Safety Benchmark of Borssele Nuclear Power Plant

Second report of the Borssele Benchmark Committee – 2018


EPZ

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Summary and Conclusions

The task of the Borssele Benchmark Committee is to determine whether the Elektriciteits Produktiemaatschappij Zuid-Nederland (EPZ) ensures 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, 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 the plant in 2013 – as was politically intended – but to allow it, in principle, to continue operation until 31 December 2033, if safety requirements are met as stated in regulations and license.

This agreement was formalised in a covenant, which also included the installation of the Borssele Benchmark Committee to evaluate whether KCB meets this condition.

This document represents the second report of the Committee.

Since the publication of the first Committee report, some reactors have been permanently shut down. Therefore, the list of reactors was revised to include only the reactors still in operation by 31 December 2016 (the cut-off date set by the Committee for its assessment). The final list of reactors involved in the benchmark contains a total of 237 reactors.

To establish an expert opinion on the safety level of KCB, as compared with the other 236 water-cooled and water-moderated power reactors in operation in the EU, USA and Canada, the Committee had to develop its own methodology. There are no internationally harmonised evaluations available for all safety aspects of a nuclear power reactor that expresses the safety in one well-defined number. Requirements for nuclear safety are established in most countries in line with international safety standards of the International Atomic Energy Agency (IAEA) and (within the EU) with the guides set up by the Western European Nuclear Regulators Association (WENRA) and the European Nuclear Safety Regulators Group (ENSREG). However, the responsibility lies with the national regulatory authorities and despite the efforts of the international organisations to harmonize these requirements, national differences remain, and the importance attached to various safety aspects is not necessarily uniform.

In principle, advanced Probabilistic Safety Analysis (PSA) would make it possible to combine all relevant safety aspects of design and operations into one model. However, PSA methodologies have not been standardized, and PSAs have not been conducted for all nuclear power plants. For those plants that do have PSAs, not all of them are available to the Committee. To develop PSAs would require an enormous effort and would be hindered by the unavailability of standardised reactor specific information and data for all the 237 peer reactors.

Furthermore, opinions about what is important for nuclear safety evolve due to operating experience, including root cause analyses of incidents.

Ranking reactor safety is, therefore, a complicated, if not impossible task with a time-dependent outcome. Nevertheless, the Committee is convinced that it developed a meaningful methodology based on all available information in combination with expert assessment, that could be used to compare the safety of KCB with the other reactors the Committee had to assess.

For the second report, the Committee retained the overall structure of the methodology previously developed, and improved it to reflect recent developments. In particular, three recent developments led to refinements and additions:

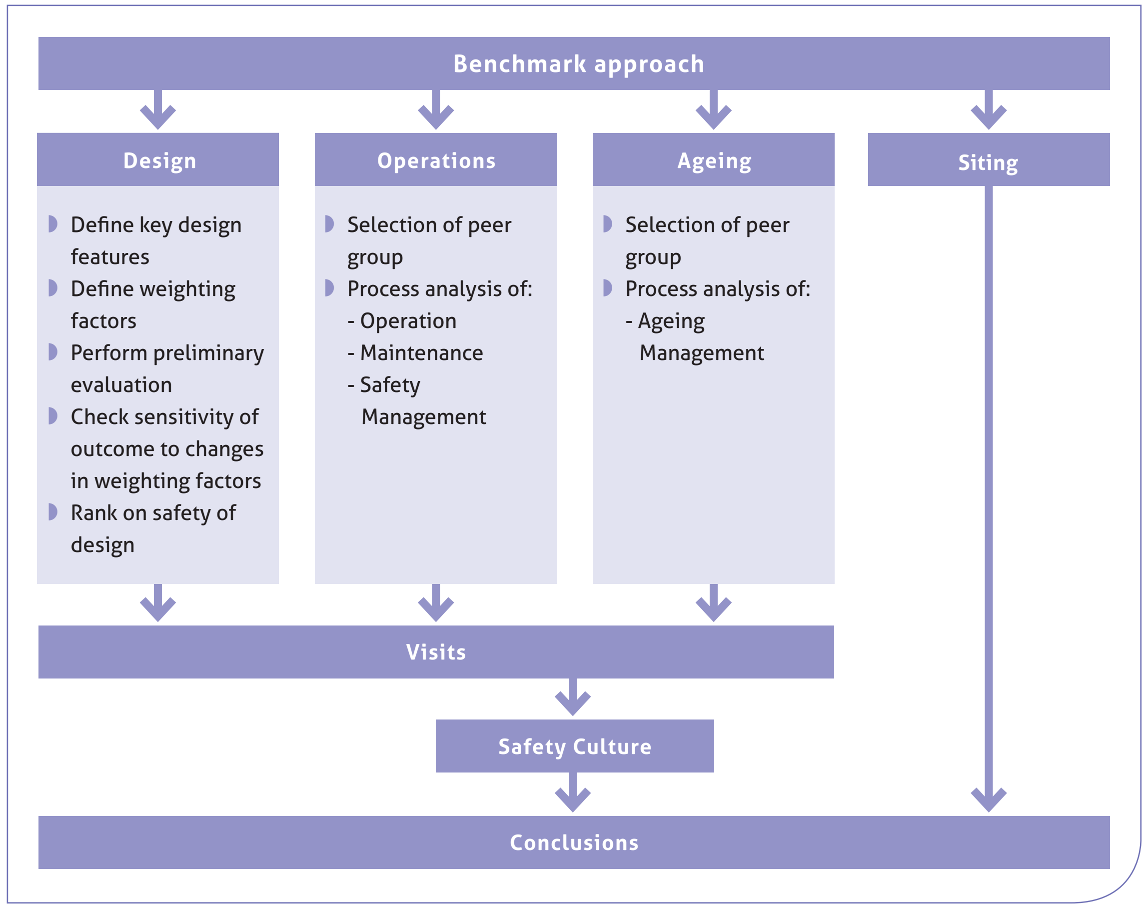
Firstly, post-Fukushima studies and their follow up brought new insights about design safety, leading to refinement and extension of the design benchmark methodology (chapter 4) and a separate evaluation of siting (chapter 6).

Secondly, the wider use of the Safety Aspects of Long Term Operation (SALTO) review mission made it possible to conduct the evaluation of ageing using the recently refined and internationally consistent methodology developed by the IAEA for SALTO review. The use of the findings in SALTO reports is now the basis of the ageing benchmark, in a way comparable to Operational safety evaluations that uses the IAEA OSART findings (chapter 5).

Thirdly, the increased worldwide consciousness about the importance of safety culture is reflected in a more consistent and standardized approach (chapter 8).

Schematically the Committee opted for the approach as shown in Figure 1-1 (see page 8).

Figure 1-1: Schematic approach for the benchmark



This methodology contains a separate safety assessment of:

- Reactor design (including reactor upgrades)
- Reactor operations (covering operation, maintenance, safety management).
- Ageing management
- Siting
- Safety Culture

Using the developed extended methodology the Committee compared the safety of 237 plants. From this assessment the Committee unanimously concluded that KCB is within the top 25% safest water-cooled and water-moderated reactors in EU, USA and Canada.

This summary provides an overview of the report and background support to this conclusion.

Design safety

Under all circumstances for nuclear reactor safety it is essential to assure:
- Reactivity control
- Core cooling (heat removal)
- Confinement of radioactivity

In the first report, the Committee discussed the contribution of specific design features to achieve these goals regarding the reactors' capabilities for accident prevention, accident mitigation and containing radioactive substances within the reactors' interior to reduce hazards for the environment. During the first reporting period, the Committee developed a dedicated methodology to assess the design characteristics of all ca. 250 power reactors in the benchmark to compare against Borssele nuclear power plant. The comparison was based on the sum of ratings established in four categories of key design features, i.e. redundancy and diversity, containment, bunkered systems and severe accident management.

The Fukushima Daiichi accident in 2011 raised questions regarding the traditional design principles. Therefore, for the second report, the Committee extended and refined the methodology for benchmarking design, in order to better represent the overall design safety, and to reflect the lessons learned from the Fukushima Daiichi accident. The Committee modified the benchmarking method by redefining the existing design features, adding sub-features and by considering one additional key design feature (design of spent fuel pool). For each of the design features and their sub-features, design solutions were identified and scoring criteria attributed reflecting their impact on design safety:
- Redundancy and diversity of safety systems
- Design of containment
- Availability of bunkered systems
- Severe accident management
- Design of spent fuel pool

A method based on probabilistic safety analysis was used to look at the relative importance of these impacts.

Due to the complexity of the evaluation method, it was necessary to test the scoring scheme in a pilot assessment, including a sensitivity analysis. In the end, it gave the Committee the confidence that the extended and refined methodology was appropriate for the evaluation of the safety of design and taking into account the new insights.

All 237 reactors were evaluated with the new methodology, using the abundant data on the design of each of these reactors. Collection of all relevant data required considerable effort and access to different sources of information (stress test reports, licence renewal applications, etc.). For the benchmark, all planned safety modifications were considered "implemented", even if their completion was scheduled after 31 December 2016.

The outcome was a score per reactor with higher scores indicating safer designs. The scores were subsequently used to identify the 25% safest reactors from the design point of view.

From the results, the Committee concluded that:

- The reactors considered had scores distributed across a wide range, with a larger number of reactors ranking in the lower range and a smaller number of reactors ranking towards the upper range.
- The extended and refined methodology provides better distinction among different reactor design characteristics than the one used in the first report.
- There was no clear relation between the age of the reactor and its score: both older and newer reactors had high scores as well as low ones.
- The results were influenced by all key design features, without any of them being dominant.

Conclusion on design

The results of the benchmarking with the extended and refined methodology indicate that from the design point of view, KCB remains well within the top 25% safest reactors.

As in 2013, the Committee is still of the opinion that KCB's favourable score in the design review is the result of prudent original design, but even more because of continuous safety improvement programs that have taken place since 1986, in particular due to periodic safety reviews.

Safety in Operation

For evaluating safety in plant operations, the Committee used the same two-step approach developed during the first benchmark period. In the first step, the top 25% best-performing plants were selected based on performance indicators. These indicators reflect operational (and not only safety) performance during the past operating period but do not assure the same performance in the future. Therefore, in the second step, the Committee analysed whether the safety performance is the result of well-defined and controlled processes directed by the plant's management. Considering the amount of information needed for detailed process analysis, this was only feasible for a sample of the plants concerned. However, to determine whether KCB's performance in the management of operations is like that of the other 25% best-performing plants in operations, it was enough to compare KCB in detailed analysis with a properly selected sample.

Step One: 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 based on several performance indicators, of which most are also relevant for evaluating safety.

The Committee had access to these performance indicators and used them in its first step to select the 25% best-performing nuclear power reactors of the 237 peers. To do so, the Committee combined the performance indicators into a composite number using weighting factors to express their relevance for reactor safety. The results were then normalized to 100.

Given the fact that scores in these types of monitoring systems can be substantially affected by incidental events, the Committee decided to use multi-year averages as during the first benchmarking period. KCB is well within the top 25% reactors with the best performance of reactor operation.

Step Two: evaluation of the plant internal processes

To evaluate if safety performance is the result of well-defined and well-managed processes directed by plant management requires extensive information on plant operations. 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 this analysis.

For the process evaluation of *operations, maintenance and safety management*, a peer group of 10 plants was selected for which recent OSART reports were available. The Committee used the scoring system developed during the first benchmarking period.

The results showed that the score of KCB is the fourth best compared to the scores obtained by the peer group. The sensitivity analysis indicated that this outcome was not sensitive to varying the weighting factors used in the scoring system. Even though the input data contained some uncertainties related to the moment in time when the OSART mission was conducted, as well as to the subjectivity of the evaluation, the approach taken assured that the results were robust.

Conclusion on operations

The score obtained by KCB supports the conclusion that the safety performance in reactor operations, maintenance, and safety management of KCB compares well to that of the 25% best-performing reactors in operations.

Ageing Management

The Committee developed a new ageing benchmark approach to replace the method used for the first report. The reason for this decision was that in recent years the internationally consistent methodology of the IAEA Safety Aspects of Long-Term Operation (SALTO) has been refined and used on a larger number of plants, allowing the Committee to undertake the evaluation in a way comparable to Operational safety evaluation that uses the IAEA OSART results. The new methodology considers safety aspects of ageing management for long-term operation as assessed in the IAEA SALTO missions and OSART missions with a SALTO module. The areas covered by the IAEA missions are consistent with areas reviewed in the first report.

The ageing benchmark methodology is structured similarly to the methodology used for the second step of the operation benchmark. The Committee developed a scoring system to combine the outcome of the SALTO missions into a composite number indicating to what extent ageing management is the result of well-controlled processes. The lower the number, the better the ageing management.

A sensitivity study confirmed that the ranking of KCB obtained using this methodology is not sensitive to varying the weighting factors used in the scoring system, and that the methodology is fit for this purpose.

The ageing management programme of KCB was benchmarked against a peer group of five water-cooled and water-moderated reactors that underwent IAEA SALTO missions or OSART missions with a SALTO module during the current reporting period and having a good geographical spread over the benchmark area.

Conclusion on ageing

The benchmarking results of KCB's ageing management programme against the peer plants show that KCB's total score is comparable to that of its peers.

Siting

Siting refers to the process of evaluating the suitability of a location for a nuclear facility, in which events of natural or human-induced origin are identified that can jeopardise the safety of the reactor. These events are called external hazards, as they originate from outside the reactor and the event itself (earthquake, high water level) is not influenced by the design of the reactor. The magnitude and probability of occurrence of external hazards are evaluated for design purposes of the reactor so that the reactor can be adequately designed to withstand these hazards.

Therefore, the key issue for evaluating siting risks is to consider how the safety implications of external hazards at a specific location are considered and how their consequences are mitigated by design characteristics.

The Committee decided to focus the evaluation on the siting aspects of KCB specifically. The goal was to assess whether the siting risks at KCB are assessed in line with 'state-of-the-art' international practices and considered in its design, and to assess whether these external hazards pose a risk to KCB.

For this evaluation, the Committee used information from the EU Post-Fukushima stress test, complemented by the underlying safety evaluations by KCB, as well as evaluations performed as part of their most recent 10-yearly Periodic Safety Review (PSR). Attention was paid to earthquakes, flooding, extreme weather conditions, airplane crashes and shipping accidents on the Westerschelde that could possibly affect KCB.

Conclusion on siting

The Committee concludes that the siting risks at KCB were thoroughly and comprehensively investigated, reflecting 'state-of-the-art' and international good practices. The latest requirements for existing nuclear power plants and the findings from Fukushima were also considered. The Committee is confident that siting issues do not negatively impact the overall safety ranking of KCB.

Site visits

The Committee visited KCB and five plants from the operations peer group. To get a well-structured result for each visit, the Committee used a detailed document comprising questions and a scoring mechanism.

The aim of the visits was twofold:

- To check whether the conclusions of the desktop analysis of operational safety management, maintenance and ageing management were supported by the impressions obtained from the plant visit of how the reactors were managed, and
- To compare KCB's safety culture with that of the other plants.

During the visits, the Committee had discussions with plant management and personnel and observed their behavior.
Plant walk downs were also part of the visits, during which the Committee observed the main control room operations, material conditions and housekeeping, workshops, areas for accident management equipment, and conditions of safety-related systems.
In all plants visited, it was highlighted that business processes in the nuclear industry were specified in detail and controlled accordingly. Although there were differences in the way plants were managed, the operational performance clearly reflects strict adherence to controlled processes and procedures.

Conclusion on site visits

Based in the site visits, the Committee concluded that their observations were in line with the results from the desktop reviews and that KCB is in line with international best practices and requirements in terms of the items examined.

Safety Culture

Safety culture cannot be benchmarked in the same way as the other aspects described in this benchmark report. Safety culture refers to the ways that safety issues are addressed in the workplace. It often reflects the attitudes, values, beliefs and behaviors that employees share in relation to safety and how management influences this behaviour. "Attitudes, values and beliefs" do not easily lend themselves to measurement. However, attributes can be identified that shape or influence safety culture.

To compare the safety culture at KCB with that at other plants, the Committee developed a method to be used during the site visits. The method is based on the assessment of:

- The way the plant is led and safely managed
- The way the organization handle (elements of) the safety management system
- The way the organization involves and motivates its people

The Committee noted that at all the visited plants, safety culture receives much more attention than it did five years ago. However, there is a large difference in the methodology and ways of implementation.

Conclusion on safety culture

The Committee noted that KCB is very active in this area. Based on the assessment performed, the Committee concludes that safety culture at KCB is equal to or better than at the nuclear power plants visited.

Acknowledgement
The Committee would like to express its appreciation to the nuclear power plants participating in this benchmark for their collaboration, particularly during the site visits.

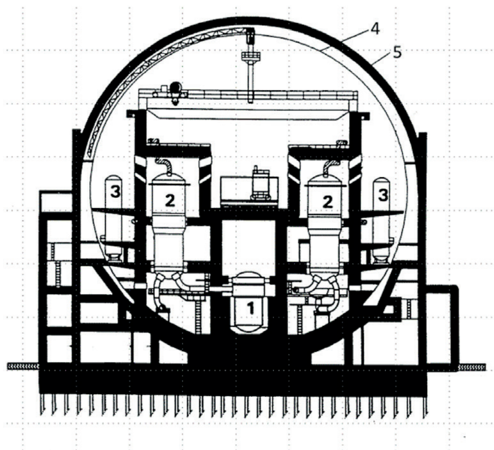
Abbreviations

AFWS	Auxiliary Feed Water System
BWR	Boiling Water Reactor
CS	Core Spray
ECCS	Emergency Core Cooling System
EU	European Union
HPCI	High Pressure Coolant Injection System
IAEA	International Atomic Energy Agency
KCB	Borssele Nuclear Power Plant (Kerncentrale Borssele)
LPCI	Low Pressure Coolant Injection System
LTO	Long-Term Operation
NPP	Nuclear Power Plant
OSART	Operational Safety Review Team
PHWR	Pressurised Heavy Water Reactor
PSA	Probabilistic Safety Analysis
PWR	Pressurised Water Reactor
SALTO	Safety Aspects of Long-Term Operation (IAEA)
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
WENRA	Western European Nuclear Regulators Association

1 Introduction

The Borssele nuclear power plant is a light water PWR with a thermal power of 1366 MW and a net electrical output of approximately 490 MW. The installation is a two-loop plant designed by Siemens/KWU. The plant has been in operation since 1973. The reactor and the primary system, including steam generators, and the spent fuel pool are in a spherical steel containment. This steel containment is enveloped by a secondary concrete enclosure.

Cross-section of the reactor building of the Borssele plant



1. Reactor pressure vessel
2. Steam generator
3. Medium-pressure core inundation buffer tank
4. Steel containment
5. Secondary concrete enclosure (shield building)

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 several conditions. This agreement was formalised in the "Convenant Kerncentrale Borssele"[1]

One of the conditions in the Covenant states:

> *"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 based on quantified performance indicators. If a quantitative comparison is not possible for the design, operation, maintenance, ageing and safety management, the comparison shall be made based on a qualitative assessment..."*

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

According to the Covenant, a committee of five independent experts, established by the covenant parties, shall assess whether this condition is met. The opinion of the Committee shall be reported to the Covenant parties every five years. The first report was prepared in 2013 and was made public by the Covenant parties.

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.

This document represents the second report of the Committee. The Committee for the second report was established in 2014.

The Committee comprises:
- P. Nabuurs, former CEO of KEMA N.V.
- J. Lyons, reactor safety specialist, former director, Division of Nuclear Installations Safety at the IAEA
- R. Stück, former head of the branch Reactor Safety Analysis, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Köln, Germany
- B. Tomic, principal consultant at ENCO, Vienna, Austria
- A.M. Versteegh, former managing director of Nuclear Research and consultancy Group, Petten, The Netherlands

The Committee's main duties are:
- To determine whether 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 based on expert qualitative assessment.
- To carry out its duties objectively, independently of the interests of industry, civil society organisations, politics, and current government policy.

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

This report contains the results of the second assessment of the Committee and its unanimous opinion based on 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 KCB, but to compare its safety with that of its "peers" as defined by the Covenant, i.e. water-cooled and water-moderated power reactors in the European Union, the United States of America, and Canada. Based on that comparison, the Committee shall 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 was anonymised to the level needed to ensure confidentiality.
- Considering its task, the Committee focuses only on safety aspects relevant for the protection of the public and environment surrounding the reactor. Safety aspects relevant only for the consequences inside the plant were not considered. These consequences were considered a (economic) risk for the plant owners.

Since the Fukushima Daiichi accident, new insights but also the requirements regarding the safety of nuclear power plants have developed. Extensive information on this subject was presented in EU Post- Fukushima Stress Tests reports, National Actions Plans, etc. The Committee reflected this information and insights in the benchmark methodology, which led to some extensions and refinements.

In the following chapters, the Committee's methodology is described in chapter 2.
Next, the separate steps in the evaluation are explained in more detail and the results are provided for design (chapter 3), operation (chapter 4), ageing management (chapter 5) and siting (chapter 6). The findings of the site visits are described in chapter 7 and safety culture aspects are considered in the last chapter (chapter 8).

2 Methodology

The first benchmark study, reported in 2013, covered approximately 250 nuclear power plants, divided into three basic types: Pressurised Water Reactors (PWR), Pressurised Heavy Water Reactors (PHWR), and Boiling Water Reactors (BWR). Since the last reporting period, some of these plants have been permanently shut down. Therefore, the list of plants was reviewed to include only the reactors that were still in operation as of 31 December 2016 (the cut-off date set by the Committee). The final list of reactors contains a total of 237 reactors including KCB.

Figure 2-1 shows the distribution of the reactor types in the benchmark population and Figure 2-2 the geographical distribution of the reactors.

To establish an expert opinion on the safety level of KCB, as compared with the other 236 water-cooled and water-moderated power reactors in operation in the EU, USA and Canada (as of 31 December 2016), the Committee had to develop its own methodology. There are no internationally harmonised evaluations available for all safety aspects of a nuclear power reactor expressing safety in one well-defined number. Requirements for nuclear safety are the responsibility of national regulatory authorities and established in most countries in line with international safety standards of the International Atomic Energy Agency (IAEA) and (within the EU) with the guides set up by the Western European Nuclear Regulators Association (WENRA) and the European Nuclear Safety Regulators Group (ENSREG). Despite the efforts of these organisations to harmonize these requirements, national differences remain, and the importance attached to various safety aspects is not necessarily uniform.

In principle, advanced Probabilistic Safety Analysis (PSA) would make it possible to combine all relevant safety aspects of design and operations into one model. However, PSA methodologies have not been standardized and

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

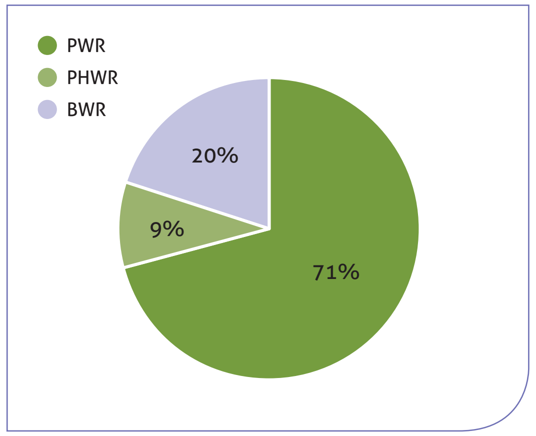
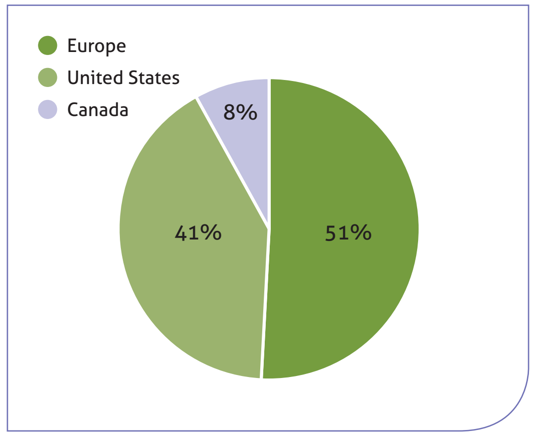


Figure 2-2 | *Geographical distribution of the benchmark population*



[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

PSAs have not been conducted for all nuclear power plants. For those plants that do have PSAs, not all of them were available to the Committee. To develop PSAs would require an enormous effort and would be hindered by the unavailability of standardised reactor-specific information and data for all the 237 peer reactors.

Furthermore, opinions about what is important for nuclear safety evolve over time due to operating experience, including root cause analyses of incidents. The accident at the Fukushima Daiichi power plant in 2011, the subsequent European stress tests, and other investigations have resulted in many new requirements and actions that influenced nuclear safety. However, the speed and possibilities to adapt reactors to new requirements differ from country to country.

Taking these considerations into account, the Committee developed its own methodology for the first report, published in 2013, which supported the Committee's assessment of the safety of KCB. The methodology used available information on the different elements of reactor safety that could be meaningfully compared among the approximate 250 reactors.

For the second report, the Committee maintained the overall structure of this methodology and improved it to reflect recent developments. In particular, three recent developments led to refinements and additions:

Firstly, post-Fukushima studies and their follow up brought new insights about design safety, leading to modification of the design benchmark methodology (chapter 4) and a separate evaluation of siting (chapter 6).

Secondly, the wider use of the Safety Aspects of Long Term Operation (SALTO) review mission made it possible to conduct the evaluation of ageing using the recently refined and internationally consistent methodology developed by the IAEA for SALTO review. The use of the findings in SALTO reports is now the basis of the ageing benchmark, in a way comparable to Operational safety evaluations that uses the IAEA OSART findings (chapter 5).

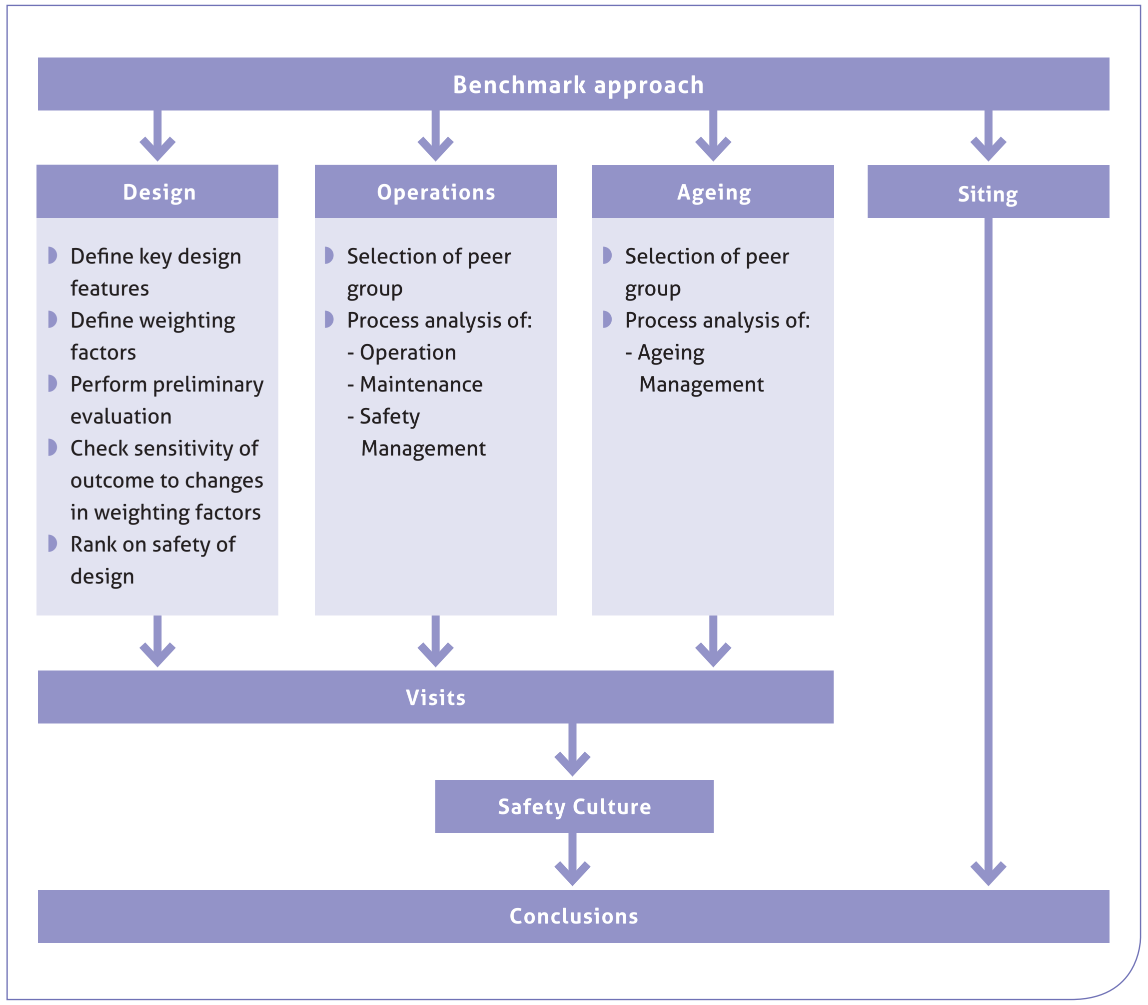
Thirdly, the increased worldwide consciousness about the importance of safety culture is reflected in a more consistent and standardized approach (chapter 8).

The Committee is convinced that it meaningfully enhanced the previously developed methodology based on the available information in combination with expert assessment. The methodology makes it possible to determine, with enough confidence, whether KCB is among the safest 25% water-cooled and water-moderated nuclear power plants in Europe, the USA, and Canada. Because of their different natures, this methodology contains a separate safety assessment of:

- Reactor design (including reactor upgrades)
- Reactor operations (covering operation, maintenance, safety management)
- Ageing management
- Siting
- Safety Culture

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

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



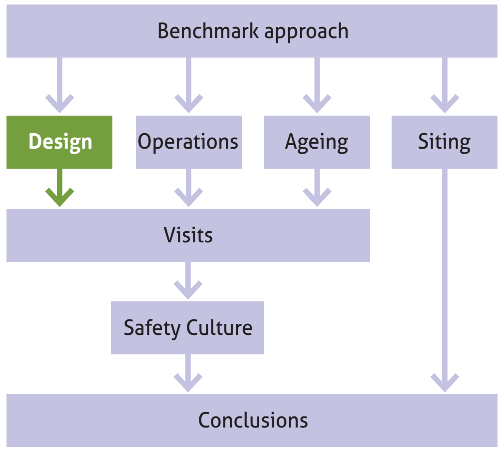
The safety assessment of design (chapter 3) was carried out for all 237 nuclear power reactors based on specified key design features. For evaluating safety in reactor operations (chapter 4), a two-step approach is used. In the first step, the top 25% best-performing reactors were selected, based on performance indicators. These indicators cover the past and reflect performance and not only safety; they do not assure the same performance in the future. In the second step, the Committee conducted process analysis to assure themselves that safety performance is the result of well-defined and controlled processes directed by plant management. Considering the amount of information needed for detailed process analysis, this was only feasible for a sample of the plants concerned. However, to determine if KCB's performance in the management of operations is like that of the 25% best-performing plants, it was enough to compare KCB in a detailed analysis with a properly selected sample.

The newly developed ageing management benchmark methodology (chapter 5) is structured similarly to the methodology used for the operation benchmark. In this benchmark, the ageing management programme of KCB was compared in a detailed analysis to that of a properly selected sample of peer plants.

Additionally, in the siting evaluation (chapter 6), it was assessed whether external hazards pose a risk to KCB and whether siting risks at KCB are evaluated according to 'state-of-the-art' methodology, and whether the design is appropriate to mitigating those events.

The results of the assessments were complemented by several site visits (chapter 7) to check whether the conclusions of the above analysis were supported by the impressions gathered on plant management during the plant visit. Additionally, during the site visit, information was collected on safety culture using a newly developed method to compare the safety culture of KCB with that of the other visited reactors (chapter 8).

3 Evaluation of Design Safety



3.1 Introduction

In the first benchmark report (2013), the Committee developed a dedicated methodology to assess the relevant safety design characteristics of the approximately 250 power reactors considered. The comparison was based on the sum of ratings established in four categories of key design features: redundancy and diversity of safety systems, containment, availability of bunkered systems, and severe accident management.

However, the Fukushima Daiichi accident in 2011 raised questions regarding the traditional design principles. Therefore, for the second report, the Committee extended and refined the methodology for benchmarking design in order to better represent the overall design safety, and to reflect the lessons learned from the Fukushima Daiichi accident, in particular, the identified vulnerabilities and the safety enhancements proposed. Based on probabilistic safety analysis, the relative importance of the different safety features was also analysed. The Committee modified the benchmarking method by redefining the existing design features, adding sub-features and by considering one additional key design feature (design of spent fuel pool). Also for each of the design features and their sub-features, design solutions were identified and scoring criteria attributed reflecting on their impact on design safety.

The implementation of this refined and extended methodology included:
- A pilot study on 20 reactors, whereupon the scoring scheme was tested and adjusted.
- The collection of design information on the peer nuclear power plants to be 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 extended and refined scoring scheme.

The reactors for the pilot study were a representative sample of peers.

3.2 Definition of key design features and categories

Ranking reactor design safety requires first defining key design features and then determining their expected relevance to potential external radiological impact of the plant.

All currently operating nuclear power reactors belong to the so-called generation II reactor design classification; this refers to the class of commercial reactors built up to the end of the 1990s. They include three basic reactor types, which were 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 various vendor countries (USA, Germany, France, Canada, USSR), the initial safety concepts and requirements of the three 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 others. With accumulated and shared operating experience and new safety concerns (e.g. lessons from the Three Mile Island 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 with different features being added to the designs to enhance safety levels.

Efforts in harmonizing design requirements to enhance safety were intensified worldwide in the last decade. Through Periodic Safety Reviews (in Europe) or the Regulatory Compliance Programme (in the USA), reactor characteristics were periodically checked against new safety insights and requirements. In many cases, adaptation of nuclear reactors (backfitting) was required.

To assure reactor safety, three fundamental safety functions need to be assured under all circumstances:
- Reactivity control
- Core cooling (heat removal)
- Confinement of radioactivity

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

The starting point for this 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. The idea behind defense in depth is to manage risk by layering diverse, defensive strategies, so that if one layer of defense turns out to be inadequate, another layer of defense will detect, compensate, or correct using the appropriate measures; it assures that there are overlapping or backstopping provisions. Applying this layered defence-in-depth concept throughout the design and operation provides stratified protection against a wide variety of anticipated operational occurrences, design basis accidents, and severe accidents; this includes those resulting from equipment failure or human action within the plant and hazards that originate outside the plant.

Table 3-1 | *Defence in depth concept (ref. WENRA report: Safety of new NPP designs, 2013)*

DEFENCE-IN-DEPTH CONCEPT				
Levels of defence-in-depth	**Objective**	**Essential means**	**Radiological consequences**	**Associated plant condition categories**
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation, control of main plant parameters inside defined limits	No off-site radiological impact (bounded by regulatory operating limits for discharge)	Normal operation
Level 2	Control of abnormal operation and failures	Control and limiting systems and other surveillance features		Anticipated operational occurrences
Level 3	Control of accident to limit radiological releases and prevent escalation to core melt conditions	**3a** Reactor protection system, safety systems, accident procedures	No off-site radiological impact or only minor radiological impact	**3a** Postulated single initiating events
		3b Additional safety features(3), accident procedures		**3b** Postulated multiple failure events
Level 4	Control of accidents with core melt to limit off-site releases	Complementary safety features(3) to mitigate core melt, Management of accidents with core melt (severe accidents)	Off-site radiological impact may imply limited protective measures in area and time	Postulated core melt accidents (short and long term)
Level 5	Mitigation of radiological consequences of significant releases of radioactive material	Off-site emergency response Intervention levels	Off site radiological impact necessitating protective measures	-

Levels 1 and 2 within defence-in-depth are mainly addressed by careful design and appropriate safe operation; both are verified by the regulatory body. The designs are verified during the initial licencing process. The safety in plant operation is verified through regulatory inspections (oversight), periodic safety reviews or other mandated regulatory checks.

Within the Committee's evaluation, levels 1 and 2 were considered, but the focus of the Committee's assessment was on safety aspects relevant for the environment surrounding the plant, as this, due to higher level protections, is where today's nuclear power plants differ from safe to very safe.

To adequately capture those aspects, the assessment focused on enhanced capabilities for accident control and accident mitigation and for containing radioactive substances within the plant's interior. These are mainly defense-in-depth levels 3 and 4.

The objectives of key engineered design features for control and mitigation of accidents include:

- 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, in the first report, the Committee identified four key design features, which determine the safety level of the reactor from the perspective of potential impact on the environment. From the post-Fukushima safety considerations and the massive investments in safety improvements, the Committee learned that it needed to redefine the originally proposed key features as well as define new key features, as follows:

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

To be able to better distinguish between plants, sub-features were added to the key features. For every feature and sub-feature, a final score was given, based on its contribution to safety.

The methodology that the Benchmark Committee deployed in the first report considered that all safety features were of "equal weight", meaning that they equally contribute to nuclear safety. When refining the methodology for the Second report, the Committee investigated how to link their scores to their relative contribution to safety, using the insights from a PSA study for a generic PWR reactor, and appropriate engineering judgement.

PSA insights confirmed that bunkered systems, mobile systems, as well as strong containment have a very strong impact to safety. Multiple sensitivity analysis was performed during the development of the methods and the results considered. This new methodology more accurately reflects the safety of nuclear power plants.

3.2.1 Redundancy and diversity

Redundancy and diversity are the major design features that strongly impact mitigating the consequences of events that might lead to leaving the reactors uncooled and in danger of overheating, which could result in "core damage" or "core melt".

Redundancy refers 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 refers 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 plants 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 functional 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 achievable in all cases.
In particular, implementing redundancy and diversity 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 must 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 are in many cases incorporated into current operating plants through modifications of the existing equipment (backfitting).

The assessment by the Committee resulted in the following categorisation (see Table 3-2) for evaluating and ranking redundancy and diversity. Although the selection and categorisation reflect international practices, lessons learned from EU Post-Fukushima stress tests and many years of experience available to the Committee, there is a certain level of subjectivity in the selection of the categories. 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.

To assess what influences a different categorisation might have on the outcome, a sensitivity analysis was performed. The conclusion was that this effect for the chosen methodology was limited and not relevant to the final categorisation.

Table 3-2 | *Definition of key feature "redundancy and diversity"*

REDUNDANCY AND DIVERSITY		
Core cooling system		
I	PWR	2x 100% or less ECCS redundancy, no diversity in AFWS
	BWR	No redundancy in HPCI; 2x 100% or 3x 50% LPCI; 1X 100% CS
II	PWR	More than 2x 100% ECCS redundancy, no diversity in AFWS OR 2x 100% ECCS redundancy, diversity in AFWS
	BWR	Redundancy, no diversity in HPCI; 4x 50% or 3x 100% LPCI; 1x 100% CS OR No redundancy in HPCI; 4x 50% or 3x 100% LPCI; 2x 100% CS
III	PWR	More than 2 x 100% ECCS, diversity in AFWS
	BWR	Redundancy and diversity in HPCI; 4x 50% or 3x 100% LPCI; 2x 100% CS
Ultimate heat sink		
I	No redundancy, no diversity	
II	Redundancy (availability of large water stocks on-site or alternative ultimate heat sink)	
III	Redundancy and diversity (availability of large water stocks on-site and an alternative ultimate heat sink)	
AC/DC power supply		
Layers of power supply		

Note:
100% redundancy implies that one of two system parts is enough to fulfil the required function.
50% redundancy implies that two of the four available system parts are enough to fulfil the required function.

3.2.2 Containment

The confinement of radioactive material in a nuclear power reactor, 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 defence-in-depth concept, 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.

In the design benchmarking methodology used during the first reporting period, all reactors were assessed in accordance with their containment design, i.e. whether they were pressure suppression containment (all types), or full pressure dry single containment, or full pressure double wall containment or full pressure double wall containment capable of withstanding large aircraft crashes. Also reflecting post-Fukushima safety considerations, this methodology was extended to include additional systems and features, which can protect the containment, enhance its safety function and minimize off-site releases resulting from core damage (i.e. features to control hydrogen, strategies for in- and ex-vessel retention of molten core, external reactor vessel cooling, containment filtered venting). With this approach, the containment and its additional features were considered as one (complete) key feature, which allows for safety considerations that go beyond the original design basis. Table 3.3 shows the containment feature, including its subfeatures.

Table 3-3 | *Definition of key feature "containment" with sub-features*

CONTAINMENT		
Containment design		Sub-features
I	Pressure suppression containment (all types) or full pressure dry single containment	Features to control hydrogen
		Strategies for in- and ex-vessel retention of molten core
		External reactor vessel cooling
		Containment filtered venting
II	Full pressure double wall containment	Features to control hydrogen
		Strategies for in- and ex-vessel retention of molten core
		External reactor vessel cooling
		Containment filtered venting
III	Full pressure double wall containment capable of withstanding large aircraft crashes	Features to control hydrogen
		Strategies for in- and ex-vessel retention of molten core
		External reactor vessel cooling
		Containment filtered venting

3.2.3 Bunkered systems

Hazards of internal or external origin, such as: explosions, 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 that might mitigate their consequences. Specially designed bunkers that contain some of the key systems (e.g. power supplies, heat removal, and basic controls) were not included in the original design of most nuclear power plants. These specially designed bunkers were added later to assure protection of safety systems from internal and external hazards and thus increase plant safety. When added to the original design, these bunkered systems also increased redundancies and solved deficiencies (e.g. inadequate spatial separation, one of the most important protective features for internal and external hazards).

Initially, bunkered systems were 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 supplies.

Natural hazards were, to a different extent, considered in the initial design of nuclear power plants. Safety improvements were made when experience or analyses later showed that additional hazards needed to be considered. Manmade hazards were also considered in some designs, like external explosions caused by nearby industrial facilities or aircraft crashes. After 2001, more severe external hazards of human origin, e.g. big commercial aircraft crashes, have called for attention. After the Fukushima Daiichi accident, special emphasis was given to the resistance to extreme natural hazards. For the new reactor designs, such hazards are typically included in the design basis. Findings from the EU Post-Fukushima Stress tests indicated that some of the older reactors that were backfitted with bunkered systems were resistant to certain (extreme) hazards beyond those included in the design basis of the reactor, attaining a safety level equivalent to newer reactors.

The methodology presented in Table 3-4 reflects new safety considerations, redefining and extending the original definition of the key feature "bunkered systems". Bunkered core cooling and heat removal systems both provided in the original design for newer plants and backfitted for older plants were considered. The scoring considered the severity of natural and man-made hazards a bunkered building can withstand, the dedicated supplies available in bunkered systems and their redundancy, and whether the Emergency Control room was bunkered.

A post-Fukushima design improvement created to increase resistance against external hazards is called the "hardened safety core" (HSC); this was added as a new category to the bunkered systems. The HSC is a set of equipment and organizational measures to assure that basic safety functions are also available in extreme situations, thus guaranteeing the protection of the plant. The HSC, while not being an integrated "bunker", assures that the equipment, including pipework and supplies, is adequately protected against external hazards.

Table 3-4 | *Definition of key design feature "bunkered systems" with sub-features*

BUNKERED SYSTEMS		
Bunker design		Sub-features
None		Emergency control room
Hardened safety core (HSC)		Emergency control room
I	Bunkered systems withstanding conventional hazards of natural and human origin	Emergency control room
		Multi train
		Multi train with extended supplies
II	Bunkered systems withstanding natural hazards and a certain limited resistance against modern threats	Emergency control room
		Multi train
		Multi train with extended supplies
III	Bunkered systems withstanding both natural and modern threats	Emergency control room
		Multi train
		Multi train with extended supplies

3.2.4 Severe accident management

Severe accidents are events where, despite the existing safety systems, the capability to maintain adequate fuel cooling is compromised, resulting in significant damage to the fuel (core melt) and possibly compromising the containment. Under certain circumstances, the containment might also be assumed to fail or to be bypassed, potentially resulting in a major radioactive release to the environment.

To enhance the protection against these events, plants are 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 by the plant operating staff during a severe accident, to include:

- Preventing core damage
- Restoring failed equipment, or using any other available equipment to prevent or minimise the consequences of the accident
- Maintaining containment integrity for as long as possible
- Minimizing offsite releases

While attitudes towards severe accident management changed in the late nineties (with individual plants starting to introduce SAMG and/or some dedicated components), the measures to manage severe accidents were more or less implemented on the plants own initiative. After the Fukushima Daiichi accident, the worldwide attitude towards severe accident management changed significantly, and now all plants have SAMGs or are in the process of introducing them.

The post-Fukushima safety consideration pointed out that the existing definition of the key design feature SAM needs improvement to reflect the increased focus on severe accidents as well as the wide-scale deployment of mobile equipment for SAM. Mobile equipment was already considered before the Fukushima Daiichi accident. However, the EU Post-Fukushima Stress test highlighted the benefits of having mobile equipment and pre-prepared connections available.

In the first report, the definition of the key feature SAM included the features to maintain reactor vessel and containment integrity as well as hydrogen mitigation and control. In the enhanced benchmark methodology, these features have been assigned to the key feature "containment" as discussed in section 3.2.2, leaving the key feature SAM defined only by features and measures that are directly connected to SAM. The availability of dedicated or qualified instrumentation and control (I&C) for severe accidents was integrated in the new definition of the key feature SAM. The availability of mobile equipment for power and water supply was included in the sub-features. Table 3-5 indicates the key features and sub-features the Committee considered relevant.

Table 3-5 | *Definition of key feature "severe accident management" with sub-features*

SEVERE ACCIDENT MANAGEMENT			
SAM		Sub-features	
I	Use of existing means, no plant specific SAMG	On-site mobile equipment	Mobile power supply or Mobile water sources/water pumps
			Mobile power supply and Mobile water sources/water pumps
		Off-site storage of mobile equipment	
II	Use of existing means following plant specific SAMGs	On-site mobile equipment	Mobile power supply or Mobile water sources/water pumps
			Mobile power supply and Mobile water sources/water pumps
		Off-site storage of mobile equipment	
III	Use of existing means and dedicated hardware following plant specific SAMGs	On-site mobile equipment	Mobile power supply or Mobile water sources/water pumps
			Mobile power supply and Mobile water sources/water pumps
		Off-site storage of mobile equipment	

3.2.5 Design of spent fuel pool

Operating nuclear reactors of all types generate spent fuel that needs to be safely managed after it has been removed from the reactor core. It is stored in the spent fuel storage pool for a cooling period as it still generates heat from radioactive decay. Later, it will be transferred to a designated wet or dry spent fuel storage facility, where it awaits reprocessing or disposal.

The Fukushima Daiichi accident showed the importance of mitigating the risk of radioactive release from fuel in the spent fuel pool. There are, however, many different spent fuel pool features and plant design features that reduce the risks of radioactive release from spent fuel pools. The Committee concluded that detailed assessment of these features was not practicable and decided to consider the characteristic that from a safety perspective has tremendous impact on the spent fuel pool in the event of an accident: the actual location of the spent fuel pool.

In some plants, the spent fuel pool is located outside the containment, making it vulnerable to external hazards (e.g. aircraft crash, earth-quake). Spent fuel pools that are located within the containment are better protected; should fuel damage occur (e.g. due to a loss of cooling) within the spent fuel pool, the resulting radioactive release would be confined to the containment. This is not always the case when the spent fuel pool is in a separate building. These considerations lead to the key features for spent fuel pool presented in Table 3-6.

Table 3-6 | *Definition of key feature "spent fuel pool features"*

	SPENT FUEL POOL
I	Spent fuel pool located outside the containment
II	Spent fuel pool located inside the containment

3.2.6 Final scoring table for the evaluation of the safety of design

Table 3-7 (see next page) summarizes the final evaluation of the safety of design, taking into account the considerations for each key feature and sub-feature as presented in the subsections above.

3.3 Pilot study results

A pilot study on 20 reactors was conducted to determine whether the required data for all considered features are available and to test the scoring scheme. In particular, the sensitivity analysis undertaken during the pilot study gave the Committee confidence that the enhanced methodology was appropriate for the more refined evaluation of design safety the Committee was envisaging to reflect post-Fukushima insights on nuclear safety. From the pilot study, the Committee concluded that:

- The reactors considered had scores distributed across a wide range, with a more significant number of reactors ranking in the lower range of scores and a smaller number of reactors ranking towards the upper range.
- The enhanced methodology showed more sensitivity to the reactor design characteristics than the one in the first report.
- The scoring system was not sensitive to reactor age; both older and newer reactors had high scores as well as low ones.
- The results were not dominated by a single key design feature, nor are they unresponsive to any of them.

These conclusions were reconfirmed in the final design benchmark.

Table 3-7 | *Final scoring table for the evaluation of the safety of design*

REDUNDANCY AND DIVERSITY			
Core cooling system			
I	PWR	2x 100% or less ECCS redundancy, no diversity in AFWS	1
	BWR	No redundancy in HPCI; 2x 100% or 3x 50% LPCI; 1X 100% CS	
II	PWR	More than 2x 100% ECCS redundancy, no diversity in AFWS OR 2x 100% ECCS redundancy, diversity in AFWS	3
	BWR	Redundancy, no diversity in HPCI; 4x 50% or 3x 100% LPCI; 1x 100% CS OR No redundancy in HPCI; 4x 50% or 3x 100% LPCI; 2x 100% CS	
III	PWR	More than 2 x 100% ECCS, diversity in AFWS	4
	BWR	Redundancy and diversity in HPCI; 4x 50% or 3x 100% LPCI; 2x 100% CS	
Ultimate heat sink			
I	No redundancy, no diversity		0
II	Redundancy (availability of large water stocks on-site or alternative ultimate heat sink)		1
III	Redundancy and diversity (availability of large water stocks on-site and an alternative ultimate heat sink)		2
AC/DC power supply			
Layers of power supply		for each layer	0.25

CONTAINMENT				
	Containment design		**Sub-features**	
I	Pressure suppression containment (all types) or full pressure dry single containment	1	Features to control hydrogen	1
			Strategies for in- and ex-vessel retention of molten core	1
			External reactor vessel cooling	0.25
			Containment filtered venting	1
II	Full pressure double wall containment	4	Features to control hydrogen	1
			Strategies for in- and ex-vessel retention of molten core	1
			External reactor vessel cooling	0.25
			Containment filtered venting	1
III	Full pressure double wall containment capable of withstanding large aircraft crashes	6	Features to control hydrogen	1
			Strategies for in- and ex-vessel retention of molten core	1
			External reactor vessel cooling	0.25
			Containment filtered venting	1

BUNKERED SYSTEM				
	Bunker design		**Sub-features**	
	None	0	Emergency control room	2
	Hardened safety core (HSC)	4	Emergency control room	2
I	Bunkered systems withstanding conventional hazards of natural and human origin	4	Emergency control room	2
			Multi train	1
			Multi train with extended supplies	1.5
II	Bunkered systems withstanding natural hazards and a certain limited resistance against modern threats	5	Emergency control room	2
			Multi train	1
			Multi train with extended supplies	1.5
III	Bunkered systems withstanding both natural and modern threats	6	Emergency control room	2
			Multi train	1
			Multi train with extended supplies	1.5

SEVERE ACCIDENT MANAGEMENT					
	SAM		**Sub-features**		
I	Use of existing means, no plant specific SAMG	0	On-site mobile equipment	Mobile power supply or Mobile water sources/water pumps	1
				Mobile power supply and Mobile water sources/water pumps	2
			Off-site storage of mobile equipment		0.5
II	Use of existing means following plant specific SAMGs	1	On-site mobile equipment	Mobile power supply or Mobile water sources/water pumps	1
				Mobile power supply and Mobile water sources/water pumps	2
			Off-site storage of mobile equipment		0.5
III	Use of existing means and dedicated hardware following plant specific SAMGs	2	On-site mobile equipment	Mobile power supply or Mobile water sources/water pumps	1
				Mobile power supply and Mobile water sources/water pumps	2
			Off-site storage of mobile equipment		0.5

SPENT FUEL POOL		
I	Spent fuel pool located outside the containment	0
II	Spent fuel pool located inside the containment	2

3.4 Results and conclusions

Figure 3.1 provides a graphical representation of the distribution of reactors among the score ranges. Scores assigned to the reactors cover the range of 6.5-32 in steps of 0.25. The wide score distribution shows that the extended benchmarking methodology, including post-Fukushima safety considerations, discriminated better among different design solutions.

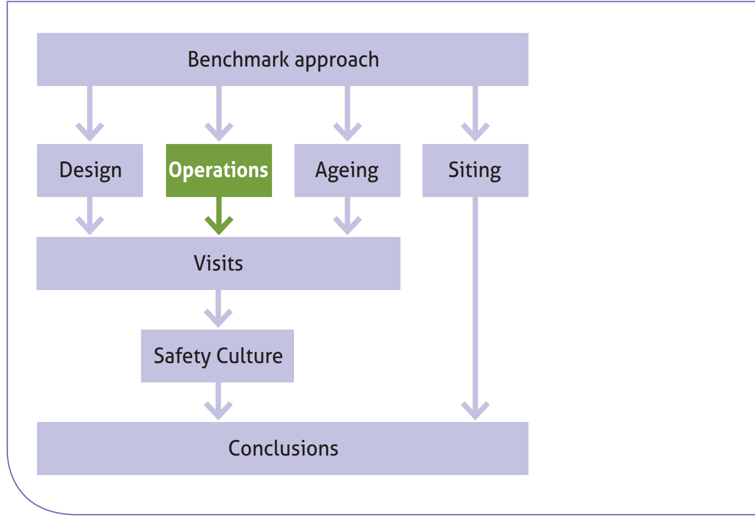
Most reactors score in the range of 8-20. Reactors with a score of 18.5 and above fall in the top 25% ("very safe reactors"), as shown in Figure 3-1. The lower boundary of the top 25% is formed by a large group of 26 reactors with an identical score of 18.5. Only 11% of the reactors score above 20. With a score of 26.25, KCB ranks well within the top 25%.

Figure 3-1 | *Distribution of reactor scores and top 25% group for design safety*

4 Evaluation of Operational Safety

4.1 Introduction

For evaluating safety in plant operations, the Committee used the same two-step approach developed during the first benchmark period. In the first step, the top 25% best-performing plants were selected based on performance indicators. These indicators reflect operational (and not only safety) performance during the past operating period but do not assure the same performance in the future. The Committee concluded that it was equally important to assess that safety performance is the result of well-defined and controlled processes directed by plant management in step two. Considering the amount of information needed for detailed process analysis, this was realistically only feasible for a sample of the plants. To determine whether KCB's performance in the management of operations is like that of the other 25% best-performing plants in operations, the Committee decided to compare KCB through detailed analysis with an properly selected sample of peers.



4.2 First step: Evaluation of operational safety

4.2.1 Introduction

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

The nuclear industry has instituted an internal reporting system to improve performance quality; it is designed to monitor operations based on several performance indicators. Most of these performance indicators are relevant for evaluating the safety performance in plant operations. The reliability of this reporting system is regularly checked during peer reviews. 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 due to 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 into 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.

4.2.2 Selection of the 25% best-performing plants

The Committee was provided confidential access to the performance indicators specified above and used them to define the 25% best-performing plants of the 237 the Committee had to consider, based on the performance indicators. To do so, the Committee used weighting factors to combine the performance indicators into a composite number.

Since scores in this type of monitoring systems can be substantially affected by one-off items, the Committee decided to use multi-year averages. The results are shown in figure 4.1, normalized to a maximum score of 100.

KCB is well within the top 25% best-performing reactors based on performance indicators.

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

The used indicators form the basis of the total operational performance, which includes operational safety, but also other aspects of operational performance. Having more or longer planned outages for maintenance or installing safety upgrades will for example negatively impact the performance rating due to lower availability of the reactor. Consequently, the rating is not purely a safety rating and the evaluation can only be considered as a first level indication of the safety performance of KCB.

A more in-depth evaluation was needed to obtain insight into whether KCB's safety performance is the result of a well-controlled process. To do so, the Committee performed in-depth process analysis of the performance of the nuclear power reactors. Reactors having a good operational performance rating are more likely to have good safety performance as well. Thus, to assess whether KCB is comparable to the top performers, it was deemed satisfactory to perform this in-depth analysis for a limited number of peer reactors from the top 25%.

4.3 Second step: Evaluation of operational safety

4.3.1 Introduction

The process analysis in the second step of the Operational Safety Benchmark focused 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. This analysis required a good understanding of how the plants are operated and managed. The Committee concluded that for process analysis of operation, maintenance and safety management, the only appropriate derestricted information available was from the reports of the Operational Safety Review Team (OSART) programme of IAEA.

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

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

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 derestricted ninety days after its issuance, unless the host country requests otherwise.

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

4.3.2 Methodology

The nuclear plants that were included in this detailed evaluation were selected using several criteria:

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

The final selection of 10 peers, besides KCB, was based on the expert opinion of the Committee and, in view of the desired geographical spread, included three plants that were (just) outside the top 25% group of best-performing plants determined in step 1 of the operational safety assessment (see Section 4.2).

While the comparison based on numerical performance indicators was rather straightforward, a process evaluation implied an understanding of the philosophy of nuclear power plant operation and the organisational, management, and operational practices that can vary significantly across the countries and operating organizations.

The Committee decided that this evaluation would require:

- Consideration and evaluation of findings of the OSART related with weaknesses (areas for improvement) identified.
- An assessment by judging the 'importance to safety' of each OSART finding.
- A ranking of KCB against the other plants in the peer group.

Categorization and classification of OSART findings

To evaluate operational safety, the Committee adopted the same method developed, tested and used in the first benchmarking period. The method involved dual considerations: the expert judgment of the OSART team and the expert judgment of the Committee on the importance for safety of the OSART findings. The latter involved identifying suitable parameters to categorize the plants' weaknesses (e.g. areas for improvement) as assessed and documented by the OSART team. All OSART findings for each of the 11 plants (peer group and KCB) were classified by their importance for safety.

OSART missions review performance in different safety areas. OSART guidelines define nine core operational safety areas and six additional safety areas that can be selected by specific missions. To make the assessment internally consistent, only the nine core operational safety areas, that were assessed for all 11 plants were considered by the Committee:

- Management, organization and administration;
- Training and qualification;
- Operations;
- Maintenance;
- Technical support;
- Radiation protection;
- Chemistry;
- Operating experience; and
- Emergency planning and preparedness.

In the OSART guidelines, these areas are further subdivided. For example, in Operations seven sub-areas are evaluated:

- Organization and functions;
- Operations facilities and operator aids;
- Operating rules and procedures;
- Conduct of operations;
- Work authorizations;
- Fire prevention and protection programme; and
- Management of accident conditions.

In total, several dozen sub-areas are defined by the OSART Guidelines. These precise sub-areas are delineated to ensure a comprehensive review of each plant and indicate the areas for improvement at a sufficient level of detail for the plant management to be able to understand where and what type of corrective or improvement measures are warranted.

From the safety point of view, no prioritization of the nine areas or their sub-areas was attempted, because acceptable performance in all of them is needed to ensure the safe operation of the plant, while deficiencies in any one of them indicates deficiencies in operational safety.

The safety significance of each OSART finding (recommendation, suggestion or note) was objectively categorized, in the same way as for the first report, based on consideration of different aspects as safety management, defence in depth, safety culture, etc. The categorization consists of five groups, listed here below in 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 impact safety, including: plant organization, 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 overall weakness in 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 competence and skills of operators, operating practices, status of systems and components, quality of procedures and adequacy of their usage. The findings in this group could be an indication of deficiencies affecting equipment and personnel, undermining prevention capabilities and/or plant safety. Thus the reason that the findings in this group were given the second highest weighting factor.

- **Human performance**
  Operational experience from the nuclear industry demonstrates that 70% of events in nuclear power plants are caused by inadequate human performance. Findings related to human factors or performance could 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. This group includes a range of issues from training and qualifications to performance and rectification of identified deficiencies. All findings regarding human performance were included in this group.

Group III

- **Functioning of plant systems and equipment, plant integrity**
  The findings in this group would be those related to the functioning of plant's systems and equipment and/or integrity of plant structures, which provide 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 were given a lower rating than the previous group.

- **Management of deviations and failures**
  OSART missions typically review 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 they lead to more serious situations. Examples of findings include operational issues, ability to timely identify and correct the faults and deficiencies related to surveillance procedures. Being preventive in nature, findings in this group were given a lower rating than the previous group.

Group IV

- **Personnel safety**
  One element of the OSART is devoted to the assessment of the radiation protection and industrial safety programmes. Even though these aspects are important safety elements, their impact primarily affects plant personnel. As the focus of the Borssele Benchmark assessment is on impacts on the public and the environment, the findings within this group could be considered 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 impacts 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 reviews on-site emergency preparedness. Any findings in this area would not be 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 could be 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. These findings would be 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 could be considered insignificant).

Besides 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; and
- **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 were made can vary in significance.

Table 4-1 | *Final ranking matrix for the evaluation of operational safety management*

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				

A third layer of classification was therefore introduced to account for the contribution of each issue to safety performance. This classification was made based on expert judgment and included three levels: high (H), medium (M) and low (L) safety significance.

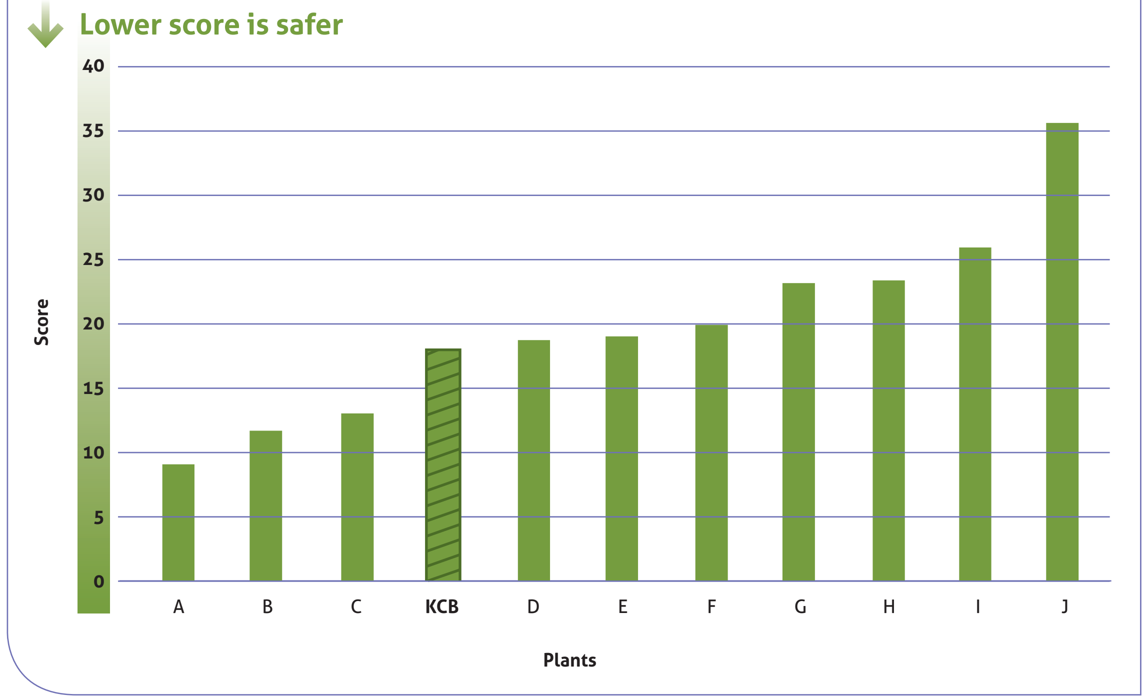
This threefold categorization and classification is represented in the resulting ranking matrix (see Table 4-1), which combines all three of the levels discussed, of which the most important is the one reflected by the five groups of evaluation criteria. The second is the OSART categorization reflected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

4.4 Results and conclusions

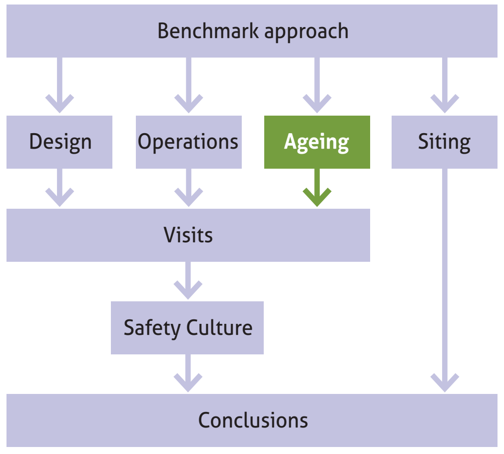
The outcome of the evaluation of operational safety management at the peer plants are presented in Figure 4-2. The scoring system is such that a lower score means a higher level of safety.

The scores obtained in this evaluation range from 9.10 (highest operational safety), to 35.65 (lowest operational safety). KCB is situated in the higher middle of the range (fourth best plant in the peer group). This supports the conclusion that KCB's safety performance in plant operations, maintenance and safety management is comparable to its peers in the top 25% in operational performance.

Figure 4-2 | *Results of the evaluation of operational safety management in the peer group*



5 Evaluation of Ageing Management



5.1 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 wear, fatigue, erosion, microbiological fouling, corrosion, embrittlement, chemical or biological reactions and combinations of these processes. Since ageing impacts both nuclear power plant safety and performance, effective management of ageing is a key element in the safe and reliable operation of nuclear power plants, especially for long-term operation (LTO).

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

Like the approach taken in the review of operation, maintenance and safety management, the ageing review focused on the question to what extent ageing management was a well-managed process. The Committee developed a new ageing benchmark approach to replace the method used for the first report. The reason for this decision was that in recent years the internationally consistent methodology of the IAEA Safety Aspects of Long-Term Operation (SALTO) had been refined and used on a larger number of plants, allowing the Committee to conduct the evaluation in a way comparable to Operational safety evaluations that uses the IAEA OSART results.

The new methodology considered safety aspects of ageing management for long-term operation that were assessed in the IAEA SALTO missions and OSART missions with a SALTO module. The areas covered by the IAEA missions are consistent with areas reviewed in the first report of the Committee.

The SALTO peer review addresses the following areas:

- Organization and functions, current licensing basis, configuration/modification management;
- Scoping and screening and plant programmes relevant to LTO;
- Ageing management review, review of ageing management programmes and revalidation of time limited ageing analyses for:
  - Mechanical components
  - Electrical and instrumentation and control components
  - Civil structures;
- Human resources, competence and knowledge management for LTO (optional);
- Management, organization and administration, training and qualification, technical support, etc. (optional).

The scope of the ageing management review, as in the first benchmark period, consisted of a comparison of KCB's ageing management programme against ageing management programmes of five peer plants.

5.2 Selection of ageing management peer group

The ageing management peer group KCB was compared against, is composed by a selection of five reactors according to the following criteria:

- Plants should be in or in preparation for LTO.
- The peer group should include different types of water-cooled and water-moderated reactors: PWR, PHWR and BWR.
- The peer group should include plants geographically spread over the benchmark area: European Union, USA, and Canada.
- Reports on IAEA SALTO missions or OSART missions with SALTO module should be available.
- To the extent possible, peer plants should have had a SALTO review within about one year of the latest KCB's SALTO review that took place in February 2014.

The IAEA initiated the SALTO mission programme ten years ago. In the Committee's view, plants having undergone a SALTO mission indicates that ageing management has a relatively high priority at the plant. The final selection of five peers, besides KCB, was based on the expert opinion of the Committee.

5.3 Methodology

The methodology employed considered safety aspects of ageing management for LTO that were assessed in the IAEA SALTO peer review service. The ageing management review methodology involved consideration from two points of view: the on-site evaluation of the SALTO team during the mission, and the expert judgment of the Committee evaluating the findings of the SALTO teams. To combine these judgements and obtain an aggregate score for a plant, each SALTO finding was sorted into the following three categories:

- Four groups based on the Committee's assessment of SALTO areas of review.
- SALTO prioritization of issues into recommendations and suggestions.
- Safety significance of issues based on the Committee's assessment.

The total score of a plant represented a composite judgement on the quality of ageing management arrangements for LTO, facilitating an overall ageing management programme benchmark comparison of KCB with the peer plants.

The Committee first combined the SALTO areas of review into four groups that were consistent with the areas addressed in the first report. The Committee reviewed and assigned the SALTO findings to the following groups:

Group I

- **Overall ageing management**
  Issues in Group I are related to the quality of governance documents of the overall plant ageing management programme, i.e. documentation of plant policy, organization and methodology for ageing management that should provide direction for effective ageing management. Because of its overall impact on plant ageing management and LTO, issues in this area were ranked at the highest level in the scoring scheme.

Group II

- **Scope of ageing management for LTO**
  Issues in Group II are related to the completeness of the scope of ageing management for LTO, including the scoping process and criteria, and the list of structures and components included in an ageing management programme for LTO. These issues were ranked in the middle of the scoring scheme.

Group III

- **Ageing management programmes for specific structures and components and specific ageing mechanisms**
  Issues in Group III are related to the extent to which ageing management programmes for specific structures and components and specific ageing mechanisms were consistent with international generic ageing lessons learned.

Group IV

- **Time limited ageing analyses**
  Issues in Group IV are related to the quality of time-limited ageing analyses. Groups III and IV issues were set at the lowest level ranking in the scoring scheme because the benchmark focused on the ageing management programme and not on ageing itself. Group III and IV issues reflect the current ageing management situation at the systems, structures and components level.

The following scores were assigned to different groups based on the Committee's assessment of their respective safety significance relating to ageing management.

Group		Score
I	Overall ageing management	3
II	Scope of ageing management for LTO	2
III	AMPs for specific SCs and specific ageing mechanisms	1
IV	Time limited ageing analyses	1

The second step in the process was to consider the valuable expert opinion of the SALTO team, based on direct information from the plant. The SALTO team prioritizes the issues identified during the SALTO mission into recommendations and suggestions.

- **Recommendations** are advice on what improvements in safety aspects of LTO should be made in the activity or programme where performance falls short of IAEA Safety Standards, Safety Reports or proven, good international practices. Absence of recommendations can be interpreted as performance corresponding with proven international practices.

- **Suggestions** are advice on what improvements in safety aspects of LTO would make a good performance more effective, to indicate useful expansions to existing programmes and to point out possible superior alternatives to on-going work.

The following weighting factors were applied to the scores of each of the four groups based on their SALTO prioritization:

- 100% for recommendations
- 50% for suggestions

In the third step, the Committee rated the safety significance of the findings identified by SALTO by assessing their effect on safe plant operation, i.e. potential degraded performance or failures of systems, structures or components and their impact on defence in depth and the fundamental safety functions of reactivity control, core cooling and confinement of radioactivity.

These three steps were combined in the resulting overall scoring matrix shown in Table 5-1.

Table 5-1 | *Final scoring matrix for the evaluation of ageing management*

Grouping	1st level Score	2nd level SALTO priorization	3rd level Safety significance		
			High	Medium	Low
Group I Overall ageing management	3	R Score	100% 3	80% 2,4	60% 1,8
		S Score	50% 1,5	40% 1,2	30% 0,9
Group II Scope of ageing management for LTO	2	R Score	100% 2	80% 1,6	60% 1,2
		S Score	50% 1	40% 0,8	30% 0,6
Group III & IV Ageing management programmes for specific systems, structures and components and specific ageing mechanisms & time limited ageing analyses	1	R Score	100% 1	80% 0,8	60% 0,6
		S Score	50% 0,5	40% 0,4	30% 0,3

Unlike the full SALTO missions, the SALTO module reviews within an OSART are performed by a single expert. Therefore, relevant supplementary information was used, consisting of information from regulatory reviews of licensee ageing management programmes and maintenance module reviews that were included in all full SALTO missions.

Ageing management-related issues identified in maintenance module reviews within OSART and issues identified in regulatory reviews of licensee ageing management programmes were extracted and processed using the overall scoring matrix to assign a score to each issue. When a regulatory or maintenance module findings was duplicated, the finding was considered only once in the total reactor score.

To facilitate comparability of reactor ageing management programmes for LTO, the KCB SALTO follow-up review of February 2014 was used. To account for improvements that were made since the initial SALTO mission of May 2012, the following modification factors were used based on the SALTO report resolution status:

- **Insufficient progress to date**
  no change in score
- **Satisfactory progress to date**
  50% reduction in score
- **Issue resolved**
  100% reduction in score

This resets KCB's ageing management baseline from May 2012 (input used in the first report) to the current five-year reporting period.

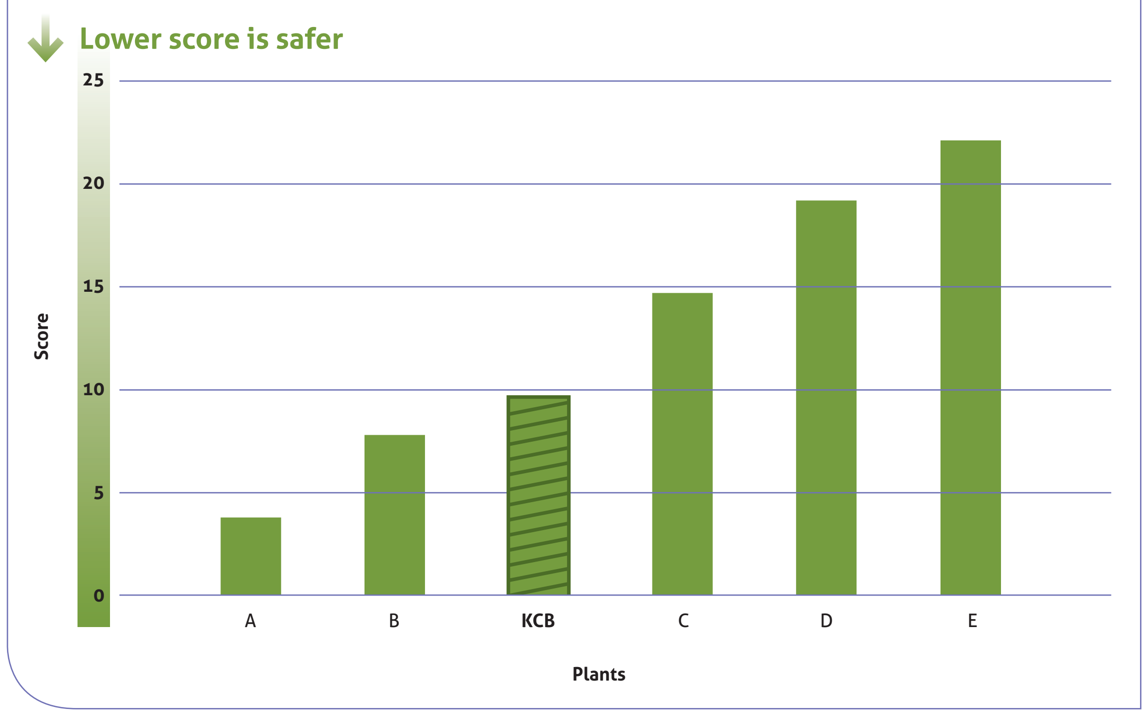
The new ageing management review methodology was tested in a pilot study for KCB and two other plants. The pilot study included a sensitivity analysis that involved varying the weight of suggestions relative to that of recommendations and varying the weighting factors assigned to the safety significance. The study showed that the ratio between the significance of the recommendations and that of the suggestions was not a dominating parameter in the scoring scheme. The scores changed but the ranking remained the same. Similarly, varying the weighting factors for high/medium/low safety significance of recommendations and suggestions reduced the total scores of all plants by 10 – 15%, but the ranking remained the same. Thus, overall, the sensitivity study confirmed the robustness of the new ageing management review methodology.

5.4 Results and conclusions

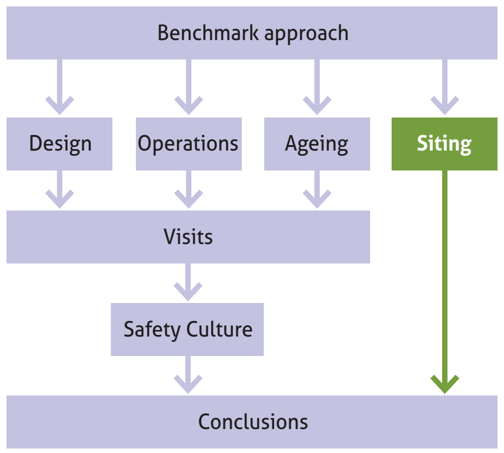
The methodology discussed in 5.3 was used to analyse the six plants in the ageing management peer group. Figure 5-1 shows the total score for each plant, with lower scores indicating better ageing management programmes.

For most plants, including KCB, the most significant deficiencies were found in group I (overall ageing management). The performance of KCB shows a clear improvement in the overall ageing management score compared to the first benchmark report. The results show that overall KCB was the third best in the peer group. The Committee concluded that ageing management of KCB is comparable to that of its peers.

Figure 5-1 | *Total score for each reactor in the ageing management peer group evaluation*



6 Evaluation of Siting



6.1 Introduction

Siting refers to the process of evaluating the suitability of a location for a nuclear facility. In this process, events are identified that can jeopardise plant safety. These events can be of natural or human induced origin and include earthquakes, aircraft crashes, explosions, releases of hazardous gases, extreme meteorological conditions, floods, cyclones, forest fires, etc. These events are called external hazards, as they originate from outside the plant and the event itself (earthquake, high-water level) cannot be influenced by the design of the plant. The magnitude and probability of occurrence of external hazards are evaluated for plant design purposes so that the plant can be sufficiently designed to withstand these hazards.

The Fukushima Daiichi accident, following a major earthquake, and a 15-metre tsunami, clearly demonstrated that siting of a nuclear power plant can have significant impact on its safety. The Fukushima Daiichi accident prompted increased international attention towards the need for realistic and complete assessments of external hazards. External events were also a major focal point of the EU Post-Fukushima Stress test.

For benchmarking the impact of external hazards on the safety of different nuclear power plants, it is not enough to compare the probability and magnitude of the external hazards that could occur at the different locations. Higher probability and more intense external hazards at a site, such as high risk of flooding, earthquakes or tsunami's, do not necessarily indicate lower safety of the plant. The safety implications of these external hazards on a plant depend on the design of the plant and its ability to withstand these hazards. If all hazards are properly considered in the design, the plant should be well protected against the hazards at the site, and these hazards should not significantly endanger the safety of the plant.

6.2 Methodology

The Committee compared how nuclear power plants were protected against external hazards at their sites, which revealed large variations in methodologies used to identify and evaluate the hazards and associated risks, and wide variations in the type and magnitude of the relevant hazards at each site. Due to these differences, the Committee concluded that a meaningful comparison, even among a selected sample of plants, was not possible.

The Committee decided to focus the evaluation on the siting aspects of KCB specifically.
The goal was to assess whether the siting risks at KCB are assessed in line with good international practices and considered in its design, and to assess whether these external hazards pose a risk to KCB.

For this evaluation, the Committee used information from the EU Post-Fukushima Stress test, complemented by the underlying safety evaluations by KCB, as well as evaluations performed as part of their most recent 10-year periodic safety review.

6.3 Evaluation of events

6.3.1 Earthquake

Following the EU Post-Fukushima Stress test and as part of the most recent 10-year periodic safety review, seismic risks at KCB were reassessed and a reference level earthquake was defined, higher than the design basis earthquake.

This reference level earthquake was subsequently used in a seismic margin assessment. In an assessment of this kind, the seismic robustness of all systems and components is undertaken. Should any components be identified with a seismic capacity below the reference level, these were modified to increase their seismic robustness. For components with a small margin, measures were identified to increase their robustness. Finally, for those components with a large margin compared to the seismic loads of the reference level, no action was taken.

This methodology is in line with international best practices for existing power plants.

6.3.2 Flooding

The risk of flooding always had attention at KCB. As such, flooding risks were accounted for from the beginning of KCB's design process. Following the stress test and as part of the most recent 10-year periodic safety review, flooding risks at KCB were reassessed. Recent studies even considered extreme flooding scenario's with very high-water levels and a breach of the dyke directly opposite KCB.

These studies showed that water levels at the KCB site would still be more than two meters below the level that the design of KCB can withstand.

6.3.3 Other extreme weather conditions

As part of the stress test the robustness of KCB against extreme weather conditions (e.g. extreme temperatures, high winds, and excessive snowfall) was assessed.

The assessment concluded that KCB is well protected against those weather conditions to be expected in the Netherlands, including strong winds, extreme rain or snow and extremely low or high temperatures.

6.3.4 Airplane crash

The airspace surrounding the KCB is restricted airspace. The airports located near KCB were considered in the risk assessment, as most crashes occur during landing or take-off. The only airport located near the KCB, at approximately 10 km, serves only light aircrafts. Like most plants in EU, the USA and Canada, several measures were provided for at KCB to protect the plant against aircraft crashes or to mitigate their consequences.

The assessment concluded that besides the strength of the reactor building and the shielding by the surrounding buildings and dyke, the spatial separation of buildings housing safety systems, contributes to the availability of at least one redundancy train for core cooling after an airplane crash.

6.3.5 Shipping accident on the Westerschelde

The assessment concluded that for KCB, industrial and military facilities as well as road and rail transportation routes are far enough away that the risk from pressure waves or toxic releases from these activities would be negligible.

The largest threat to KCB consisted of shipping activities on the Westerschelde, including gas tankers carrying flammable or toxic gases. Shipping accidents resulting in the loss of large quantities of explosive gases or toxic substances were considered in the safety analysis of KCB and KCB was found to able to withstand these events.

6.4 Conclusion

The Committee concludes that the siting risks at KCB were well investigated in line with modern international good practices and requirements for existing nuclear power plants, and considering the findings from the Fukushima Daiichi accident. The Committee is confident that siting does not negatively impact the overall safety ranking of KCB.

7 Site visits

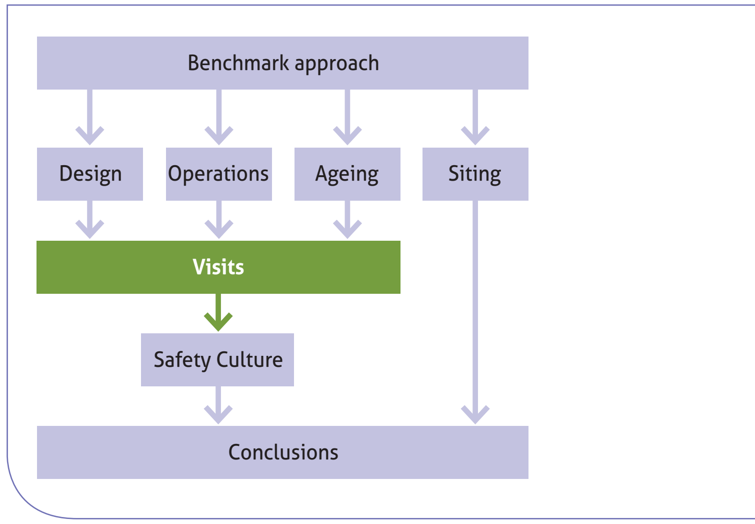
7.1 Site visit Objectives

The first objective of the site visits was to check whether the conclusions reached through the desktop analysis were supported by the impressions obtained from the plant visit of how the plants were managed. In other words, whether the strengths and weaknesses, as compared with KCB, that were identified in the peer review process were in line with the impressions obtained during the plant visits.

The second objective was to assess the safety culture at the power plant (see next chapter).

The plants selected for the site visits were chosen from the peer group used for the process analysis of operation, maintenance, and safety management. In the selection, attention was given to geographical distribution. In total five plants, beside KCB, were visited.

The site visits were carried out after finalizing the desktop analyses. Each visit in Europe was conducted by three Committee members and in North America by two.



7.2 Site visit Organisation

The visits consisted of two parts, one being the presentation by the host plant management, followed by discussion or clarification on several topics, and the other being a plant tour. The Committee asked the plant management to cover in their presentation 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

- **Safety Culture**
  - The way the plant is led and safely managed
  - The way the organization deals with (elements of) the safety management system
  - The way the organization involves and motivates its people

- **Post-Fukushima modifications and other safety upgrades.**

- **Information on the safety re-evaluations performed after the EU Post-Fukushima Stress test, and scope and methodology adopted.**

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 (AM)
- Conditions of safety related systems, in particular the systems to be utilised in 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.

In general, the information received and the insights gained during the visits made it possible for the Committee to get an overall impression of the way the plant is managed, and that the information can be meaningfully used for the purposes of comparison among the peer plants.

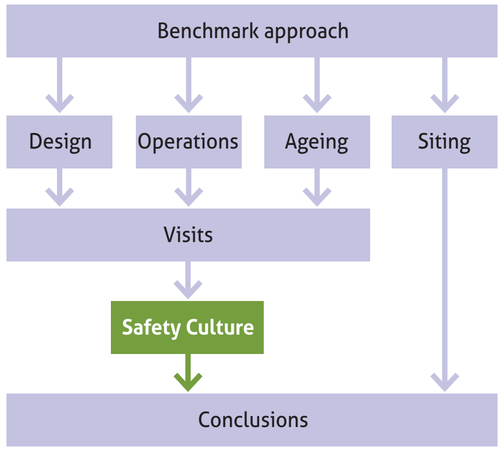
7.3 Results and conclusions

From the overall result of the site visits, the Committee concluded that their impressions were in line with the results from the desktop reviews and that KCB is in line with international best practices and requirements in terms of the items examined.

Below some observations of the Committee that were the result of the visits. Specific observations on safety culture are addressed in the next chapter.

- Compared to five years ago, the Committee noticed an increased attention to improve safety awareness and safety culture. However, the approaches chosen differ from plant to plant, also because of cultural differences or whether the plant operates stand alone or in a plant with more units.
- Post-Fukushima safety improvements have taken place at all plants. Some differences were noticed, however, among plants in North America and in Europe where the stress test contributed to a more harmonized approach.
- Both operational safety management and ageing management were well embedded in the operation programmes.
- To improve efficiency and safety in operations and maintenance, an increase in the use of simulators to train operators was noticeable. However, differences can be observed between plants, in the way simulators were used and operators were trained.

8 Evaluation of Safety Culture



8.1 Introduction

In the first report, the Committee stated that improving safety awareness and safety culture received a great deal of management attention in nuclear power plants and that it was evident that translating this concept into effective measures was not an easy task. The Committee noted that it takes time to convince the organization of the importance of the concept and that cultural differences play a role in translating it into effective measures. As a result, the approaches chosen differed, as well as the progress plants made in this area.

The Committee decided to give safety culture more attention in the second report, but also, realized that it was very difficult to assess. The IAEA defines safety culture as follows:

"Safety Culture is that assembly of characte-ristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receives the attention warranted by their significance."

The IAEA started to organize Independent Safety Culture Assessments and sometimes safety culture is part of an OSART review mission; however, these IAEA services have not yet been utilized by a sufficient number of plants to be useful for benchmarking. The Committee, therefore, decided to give Safety Culture a prominent place during the plant visits.

8.2 Methodology

The Committee developed a custom tool based on eleven indicators of safety culture quality (see Table 8-1, next page). This tool provides a consistent approach to management discussions and walk downs in the visited plants. The tool comprises eleven items, grouped into three closely related clusters:

- **The way the plant is led and safely managed**
  - Leadership
  - Safety and productivity
  - Safety and management
  - External contractors working on site

These items address the link between safety culture and the broader organizational culture. This is closely related to safety leadership and the way safety is integrated into production activities.

- **The way the organization carries out (elements of) the safety management system**
  - Procedures
  - Valuing and following up internal and external audits and inspections
  - Registration of deviations
  - Learning driven safety culture

The functioning of the safety management system is mainly a matter of dedicated plans and actions, and not primarily a matter of the safety culture, but there are many ways in which the functioning of safety management is influenced by the culture of the organization.

- **The way the organization involves and motivates its people**
  - Commitment and participation of the workforce
  - Safety communication
  - Physical and mental fitness

Most directly, safety critical processes occur at the shop floor. The way safety culture is 'alive' in the awareness and actions of the people, especially on the shop floor, is therefore vital for a proficient safety culture.

Table 8-1 | *The elements of the survey*

Leadership
To what degree is senior management committed to safety?
To what degree is senior management competent with respect to safety?
To what degree is senior management visible on the shop floor to promote safety proactively?

Safety and Productivity
To what degree do actions of senior management show a good balance between safety and production?
To what degree is it ensured that (on-going) cost reductions do not gradually undermine safety margins?

Safety & Management
To what degree does senior management sees itself as responsible for the causation of incidents and deviations?
To what degree do the supervisors motivate their team to improve safety?
To what degree is safety integrated into processes of organisational change?

External people working on site (Contractors and subcontractors)
To what degree does the company improve (process) safety jointly with their contractors and subcontractors?
To what degree does senior management demonstrate its responsibility for the safety of contractor and subcontractor personnel?

Procedures
To what degree are the inspecting authorities regarded as helpful to improve safety?

Valuing and following up internal and external audits and inspections
To what degree are the inspecting authorities regarded as helpful to improve safety?
To what degree does the organisation timely implement all findings from safety audits and inspections?
To what degree does senior management use the outcomes of audits and inspections as input for management reviews?

Registration of deviations
To what degree are employees effectively stimulated to reflect on safer work practices?
To what degree do employees see it as important to report deviations?
To what degree does the reporting system function well (frequency of use, timely feedback, etc.)?

A learning driven safety culture
To what degree does the plant implement lessons learned from deviations and incidents?
To what degree does the organisation monitor the safety climate (the perceptions and attitudes of the employees that are relevant for safety)?
To what degree does the plant actively identify lessons learned from deviations occurring in other organisations?

Commitment of the workforce and participation
To what degree do employees employ initiatives to improve safety?
To what degree is safety 'alive' at the shop floor?
To what degree are employees actively involved by the managers in dealing with safety issues?

Safety communication
To what degree does the organisation communicate a clear and consistent safety message?
To what degree do managers and employees freely exchange information about safety?

Physical and mental fitness of the workforce
To what degree does the organisation address factors that might affect the physical and mental fitness of its workforce (overtime work, lifestyle, stress, conflicts, alcohol, drugs, etc.)?

8.3 Results and conclusions

Safety culture is a multi-faceted and multi-layered concept. Nevertheless, using the safety culture assessment tool was very helpful in structuring a systematic assessment of safety culture during the limited plant visit timeframe. The Committee is convinced that by working systematically and consistently with the customized tool, a meaningful comparison among peer plants could be made.

The Committee noted that at all the visited plants, safety culture receives much more attention than it did five years ago. However, large differences in methodology and ways of implementation continue to exist from plant to plant.

The Committee noted that KCB is very active in this area. Based on the results of the assessment undertaken, the Committee concludes that safety culture at KCB is equal or better than at the nuclear power plants visited.