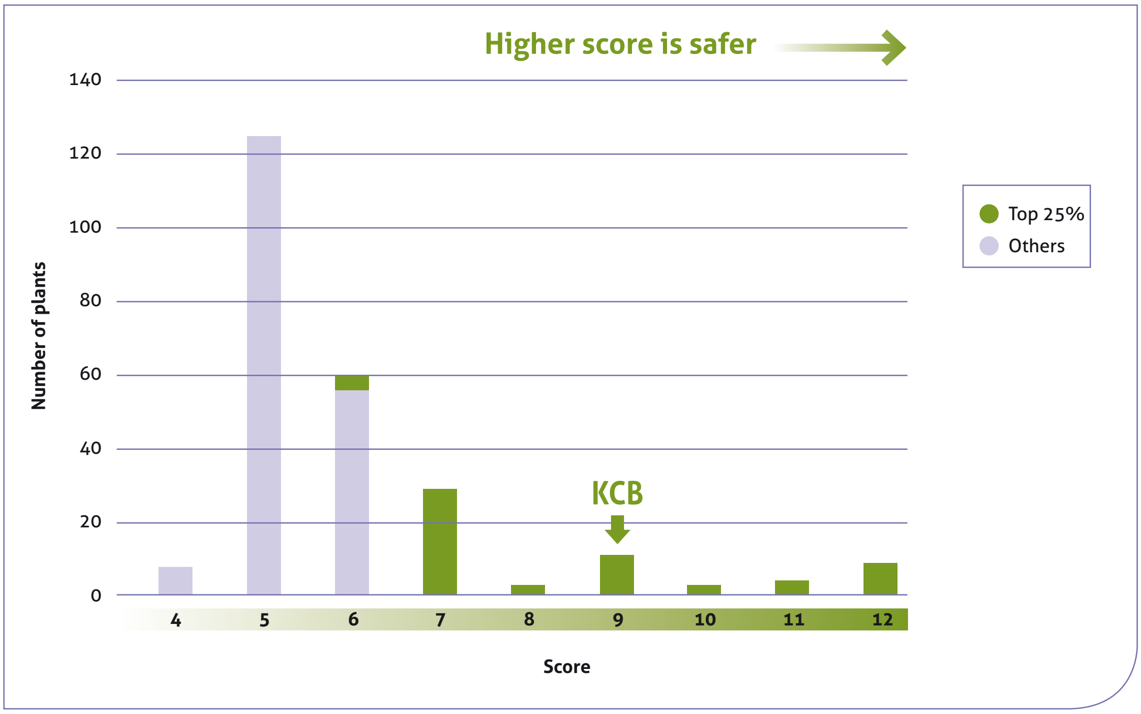
A graphical representation of the distribution of plants among the score categories is given in Figure 3-1. This figure also shows the relative fractions of the major plant types.

Figure 3-1 | *Distribution of plants of different types within the score categories*

Scores assigned to the plants cover the range of 4-12, with a denser population in the low-middle part of the scale range (score categories 5-7) and scarcer towards both ends (fractions of plants with the score of 4 and 12 are ~3.2% and ~3.6%, respectively). The majority of plants have the score in the range of 5-6 (total fraction with his score range is 73%).
The categorization chosen by the Committee enhances the differentiation between safe and very safe, resulting in a non-linear distribution of the results: most plant score in the range 4-6 ("safe") while the score of the top 25% ("very safe plants") spreads over the range 6-12.
The scoring system is insensitive to the reactor type. Pressurised Water Reactors as well as Boiling Water Reactors are distributed equally among the broad range of the score categories. Pressurised Heavy Water Reactors are in the "safe" range.
The top 25% fraction of the plant population (~60 plants) basically included the plants with a score from 12 to 7, as shown in Figure 3-2. KCB scores 9 points, ensuring a place well within the top 25%.

The scoring scheme utilised in the benchmark places all the four features of design safety at the same level of importance. To analyse the sensitivity of the results for this assumption the Committee performed a sensitivity analysis changing the relative importance of the four features. In this analysis the importance of each of the design features was, in turn, reduced to

Figure 3-2 | *Distribution of plant scores and top 25% group for design safety*



half or doubled, while maintaining the same score for the other three.

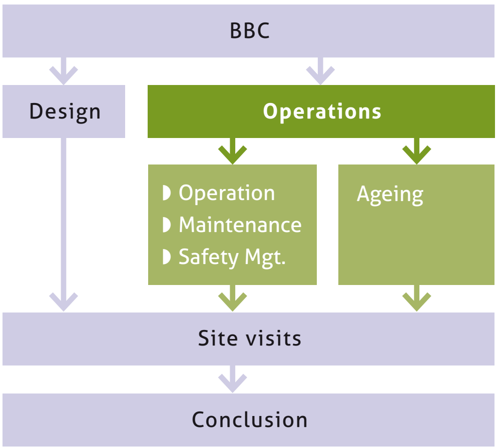
This sensitivity analysis showed that:

- The top 25% does not change in any of the studied cases.
- A few plants change their position within the top 25% but none are leaving the top.
- KCB rises one position in one case and descends one or two positions in the other seven cases but keeps its score of 9 points thus remains well within the top 25%.

The results of the sensitivity study show, thus, that the benchmark results for design safety are not driven by any single design feature of the four considered, but by their combination.

4

Evaluation of Operational Safety



For evaluating the safety of the way the plants are operated the Committee has opted for a two-step approach. In the first step it decided to select the top 25% of best performing plants on the basis of performance indicators. These indicators reflect performance in the past but do not assure the same performance in the future. To cover this, the Committee concluded that it is equally important to assess, in the second step, whether safety performance is the result of well-defined and well-managed processes directed by the plant's management.

4.1 First step evaluation of operational safety

4.1.1 Introduction

The first step of the Operational Safety Benchmark focuses on the selection of the top 25% of best performing plants against which KCB is to be compared. For this selection a set of internationally accepted performance indicators is applied.

To improve the quality of performance, the nuclear industry has instituted an internal reporting system to monitor operations on the basis of a number of performance indicators. The reliability of the reporting is regularly checked in peer reviews. Most of these performance indicators are relevant for evaluating the safety performance in plant operations. It includes the following indicators:

- **Unit Capability Factor**
  This performance indicator is generally accepted in the utility industry to indicate the effectiveness of plant programs and practices in maximising the electrical power generation. It provides an overall indication of how well plants are operated and maintained.

- **Forced Loss Rate**
  The outage time and power reductions that result from unplanned equipment failures, human errors, or other conditions during the operating period (excluding planned outages and their possible unplanned extensions) are a good indicator for the effectiveness of plant programs and practices in maintaining systems available for safe electrical generation when the plant is expected to be at the grid dispatcher's disposal.

- **Unplanned Automatic Plant shutdowns (scrams)**
  The number of unplanned automatic scrams is a generally accepted indicator to

monitor plant safety. It includes the number of undesirable and unplanned thermal-hydraulic and reactivity transients that result in reactor scrams, and thus gives an indication of how well a plant is operated and maintained. Manual scrams and, in certain cases, automatic scrams as a result of manual turbine trips to protect equipment or mitigate consequences of a transient are not counted because operator initiated scrams and actions to protect equipment should not be discouraged.

- **Safety System Performance**
  Monitoring the readiness of important safety systems to perform their functions in response to off-normal events or accidents, gives insight in the effectiveness of operation and maintenance practices.

- **Fuel Reliability Indicator**
  Failed fuel represents a breach in the initial barrier preventing off-site release of fission products. Failed fuel also increases the radiological hazard to plant workers.

- **Chemistry Performance Indicator**
  This indicator monitors the concentrations of important impurities and corrosion products in selected plant systems to give an overview of the relative effectiveness of plant operational chemistry control.

- **Collective Radiation Exposure**
  Collective radiation exposure to plant workers is an important indicator for the radiation exposure within the plant and the effectiveness of radiological protection programs.

- **Industrial Safety Accident Rate**
  Industrial safety accident rate was chosen as the personnel safety indicator over other indicators, such as injury rate or severity rate, because the criteria are clearly defined and most utilities currently collect this data.

4.1.2 Selection of the top 25% of best performing plants

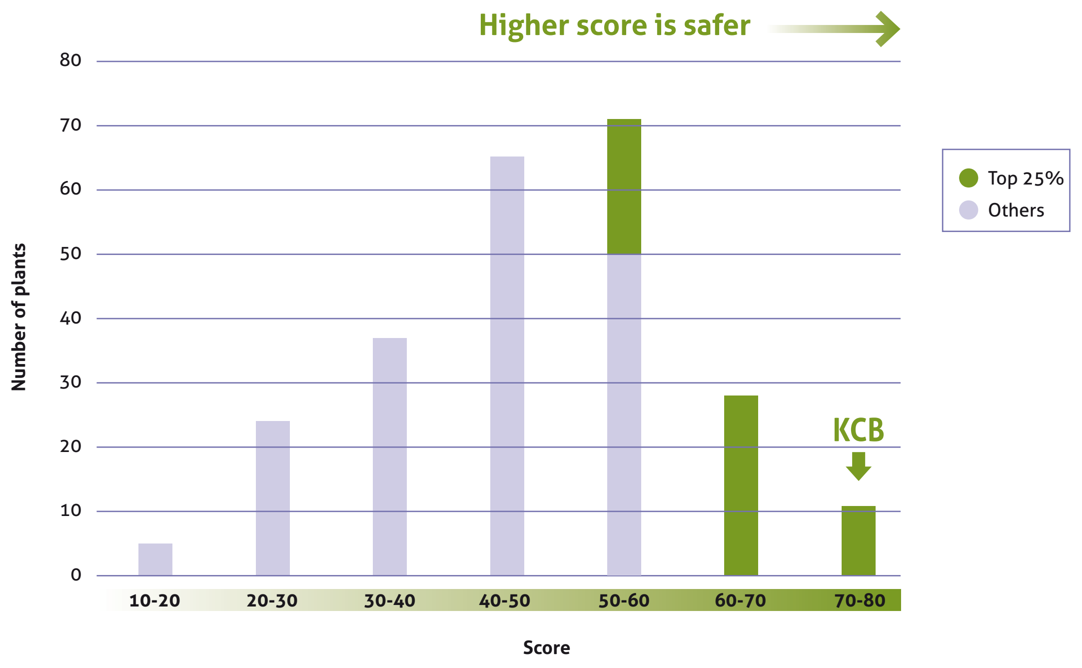
On a confidential basis the Committee had access to the performance indicators indicated above and used them to define the top 25% of plants with the best safety performance in plant operation of the approximate 250 nuclear power plants the Committee had to consider. To do so the performance indicators had to be combined into a composite number using weighting factors expressing their relevance for plant safety.

The first time the Committee did an evaluation of the safety performance of plant management was in 2008. Updates were provided up to 2012. Considering the fact that scores in such type of monitoring systems can be substantially affected by one-off items, the Committee decided to use multi-year averages.

In all evaluation updates KCB was in the top part of the 25% plants that had the best scores on the basis of these indicators. The results for the latest evaluation are given in Figure 4-1, which gives the number of plants in a certain scoring range. A higher score (horizontal axis) is related to a better result in the evaluation.

With a score just above 70, KCB is well within the top 25% plants with the best performance in the safety of plant operation on the basis of performance indicators, as shown in Figure 4-1.

Figure 4-1 | *Distribution of plant scores and top 25% reference group for operational safety*

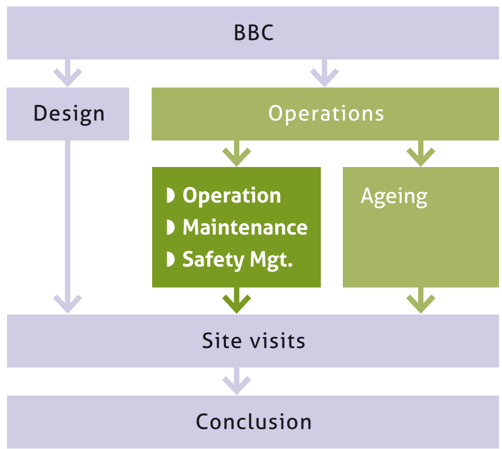


The Committee was aware that although these indicators cover a broad range of aspects that are of importance for the safety of plant management, such an evaluation could only be considered as a first level indication of the performance of the KCB compared within this group. A more in-depth evaluation was needed to obtain insight into whether KCB's performance is the result of a well-controlled process. To do so the Committee performed an in-depth process analysis of the performance of the nuclear power plants. In this study it looked separately to the elements of relevance including Operation, Maintenance, and Safety Management on the one hand and Ageing Management, on the other hand. Considering the nature of this process evaluation (see section 4.2 and 4.3) it was deemed sufficient to compare KCB in a detailed analysis with a properly selected sample of the group of best performing plants.

4.2 Operation, Maintenance and Safety Management

4.2.1 Introduction

The process analysis in the second step of the Operational Safety Benchmark focuses on the extent to which the safety performance of a plant is the result of a well-controlled process directed by the plant's management. Such an analysis requires good understanding about the way the plants are operated and managed. The Committee concluded that for a process analysis of operation, maintenance, and safety management the only appropriate derestricted information are the available reports from the Operational Safety Review Team (OSART) programme of IAEA.



Under the OSART programme, a team of experts conducts an in-depth review of operational safety performance, addressing the issues that are affecting the management of safety and the performance of personnel. It is important to stress that the OSART's, while being peer reviews (team members are typically senior management of nuclear power plants or regulatory bodies) are conducted to the same set of guidelines and to the unique criteria, those being the standards and guides of the IAEA. By identifying problems and areas of concern, the OSART programme provides advice and assistance to the nuclear power station management on enhancement of the operational safety.

In addition, the OSART programme provides an opportunity to disseminate information on "good practices" which are recognised during OSART missions.

The OSART missions review performance in the following areas:

- Operations
- Maintenance
- Technical support
- Radiation protection
- Chemistry
- Operating experience
- Emergency planning and preparedness
- Management, organisation, and administration
- Training and qualification
- Commissioning

The result of an OSART mission is a report presenting the team's observations and conclusions. It includes the discussion and references to all recommendations, suggestions, and good practices identified by the team. The OSART report is (automatically) derestricted ninety days after its issuance, except if the host country requests otherwise.

Due to the level of details and its coverage, its high professional assessment as well as using a unified set of criteria, the OSART reports constitute an adequate basis for benchmarking the safety performance in operations, maintenance, and safety management of KCB against its peers.

Ageing management is not addressed in the OSART reviews and an equivalent source of information is lacking. Considering also the fact that for a meaningful evaluation of KCB's performance in this area a different peer group would be needed, the Committee decided to do process evaluation of ageing management separately.

4.2.2 In-depth review of operation, maintenance, and safety management

While the comparison based on (numerical) performance indicators is rather straightforward, a process evaluation implies understanding of the philosophy of operation of a nuclear power plant and the organisational, management, and operational practices that may vary significantly across the countries and operating organisations.

The Committee decided that such an evaluation would require:

a) Consideration and evaluation of all findings of the OSART: strengths (good practices) and weaknesses (areas for improvement).
b) An assessment by judging the 'importance to safety' of each OSART finding.
c) A ranking of KCB against the other plants in the peer group. The results of the ranking were expressed in relative terms, equally, better or worse than each peer.

The nuclear plants that were included in this detailed evaluation were selected using a number of criteria:

- Good geographical spread over the benchmark area: European Union, USA, and Canada.
- High score on operational performance, preferably ranking in the top 25% on the basis of performance indicators.
- OSART mission in the recent years for which the report is publicly available.

The final selection of 10 peers was based on the expert opinion of the Committee and, in view of the desired geographical spread, included three plants outside the top 25% group.

Out of the peer group of ten nuclear power plants, the Committee selected three plants for a pilot study, in order to verify the applicability of the benchmarking method developed by the Committee. The plants selected for the pilot study have different reactor designs, and also different ranking based on the performance indicators. After the methodology was defined and tested in the pilot study, the remaining seven nuclear power plants and KCB were also evaluated.

Categorization and classification of OSART findings

The methodology chosen for the benchmark involved the operational safety of a nuclear power station as seen by the expert judgment of the OSART team and the Committees' judgment on the 'importance for safety' of the OSART findings. The latter involved identifying suitable parameters to categorize the plant's strengths (e.g. good practices) and weaknesses (e.g. areas for improvement) as indicated by the OSART. First of all we decided not to use the "good practices" in our ranking scheme. Subsequently, all other OSART findings for each of the 11 plants (peer group and KCB) needed to be classified by their importance for safety. A relative large number of approaches to evaluate the importance of findings for nuclear safety, considering the safety management, defence in depth, and safety culture was studied. On the basis of this study, a structure to objectively categorize the safety significance of each OSART finding (recommendation, suggestion or note) was proposed. The categorization consists of five groups with a decreasing safety importance:

Group I

- **Overall safety management**
  Findings categorized in this group would be those related to the managerial aspects of safety. This includes findings related to the management of plant programs and activities that are impacting safety, including plant organisation, safety assessments and reviews, risk evaluations, procedures and training for the management and supervisory personnel, reporting and corrective actions, including use of operational experience feedback, etc. Because of its cross-cutting potential to weaken the overall operational safety performance (i.e. multiple safety barriers could be affected) this group was given the highest weighting factor. Findings in this group could be an indication of a weak overall operational safety performance.

Group II

- **Plant operation during normal and abnormal situations**
  Findings categorized in this group would be those where plant safety has been challenged, including plant's compliance with its operational limits and conditions and/or its ability to withstand deviations from normal operation. These findings cover issues such as the competence and skills of operators, operating practices, the status of systems and components, the quality of procedures, and adequacy of their usage. The findings in this group may be an indication of deficiencies affecting equipment and personnel, undermining prevention capabilities and/or plant safety. This is the reason that findings in this group have the second highest weighting factor.

- **Human performance**
  Operational experience from the nuclear industry demonstrates that 70 % of events in nuclear power stations are caused by inadequate human performance. Findings related to human factors or performance may be an indication of weakened safety, and are thus very important for the overall safety of the plant. Therefore the findings in this group were also given the second highest weighting factor. The issues in this group include a range of issues from training and qualifications to performance and rectification of identified deficiencies. All findings regarding human performance are included in this group.

Group III

- **Functioning of plant systems and equipment, plant integrity**
  The findings in this group are related to the functioning of plant's systems and equipment and/or integrity of plant's structures. As such, those are providing the support for safe operation of the plant. Findings in this group are related to equipment maintenance programme,

engineering support activities, and other specialized programmes, including e.g. equipment qualification, fire protection, chemistry control, etc. Being a support rather than a front line function, the findings of this group receive lower rating than the previous group.

- **Management of deviations and failures**
  OSART missions typically look into the conduct of preventive activities at a plant, thus identifying deficiencies related to control of deviations and/or failures of plant systems and equipment before those would lead to more serious situations. Examples of findings include operational issues, ability to timely identify the faults and deficiencies related to surveillance procedures. Being of a preventive nature, findings in this group receive lower rating than the previous one.

Group IV

- **Personnel safety**
  One element of the OSART is devoted to the assessment of the radiation protection and industrial safety programmes. Despite the fact that these aspects are important safety elements, their impact is primarily affecting plant personnel. As the focus of the assessment, within the Borssele Benchmark, is on the impact on the public and environment, the findings within this group could be assessed as being less significant than those belonging to groups I-III.

- **Emergency preparedness**
  The basic principle of nuclear safety is to operate the plant in such a manner to exclude the potential impact on the public and environment. In the unlikely case of a radioactive release to the environment, the direct threat to population and environment is minimized through adequate emergency planning and preparedness, which generally is the responsibility of off-site authorities. OSART looks into the on-site emergency preparedness. Any findings in this area are not directly related nor an indication for the overall safety status of the plant. Therefore, similarly to those related to personnel safety, findings in this area could be considered less significant than those belonging to groups I - III.

Group V

- **Insignificant issues**
  There are comments in the OSART reports related to different aspects of plant operations that do not relate to or have significant impact on the plant safety level. Such type of findings is primarily meant to be opportunities for enhancement, rather than an indication of safety challenges. Therefore, the findings of group V do not warrant consideration in the ranking scheme (i.e. the impact is considered null).

Further to the weighting factors for each of these five groups, a second categorization of significance for plant safety was added, based on the OSART categorization of the issues in:

- **Recommendations; R** - being a very significant finding, deserving prompt rectification
- **Suggestions; S** - being a finding where management might consider making a change
- **Notes; N** - being a remark not obliging plant management to act

The Committee also considered that from the point of view of their potential impact, the issues for which recommendations, suggestions or notes are made can vary in significance. A third layer of classification was therefore introduced to account for the contribution of each issue to a safety performance. This classification was made based on expert judgment and includes three levels: high (H), medium (M) and low (L) safety significance.

This threefold categorization and classification is represented in the resulting ranking matrix (Table 4-1), which combines all the three above discussed levels, of which the most important is the one reflected by the 5 groups of evaluation criteria, the second is the OSART categorization reflected within each of the groups, and the third is the consideration of the impact on the safe plant operation of each individual issue.

Table 4-1 | *Final Ranking Matrix*

Criterion	Value	Issue Type	Significance		
			High	Medium	Low
Group I 1. Overall safety management	**4**	**R** Score	**100%** 4	**80%** 3,2	**60%** 2,4
		S Score	**50%** 2	**35%** 1,4	**20%** 0,8
		N Score	**15%** 0,6	**10%** 0,4	**5%** 0,2
Group II 2. Plant operation during normal and abnormal situations 3. Human performance	**3**	**R** Score	**100%** 3	**80%** 2,4	**60%** 1,8
		S Score	**50%** 1,5	**35%** 1,05	**20%** 0,6
		N Score	**15%** 0,45	**10%** 0,3	**5%** 0,15
Group III 4. Functioning of plant systems and equipment, plant integrity 5. Management of deviations and failures	**2**	**R** Score	**100%** 2	**80%** 1,6	**60%** 1,2
		S Score	**50%** 1	**35%** 0,7	**20%** 0,4
		N Score	**15%** 0,3	**10%** 0,2	**5%** 0,1
Group IV 6. Personnel safety 7. Public and environment	**1**	**R** Score	**100%** 1	**80%** 0,8	**60%** 0,6
		S Score	**50%** 0,5	**35%** 0,35	**20%** 0,2
		N Score	**15%** 0,15	**10%** 0,1	**5%** 0,05
Group V Insignificant/out of scope issues	**0**				

4.2.3 Results

The above evaluation matrix was tested in the pilot study. Its aim was to assess the robustness of the methodology as a whole and of the system for classification in particular. Upon completing the assessment on each of the three plants in the pilot study, the results were reviewed by an independent expert who was familiar with the status (and operational safety) of each plant. The expert concluded that the results obtained are in line with the general perception of the OSART teams when visiting the plants. This assessment gave confidence that the proposed approach and the methodology would be appropriate for the utilization for the Borssele benchmark. Following the pilot study the remaining seven peers and KCB were then evaluated using the same methodology.

The results of the evaluation are presented in the graph in Figure 4-2.

4.2.4 Sensitivity analysis

To confirm the adequacy of the benchmark methodology, the variation of the results with changes in the parameters that have potential for significant impact on the results, was studied. The sensitivity analysis was performed twice, first for the pilot study plants, and secondly, reconfirmed for all plants. In particular it was studied whether taking the OSART Notes into account would have a large influence on the ranking. In Figure 4-3 the

Figure 4-2 | *Results of the evaluation*

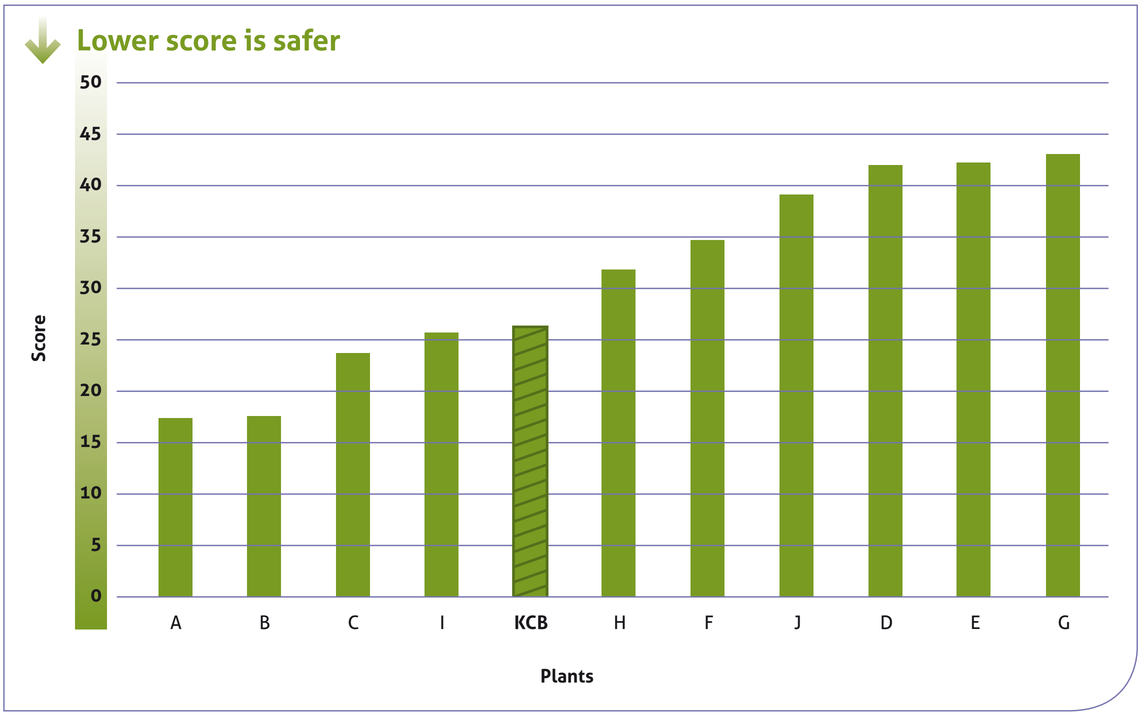
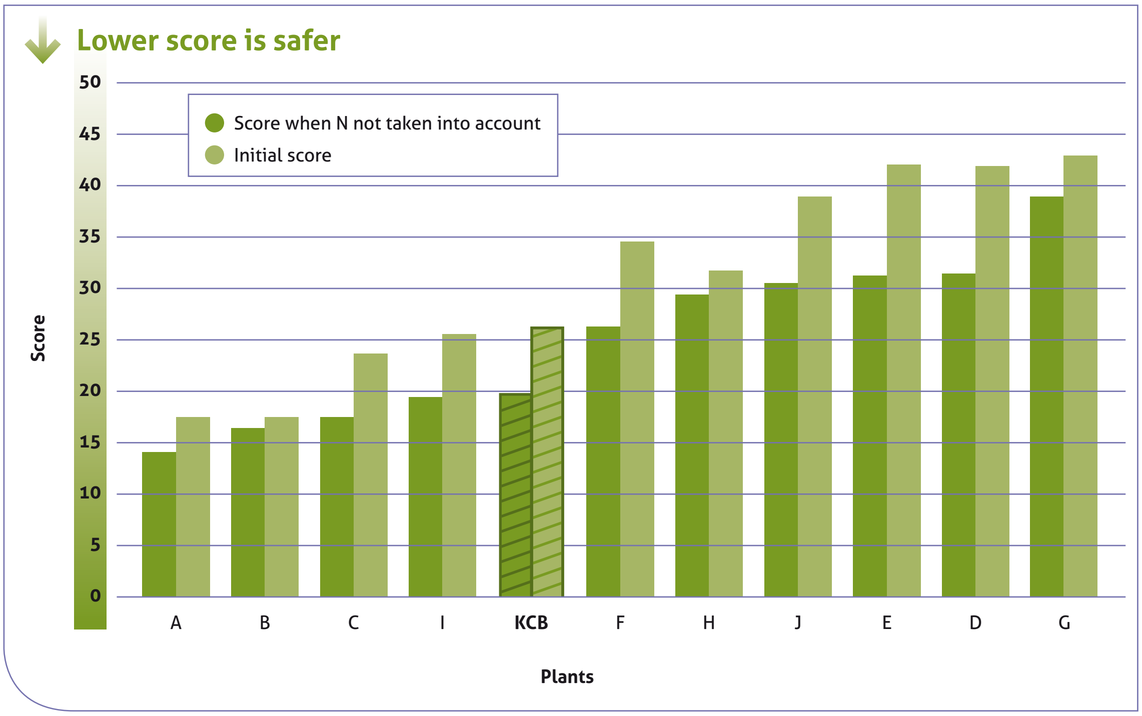


Figure 4-3 | *Results of the evaluation without Notes*



results are given if Notes are not taken into account.

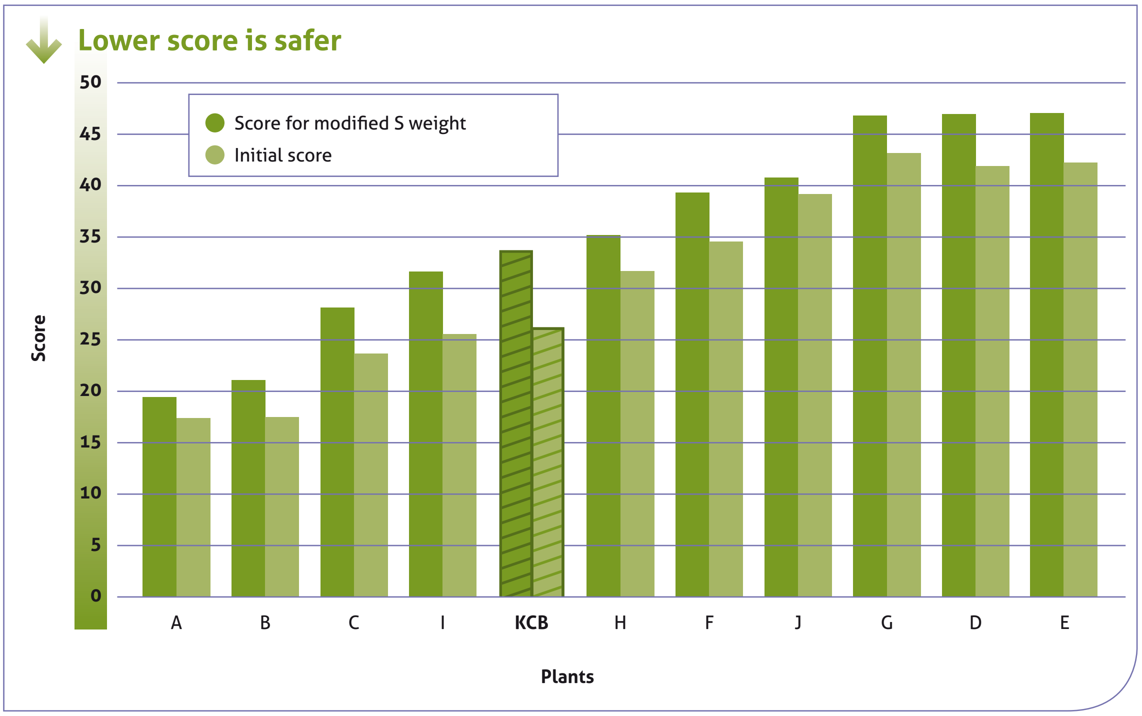
It was concluded that this does not have a large influence. Although some plants change place in the ranking, the net effect is not significant.

The second parameter that was thought potentially to have significant impact on the results was the importance of Suggestions relative to the Recommendations, which were considered to be the most important conclusions of the OSART missions. For this sensitivity analysis the significance of the Suggestions was increased in the evaluation scheme (see Figure 4-4).

This resulted of course in higher scores for all plants, but their ranking remains basically the same. This outcome shows that the ratio between the significance of the Recommendations and that of the Suggestions is not a dominating parameter of the evaluation scheme.

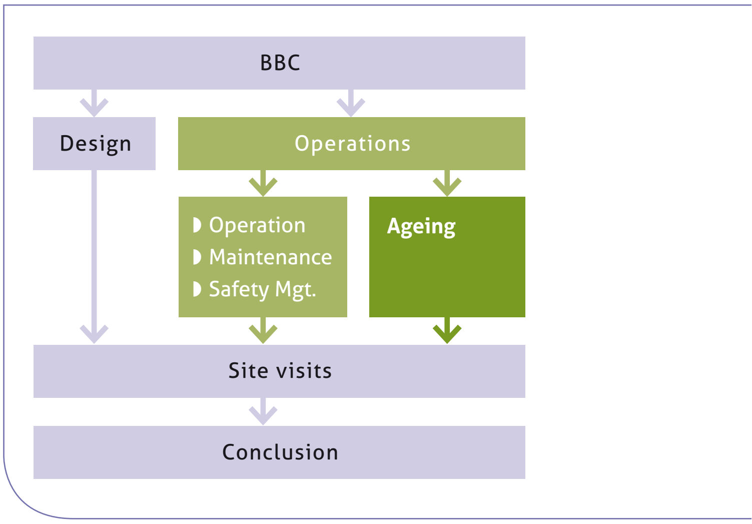
The purpose of detailed analysis of KCB's performance in control, maintenance and safety management was to obtain a credible insight into whether KCB's safety performance is the result of a well-managed process, when compared to its peers. The approach combines all the aspects of operational safety into a composite judgement. Regardless of specific management or technical approaches that a plant might follow, it focuses on the results/outcomes, thus avoiding impact of the technology or plant's organisation or management structure.

Figure 4-4 | *Results of the evaluation with increased significance of Suggestions*



The results show that the score of KCB is in the middle of the range (fifth position) of the scores obtained by the 11 peer nuclear power plants. The sensitivity analyses indicate that this outcome was not affected by the choice of the evaluation methodology, nor by specific "weighting" elements. Even though the input data contain a number of uncertainties related to the moment in time when the OSART mission was conducted, as well as to the uniformity and subjectivism of the evaluation, the approach taken assures that the results are robust. The middle score obtained by KCB supports the conclusion that also from a process management point of view, the safety performance in plant operations, maintenance and safety management of KCB is comparable to that of the selected peer group.

4.3 Ageing



4.3.1 Introduction

Ageing refers to the general process in which characteristics of a system, structure, or component gradually change with time or use. Examples of ageing mechanisms include curing, wear, fatigue, creep, erosion, microbiological fouling, corrosion, embrittlement, chemical or biological reactions, and combinations of these processes e.g. erosion-corrosion, creep-fatigue. Since ageing can have impact on both nuclear power plant safety and performance, effective management of ageing is a key element in safe and reliable operation of nuclear power plants.

To maintain plant safety and preserve the option of plant life extension, plant personnel must be able to effectively manage the physical ageing of plant components important to safety by controlling significant ageing mechanisms and detecting and mitigating their effects before failures occur. Ageing management means engineering, operations, and maintenance actions to control, within acceptable limits, the ageing degradation and wear out of systems, structures, and components.

The effect of ageing on the actual physical/ material conditions of the systems, structures and components influences the plants performance, which will show in the performance indicators to evaluate these conditions. Similarly to the approach taken in the detailed review of operation, maintenance, and safety management (see section 4.2), the ageing review focuses on the question to what extent ageing management is a well-managed process. To determine this, the Committee opted for a process evaluation of the management of physical/material ageing of systems, structures, and components. The review consisted of an evaluation of ageing management governance and of ageing management implementation at KCB versus a properly selected sample of nuclear power plants (the peer group) from the top 25% of plants with the best performance in operations.

The first step in developing the method for the evaluation of ageing consisted of the selection of practical Ageing Management Criteria (AMC) based on IAEA guidelines on ageing management. These guidelines provide recommendations for managing ageing of systems, structures and components important to safety, including recommendations on key elements of effective ageing management and an outline of a review of ageing management for long term operation. The guidelines contain generic attributes, which should be part of every component-specific Ageing Management Programme (AMP) (see Table 4-2).

Table 4-2 | *Generic attributes of an effective Ageing Management Programme (AMP) (Ref: IAEA Safety Guide NS-G-2.12)*

Attribute/ Review topic	Description/ Assessment basis/criteria
1. Scope of the ageing management programme based on understanding ageing	◗ System, structure (including structural elements) or component (SSC) subject to ageing management. ◗ Understanding of ageing phenomena (significant ageing mechanisms, susceptible sites): – SSC materials, service conditions, stressors, degradation sites, ageing mechanisms and effects – SSC condition indicators and acceptance criteria – Quantitative or qualitative predictive models of relevant ageing phenomena
2. Preventive actions to minimize and control ageing degradation	◗ Identification of preventive actions. ◗ Identification of parameters to be monitored or inspected. ◗ Service conditions (i.e. environmental conditions and operating conditions) to be maintained and operating practices aimed at slowing down potential degradation of the SSC.
3. Detection of ageing effects	◗ Effective technology (inspection, testing and monitoring methods) for detecting ageing effects before failure of the SSC.
4. Monitoring and trending of ageing effects	◗ Condition indicators and parameters to be monitored ◗ Data to be collected to facilitate assessment of SSC ageing ◗ Assessment methods (including data analysis and trending).
5. Mitigating ageing effects	◗ Operations, maintenance, repair and replacement actions to mitigate detected ageing effects / degradation of the SSC.
6. Acceptance criteria	◗ Acceptance criteria against which the need for corrective action is evaluated.
7. Corrective actions	◗ Corrective actions if the SSC fails to meet the acceptance criteria.
8. Operating experience feedback and feedback of R&D results	◗ Mechanism that ensures timely feedback of operating experience and R&D results (if applicable), and provides objective evidence that they are taken into account in the ageing management programme.
9. Quality management	◗ Administrative controls that document the implementation of the ageing management programme and actions taken. ◗ Indicators to facilitate evaluation and improvement of the ageing management programme. ◗ Confirmation (verification) process for ensuring that preventive actions are adequate and appropriate and all corrective actions have been completed and are effective. ◗ Record keeping practices to be followed.

The criteria developed by the Committee were on the one hand aimed at evaluating the ageing management governance (i.e. documentation of a plants policy, organisation, and methodology for ageing management) and on the other hand, at evaluating the way ageing management was implemented in practice. To do so the Committee evaluated the actual ageing management for a selected number of relevant systems, structures and components.

The second step in developing the method for the evaluation of Ageing involved the development of a methodology for the comparison of the ageing management programs at KCB with those of the selected group of plants in Canada, USA, and Europe. To check and refine this methodology a pilot study was performed on one EU, one US and one Canadian plant.

4.3.2 Selection of a peer group for ageing management review

Following the pilot study, a peer group of nuclear power plants was selected for the evaluation of ageing management using the following selection criteria:

- Belonging to the top 25% of plants with the best performance in operations (see section 4.1).
- Having an age relevant for this benchmark.
- Availability of recent information on ageing management programmes relating to ageing management criteria, such as reports on:
  - US License Renewal
  - Periodic Safety Review including a review of ageing management consistent with the IAEA Safety Guide on periodic safety reviews
  - IAEA SALTO review for which the report is publicly available
  - Other information on ageing management programmes for the evaluation using ageing management criteria available directly from the nuclear power plant

On the basis of these selection criteria, a peer group of five plants was selected.

4.3.3 Ageing management review methodology

To perform the ageing benchmark evaluations for KCB and the selected peer group, the following aspects of ageing management were reviewed:

a) Overall Ageing Management Programme
   - The plants' policy, organisation, and methodology for ageing management

b) Specific ageing management programmes for:
   - Flow accelerated corrosion of high energy carbon steel piping and valve bodies
   - Insulation materials of electrical cables
   - Buried piping and tanks

c) Ageing management scope for long term operation
   - Systems, structures, and components important to safety included in an ageing management programme for long term operation of a nuclear power plants

d) Specific time limited ageing analyses for:
   - Reactor vessel neutron embrittlement
   - Metal fatigue of class 1 piping
   - Environmental qualification of electrical equipment.

The specific ageing management programmes and time limited ageing analyses were chosen to make the evaluation result-oriented. The specific items were selected by the Committee because of their importance for effectiveness of ageing management for older plants.

The first step in the evaluation was a desk top review. The next step was the clarification and verification of review topics of which information was not clear or incomplete, executed by using e-mail, telephone, and in some cases a site visit.

The evaluation of each of the selected plants was reported in a narrative form and by a matrix. This matrix shows for each of the selected items the extent (full, partial or not) to which the plant meets the ageing management criteria defined by the Committee on the basis of international standards.

Following this analysis, the performance of KCB regarding ageing management was compared with the performance of each of the other plants in the peer group. Results of the comparison were expressed for each item in relative terms indicating whether the performance of the KCB ageing management was better, equal or worse than that of the plants in the peer group.

4.3.4 Results

Ageing management assessment for KCB

The evaluation of the ageing management programme (AMP) of KCB and its compliance with the ageing management criteria (AMC) is summarised in Table 4-3.

Summarising, the KCB ageing management activities are currently largely consistent with the ageing management criteria (based on the

Table 4-3 | *Ageing management matrix for KCB*

AMP aspect	AMC fully met	AMC partly met	AMC not met	Remarks
Overall plant AMP		√		Ageing Management Programme (AMP) governance does not meet criteria; development in progress.
Specific AMPs for: ◗ FAC ◗ Buried piping ◗ Insulation of el. cables	√ √ √			The ageing management criteria for the following specific systems, structures, and components that are important to safety are met: ◗ Flow accelerated corrosion (FAC) of high energy carbon steel piping ◗ Buried piping and tanks ◗ Insulation materials of electrical cables
AMP scope for long term operation	√			Adequately identified in accordance with IAEA guidelines and ageing management criteria.
Time limited ageing analyses revalidation: ◗ Embrittlement ◗ Metal fatigue ◗ Environmental qualification of electrical equipment	√ √	√		Revalidation of the following time limited ageing analyses has been performed and accepted by the regulator: ◗ Reactor vessel neutron embrittlement ◗ Metal fatigue of class 1 piping ◗ Environmental qualification (EQ) of electrical equipment Some recommendations on the last one are still to be implemented but they are included as a condition in the new license and thus secured.

IAEA guidelines). However, there is a need for improvement of coordination and traceability of ageing management by properly implementing and documenting a formal ageing management governance. Since this is still underway, the overall plant ageing management programme only partly meets the relevant ageing management criteria. The three reviewed system, structure, and component-specific ageing management programmes, as well as the programme scope for long term operation meet the relevant ageing management criteria. Two of the three time limited ageing analyses fully meet the relevant ageing management criteria; the 3rd one meets partly the criteria as some items are still to be implemented by the end of 2013.

Comparison of KCB with the ageing management peer group

To compare KCB with the peer group for ageing management review, all five nuclear power plants in the peer group were analysed in a similar manner as KCB. The results of these analyses were then compared with that of KCB. The outcome of this comparison is summarised in Table 4-4.

Table 4-4 | *Ageing management comparison*

AMP aspect	KCB better than peers	KCB and Peers equal	Peers better than KCB	Remarks
Overall plant AMP			5	KCB governance does not meet criteria; implementation and documentation of ageing management system for overall plant AMP in progress; completion planned by the end of 2013
Specific AMPs for: ◗ FAC ◗ Buried piping ◗ Insulation of el. cables	3	5 5 2		Some ageing management programme attributes not yet implemented at 3 plants
AMP scope for long term operation		4		Not relevant for 1 plant
Time limited ageing analysis (TLAA) revalidation: ◗ Embrittlement ◗ Metal fatigue ◗ Environmental qualification of electrical equipment		3 3 1	2	Not relevant for 2 plants Some items of TLAA revalidation for EQ of el. equipment are still to be implemented by KCB before the end of 2013

The following comments should be added:

- **Overall plant ageing management programme** deals with the ageing management governance, i.e. the plant's policy, organisation, and methodology for effective ageing management. At present implementation and documentation of an ageing management governance that meets the criteria (based on IAEA guidelines) is for KCB less complete than for each of the five peer nuclear power plants. This is due to the fact that until recently, KCB ageing management was implemented using existing operation and maintenance programmes and activities subject to on-going optimization based on relevant operating experience under the guidance of an Ageing Management Team. KCB explained to the Committee its plans to implement an ageing management system, consistent with the Dutch regulations and IAEA guidelines on ageing management (and thus with the ageing management criteria) which includes organisational arrangements for appropriate assignment of responsibilities for implementation of ageing management programmes of specific systems, structures, and components. This should be completed by the end of 2013. Once this is completed and meets the ageing management criteria, the overall plant ageing management programme could be rated as "equal to peers".

- **Specific ageing management programmes** deal with the extent to which the ageing management programmes for three specific systems, structures, and components have the generic attributes of an effective ageing management programme given in the IAEA Safety Guide on ageing management. KCB's ageing management programmes for flow accelerated corrosion (FAC) of carbon steel piping and valve bodies containing high-energy fluids and for buried piping and tanks are equal to its peers. KCB is equal to two and slightly better than three peer plants in the quality of ageing management procedures and their implementation for insulation materials of electrical cables.

- **Ageing management programme scope for long-term operation** involves a check if a representative sample of systems, structures and components (consisting of 19 passive and long-lived systems, structures and components that are not subject to replacement based on a qualified life or specified time period) is included in a plant ageing management programme. In this aspect, KCB is equal to four peer nuclear power plants; the fifth one does not plan long term operation, at present.

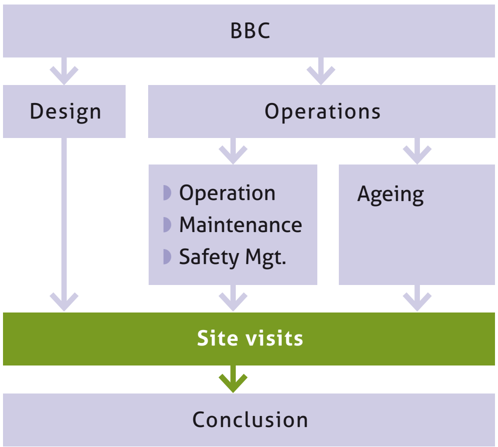
- **Time limited ageing analyses revalidation** deals with the extent to which there is, for the three selected time limited ageing analyses, an adequate demonstration that one of the following criteria is met:
  - The analysis remains valid for the period of long-term operation.
  - The analysis has been projected to the end of the period of long-term operation; or
  - The effects of ageing on the intended function(s) will be adequately managed for the period of long-term operation.

In the ageing management aspect of environmental qualification (EQ) of electrical equipment, KBC is equal to one peer plant and two peer plants are slightly better because some items of the revalidation of time limited ageing analysis for environmental qualification of electrical equipment are still to be implemented by KCB before the end of 2013. Time limited ageing analysis revalidation is not relevant for the two other peer plants.

Overall, the evaluation leads to the conclusion that the ageing management of KCB and its peers are in general comparable, which could be expected because they all generally follow the IAEA guidelines. Outstanding ageing management work at KCB that is being currently addressed includes; issuing formal KCB documentation articulating ageing management strategy, organisational arrangements, and methodology, completion of a few items from the revalidation of time limited ageing analysis for environmental qualification of electrical equipment, and documentation of procedures for more systems, structures and components-specific ageing management programmes.

It is envisaged that the performance of the ageing management of KCB and its five peer plants will be equal at the end of 2013 (i.e. before the start of KCB's long term operation as required by the operating licence) when KCB's formal ageing management implementation and documentation and all time limited ageing analysis should be completed.

5 Site visits



5.1 Site visit objectives

The objective of the site visits was to check whether the conclusions that were obtained through the desk top analysis are supported by the impression on-site on the way the plants are managed. In particular whether the strengths and weaknesses, as compared with KCB, that were identified in the peer review process are in line with the impressions obtained during the plant visits.

The aim was further to understand how specific operational safety aspects are managed at each of the visited plants and to compare these with operational safety aspects at KCB.

The plants for the site visits were with one exception selected from the peer group used for the process analysis of operation, maintenance, and safety management. In the selection attention was given to a proper geographical distribution. This led to a selected list of 5 plants in:

- France
- Germany
- Belgium
- Slovenia
- Canada

The site visits were carried out after finalization of the desk top analyses and were conducted by three Committee members in Europe and by two in Canada.

5.2 Site visit organisation

The visit consisted of two parts, one being the presentation by the host plant management followed by discussion or clarification on a number of topics, and the other being a plant tour.

The committee asked the plant management to cover in their presentation in particular the following items:

Operational Safety Management
- Control of plant status and configuration
- Monitoring and measuring of safety performance
- The corrective measures process
- Operator knowledge and skills
- Operational Experience Feedback

Maintenance
- Condition based maintenance
- Risk informed approaches in maintenance
- Monitoring of maintenance performance
- Outage management
- Management of contractors

Ageing Management
- Overall plant ageing management programme (AMP)
- Systems, structures and components - specific ageing management programs
- Ageing management programme scope for long term operation (LTO)
- Validity of time limited ageing analyses (TLAA) for the planned period of long term operation

Operation Safety Culture
- Implementation of safety policy
- Management commitment and leadership
- Attitudes of individuals
- The learning organisation
- Reporting culture

During the plant tour the Committee Experts aimed at obtaining an impression regarding issues such as:
- Main control room operations and the status of the reserve/emergency control room
- Material conditions and housekeeping
- Maintenance working places (maintenance shops as alternative)
- Specific areas to observe the equipment dedicated to accident management
- Conditions of safety related systems, in particular those to be utilised in the emergency situations (emergency power/ ultimate heat sink/accident management equipment/bunkered systems)

An additional aspect of the plant tour was to observe, as far as possible, the behaviour of the plant managers and personnel in the execution of their functional responsibilities.

5.3 Site visit results

As overall result of the site visits the Committee concluded that the impressions were in line with the results from the desk top reviews.

In all plants visited it was illustrated that business processes in the nuclear industry are specified in great detail and controlled accordingly. Although there are differences in the way plants are managed the operational performance of these plants is definitely the result of strictly specified and controlled processes and procedures.

Using operational experience to further improve very detailed and controlled business processes can easily lead to ever expanding manuals. This can impede their effectiveness also regarding safety. The nuclear industry recognised this danger and concluded that improving the quality of business processes required not more command and control, but better command and control. This applies in particular to improving plant safety. Promoting a safety-oriented attitude at all hierarchy levels could lead to better specifications of processes and more effective control. The committee was particularly interested to obtain insight during the plant visits what progress had been made in this area.

Below the observations of the Committee on this item are given together with other observations that were the result of these visits:

- Improving safety awareness and safety culture gets a lot of management attention in the plants visited. It was evident that translating this concept into effective measures is not an easy task. It takes time to convince the organisation of the importance of the concept and cultural differences play a role in translating it into effective measures. As a result, the approaches chosen differ, as well as the progress plants have made in this area.

- An important tool for mobilizing the organisation for improvements is to lower the threshold for reporting events/incidents and suggestions for improving performance. Improving feedback to the organisation on actions taken as a result of event reports and suggestions for improvement is equally important. The committee noted that the amount of reports/suggestions per plant vary considerable between plants (from 1000 to 5000/year) as well as the level of feedback. Nevertheless this tool is used in all plants visited.

- Another tool to improve safety is to incorporate the insights of probabilistic safety analyses (PSA) in the decision-making. Not only in the large decisions related to improvements of design by backfitting but also in day-to-day decisions, for instance for the planning of maintenance. The plants most advanced in this area introduced the concept of a “living PSA”; this implies the use of PSA to monitor the safety level at any time, by quantifying the effect on safety of every realised or intended change in the plants. The use of a “living PSA” is specifically useful in planning maintenance.

- Safety is not a static, but a dynamic item. Not only will requirements become stricter over time as a result of operating experience and the root cause analyses of nuclear incidents, but plants will also improve the safety of their design and operations continuously. All plants visited were working on improvements in response to the Fukushima incident. Continuous improvement through a learning organisation is very much a characteristic of the nuclear industry.

- In operations the increased use of simulators to train operators is noticeable. They are not only used for initial training but also for regular refreshment courses and dry runs for complicated operations. Simulators are a 1 to 1 copy of the actual control room and to facilitate a more integrated use in daily operations some plants install them nowadays on site.

- The committee was interested to see if there was a difference in managing safety between plants that operate stand-alone and plants that are part of a larger organisation with more nuclear plants. The Committee noted that there are larger organisations with more nuclear plants that opt for a centralised structure with a large supporting staff, but others that opt for decentralised operations with fairly autonomous plants. The smaller stand-alone plants like KCB are noticeably more active in international exchange of information and experience, for instance through participation in IAEA reviews of operations and ageing management and similar activities within the nuclear industry by WANO.

6 The stress tests on EU Nuclear Power plants by the European Council

In the aftermath of the Fukushima accident, in March 2011 the European Council concluded, "the safety of all EU nuclear plants should be reviewed, on the basis of a comprehensive and transparent risk and safety assessment" ("stress test"). The European Nuclear Safety Regulatory Group (ENSREG) endorsed the approach proposed by the Western European Nuclear Regulators Association (WENRA) where all plants were to be evaluated in a coordinated manner, with peer review being undertaken at EU level. Switzerland and Ukraine also participated in the stress test. The assessments were conducted by operators of all EU nuclear power plants, reviewed by national authorities who developed national stress test reports and submitted to ENSREG at the end of 2011.

The national reports were subject to an extensive EU wide peer review process, managed by the ENSREG Peer Review Board, and carried out from January to May 2012. The peer review included topical reviews of national reports and the on-site review of one nuclear power plant in every participating country. The review results were summarised in the stress test peer review report that reported on the safety level of the nuclear power plants in the whole of Europe. Seventeen individual national reports were also prepared with detailed recommendations relevant for each country. As a follow up, the National action plans in response to both national and Europe-wide recommendations were developed and submitted to the ENSREG for a peer review, which took place in April 2013.

The Europe-wide findings of the stress tests were that there are no technical reasons requiring immediate shutdown of any nuclear power plant in Europe. A series of good practices and recommendations were also identified. It was also concluded that for all plants there is room for safety improvements. Those were, on plant specific basis, taken on board and further developed in the National action plans.

When considering the approach of the ENSREG's stress test and the methodology used by the Borssele Benchmark Committee (BBC) it is important to note the differences between the two:

- **The subject was different:** within the stress test 143 nuclear power plants in the EU, Switzerland, and Ukraine were evaluated. The Committee compared the safety of KCB, against that of the approximate 250 water-cooled and water-moderated nuclear power plants in the EU, US, and Canada.

- **The objective was different:** the stress test was aimed at assessing the robustness of each nuclear power plant in the group in terms of their resilience against extreme external events. The stress test determined the safety margins that each nuclear power plant has against a set of (external) events that could be postulated to occur at the site. The Committee's objective was to determine whether the KCB is among the 25% safest of its kind in the EU, the US, and Canada. The Committee did so by comparing safety features of KCB against those of the other plants considered.

- **The scope was different:** the stress test assessments were very comprehensive and focussed on a number of safety issues

that were the initiators of the Fukushima accident and issues that influenced the propagation and hampered the mitigation of the consequences:
- External events (earthquakes, flooding, extreme weather conditions)
- Loss of electrical power and loss of ultimate heat sink
- Severe accident management

For its assessment the Committee did not consider the extreme challenges of Fukushima type. However it assessed all relevant safety aspect but assuming that the plants operated within their design base as defined within their licensing regimes, while taking into account design and operational upgrades (e.g. safety systems located in protected bunkers and introduction of severe accident managements guidelines, SAMGs).

The conclusion is that while both the stress test and the Committee assessed safety of power plants, their findings are not comparable. The results of the stress test cannot be used for any meaningful ranking of safety and, thus cannot be used to support the evaluation by the Committee.

Despite of all these differences, and in order to assure that no important elements were missed in the evaluation, the Committee looked at the stress test peer review report and the 17 country reports prepared by the peer review board. On the basis of these reports the Committee concluded that the KCB compares rather well with the other nuclear power plants evaluated in the stress test. This can be seen as an indication that the stress test results are in line with the results of the assessment by the Committee. An important finding is that the plants that scored high in the Committee design evaluation were found by the stress test to be robust and to possess large(r) safety margins.