2023

Safety benchmark of Borssele Nuclear Power Station

Report of the Borssele Benchmark Committee

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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 third report of the Committee.

Since the Committee started its work, some reactors have been permanently shut down and some others started operation. Each report of the Committee includes only the reactors still in operation by the cut-off date set by the Committee for its assessment. For this third report the cut-off date was set at 31 December 2021 and the list of reactors involved in the benchmark contains a total of 220 reactors.

To establish an expert opinion on the safety level of KCB, as compared with the other 219 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 express 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 fully 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 220 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 third report, the Committee retained the overall structure of the methodology previously developed. Compared with the second report no major adaptions are applied.

Schematically the Committee opted for the approach as shown in the figure on page 6.

This methodology contains a separate safety assessment of:

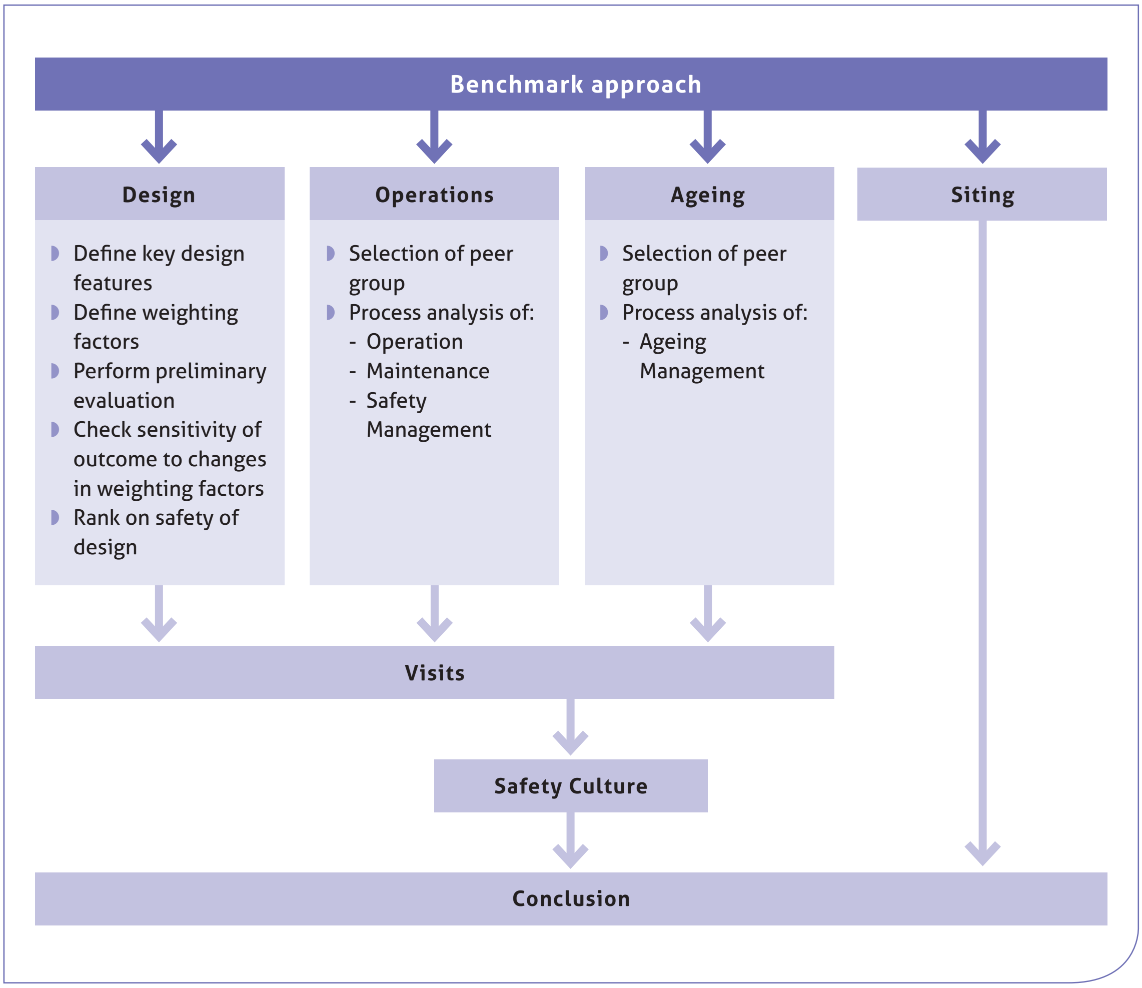
- Reactor design (including reactor upgrades)
- Reactor operations (covering operation, maintenance, safety management)
- Ageing management
- Siting
- Safety Culture

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

Conclusion

Using the developed methodology the Committee compared the safety of 220 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.

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



Design safety

Under all circumstances for nuclear reactor safety it is essential to assure:
1) Reactivity control
2) Core cooling (heat removal)
3) 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 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. In particular a fifth key design feature was added (spent fuel pool) and sub-features were included which allowed for a better differentiation among the safety levels achieved.

For the third report, the Committee again reviewed the methodology to assess whether there are some reasons to update the methodology. The Committee concluded that since the second benchmark there were no major changes in the safety philosophy that would warrant altering the methodology.

For each of the following 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

All 220 reactors were evaluated with the 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, license renewal applications, etc.).

The outcome was a score per reactor, which was 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.
- 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 indicate that from the design point of view, KCB remains well within the top 25% safest reactors.

As in 2013 and 2018, 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 and applied also in the second report. 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 220 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.

Scores in performance indicators can be substantially affected by occasional events. In order to avoid too much influence of such events on the results, the Committee decided to use multi-year averages as during the first and second benchmarking period.

The result of step one in benchmarking operation is that 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 is still the only appropriate available source of information for this analysis.

For the process evaluation of operations, maintenance and safety management, a peer group of 12 plants was selected which were among the 25% best-performing plants in operations and for which recent OSART reports were available. The Committee used the scoring system developed during the first benchmarking period. A change is that the full OSART reports are in most cases no longer available. The recommendations and suggestions from the OSART reports are still available, but the notes are not. To check whether the methodology was still sound without the use of notes, a check was done on the results of the 2018 report without notes. Although all plants scored lower, the relative change was similar for all plants and the ranking stayed almost the same, with only two peers with close scores swapping places. From this the Committee concluded that without the use of notes, the methodology could still be used.

The results showed that the score of KCB is well within the scores obtained by 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.

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

For the third report the Committee decided to use the same methodology as used for the second report, based on the safety aspects of ageing management for long-term operation as assessed in the IAEA SALTO missions.

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. A study confirmed that the methodology is still up to date and did not need to change.

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 during the current reporting period, are in the top 25% best-performing plants in operations and have a good geographical spread over the benchmark area as far as possible. As KCB didn't have a recent SALTO review, the Committee organized a similar review following the methodology of the SALTO guidelines to obtain the relevant data for the benchmark.

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

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.

In the second report, the Committee focused on the KCB and evaluated whether the siting risks at KCB are assessed in line with international good practices and are properly considered in the design. The Committee concluded that the siting risks at KCB were well investigated in line with modern international good practices and requirements for existing nuclear power plants, and considered the findings of the Fukushima Daiichi accident.

The Committee is of the opinion that revisiting the same evaluation of the siting risks as in the second report is of limited added value and would not result in any new insights.
However, as WENRA (Western European Nuclear Regulators Association) recommended that external hazards should be more systematically reviewed in the periodic safety review of nuclear power plants, the Committee decided to investigate how systematically external hazards are reviewed in the periodic safety review of nuclear power plants in various countries in the EU, USA and Canada.

Conclusion on Siting

The Committee concludes that the way KCB treats siting aspects in the periodic safety review is similar to most plants in the benchmark and better than some. The Committee is confident that siting does 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 behaviour. 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, in all plants visited the operational performance clearly reflects strict adherence to controlled processes and procedures.

Conclusion on Site visits

Based on 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 way safety issues are addressed in the workplace. It often reflects the attitudes, values, beliefs and behaviours 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 them and therefore 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 based on the World Association of Nuclear Operators (WANO) Principles document, *Traits of a Healthy Nuclear Safety Culture, 2013*. The method is based on the assessment of:

- Individual Commitment to Safety
- Management Commitment to Safety
- Management Systems

The Committee noted that at all the visited plants, safety culture receives a lot of attention. However, there is still a large difference in the methodology and ways of implementation.

Conclusion on safety culture

The Committee noted that KCB continues to be 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 other 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.

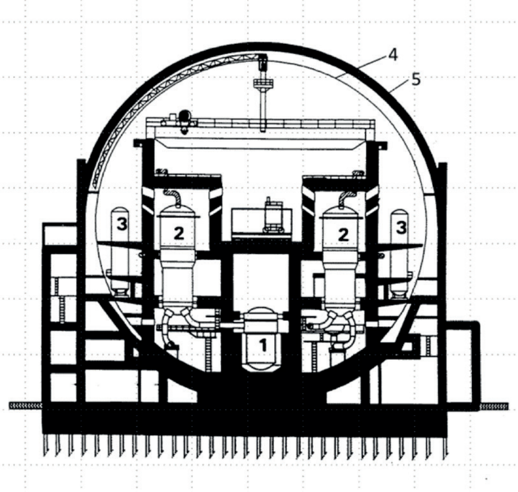
Abbreviations

AFWS	Auxiliary Feed Water System
BWR	Boiling Water Reactor
CS	Core Spray
ECCS	Emergency Core Cooling System
EPZ	Elektriciteits Produktiemaatschappij Zuid-Nederland
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
PSR	Periodic Safety Review
PWR	Pressurised Water Reactor
SALTO	Safety Aspects of Long-Term Operation (IAEA)
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SNF	Spent Nuclear Fuel
WANO	World Association of Nuclear Operators
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.

Figure 1-1 | *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 (see art. 4) 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 and second report were published in 2013 respectively 2018.

[1] Convenant Kerncentrale Borssele, June 2006 (https://zoek.officielebekendmakingen.nl/stcrt-2006-136-p29-SC76083.html)

[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 in European initiatives on nuclear safety. Swiss power plants were therefore included in the benchmark.

This document represents the third report of the Committee. The Committee for the third report was established in 2019.

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 third 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.

Following the Fukushima Daiichi accident, new insights were gained and new requirements regarding the safety of nuclear power plants were developed. The Committee reflected this information and insights in the benchmark methodology of the second report, which led to some extensions and refinements.

During this third reporting period, the Committee made no changes in the methodology.

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). Due to shutdowns, 237 plants were included in the second benchmark. For the third benchmark, the list of plants was again reviewed to include only the reactors that were still in operation or had entered operation as of 31 December 2021 (the cut-off date set by the Committee). The final list of reactors contains a total of 220 reactors.

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

To establish an expert opinion on the safety level of the KCB, as compared with the other 220 water-cooled and water-moderated power reactors in operation in the EU, USA and Canada (as of 31 December 2021), 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.

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

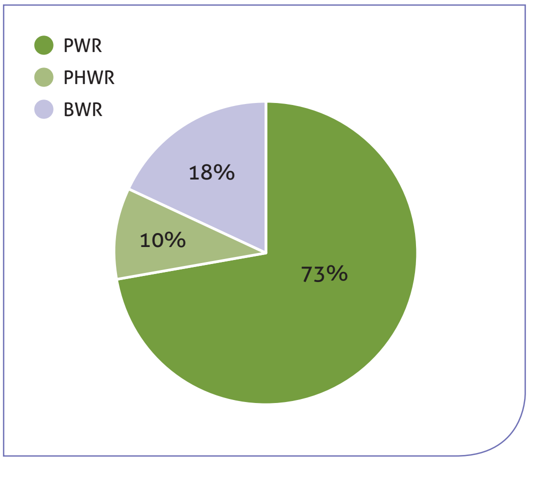
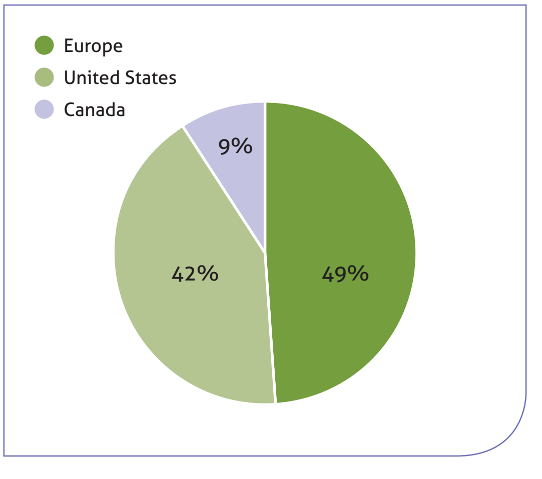


Figure 2-2 | *Geographical distribution of the third benchmark population*



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 fully standardized yet and PSAs have not been conducted for all nuclear power plants. For those plants that do have PSAs, it should be noted that 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 220 peer reactors.

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 the KCB. The methodology used available information on the different elements of reactor safety that could be meaningfully compared among the reactors.

For the second report, the Committee maintained the overall structure of this methodology and improved it to reflect recent developments. In particular the lessons learned from the Fukushima Daiichi accident were incorporated. Second, the Safety Aspects of Long Term Operation (SALTO) review missions developed and carried out by the IAEA were incorporated in the ageing benchmark. Finally, the safety culture benchmark was improved.

For the third report, the Committee made only slight adjustments to the methodology. In particular, the continued worldwide developments in safety culture were reflected in the safety culture benchmark, and the approach was adjusted to more closely fit the newest international insights and safety standards related to the topic. Also, the siting evaluation was adjusted to reflect the newest insights and available information.

In addition, due to restrictions on the availability of OSART reports, the 'notes' used in the methodology for the second report were no longer available to the Committee. In order to determine the effect of the use of the 'notes' on the benchmark results, the Committee applied the modified methodology to the data of the second benchmark. While this resulted in small changes in the ranking of plants that were very close to each other in ranking, it didn't influence the final result. Therefore the Committee concluded that the revised methodology gives comparable and reliable results and can be used for the ranking.

Finally, the Committee looked into the impact of the COVID pandemic on nuclear safety but found no meaningful impact and thus decided to not include this particular event in the benchmark. One particular challenge for the Committee due to the COVID pandemic was the limited number of recent OSART and SALTO mission reports available, as no missions could be carried out during the COVID years.

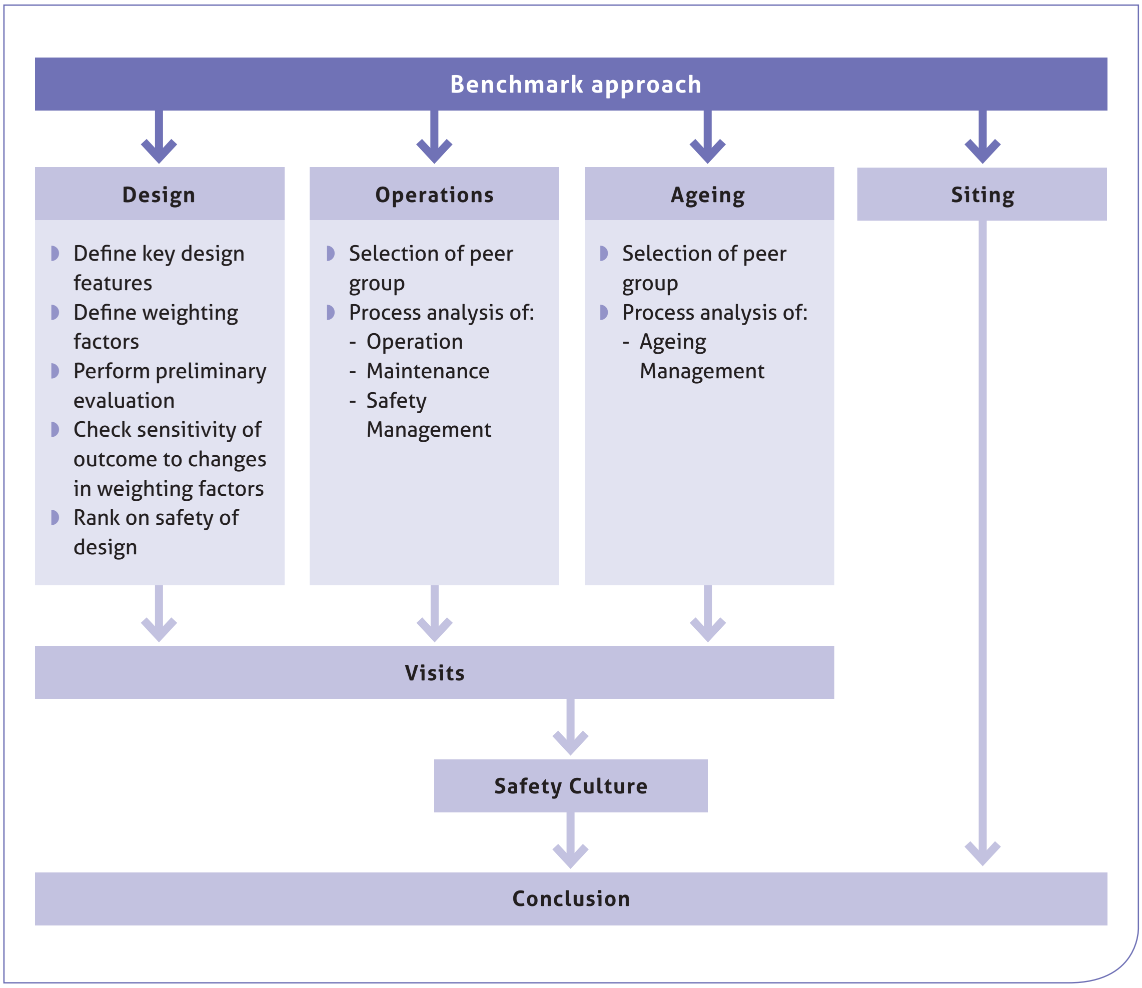
The Committee is convinced that it meaningfully enhanced the previously developed methodology based on all available information in combination with expert assessment. The methodology makes it possible to determine, with enough confidence, whether KCB is among the safest 25% water-cooled and water-moderated 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.

The assessment of the design (chapter 3) was carried out for all 220 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

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



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 was 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 of peer plants.

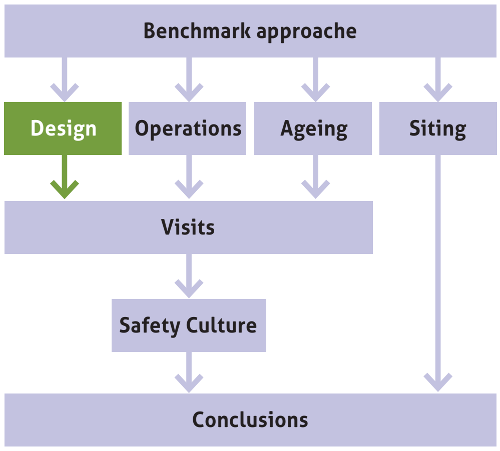
The ageing management benchmark methodology (chapter 5) is structured similarly to the methodology used for the second step of the operation benchmark. In this benchmark, the ageing management programme of KCB was compared in a detailed analysis to that of a properly selected sample of peer plants.

Additionally, in the siting evaluation (chapter 6), the Committee assessed if external hazards are systematically reviewed in the periodic safety review of the KCB according to the state-of-the-art, and if this is on a similar level as other nuclear power plants within the scope of the benchmark.

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 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 to better represent the overall design safety, and to reflect the lessons learned from the Fukushima Daiichi accident, in particular, identified vulnerabilities and safety enhancements proposed. Moreover, based on probabilistic safety analysis, the relative importance of the different safety features was 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). For each of the following 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.

For the third benchmark, the Committee again reviewed the methodology to assess whether there are some reasons to update the methodology. The Committee concluded that since the second benchmark there were no major changes in the safety philosophy that would warrant altering the methodology. There are ongoing safety improvements, in particular the completion of the post-Fukushima measures in the EU and USA/Canada, but no major changes in the approach, methodology or requirements that would require changes in the design safety of nuclear power plants, which would in turn impact the methodology to be used in the third design benchmark.

Furthermore, the review of the methodology that was initiated during the third design benchmark confirmed that the existing approach is fit for purpose and allows for the differentiation of design safety from the perspective of the off-site impact.

The main focus of the third design benchmark was the collection and verification of the data on the status of the plants, and then assessing and cataloguing the data to generate the ratings.

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 nuclear power reactors included in this benchmark 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 Miles 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: "defence-in-depth" (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 defence-in-depth is to manage risk by layering diverse defensive strategies, so that if one layer of defence turns out to be inadequate, another layer of defence will detect, compensate, or correct the safety issue 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 a stratified protection against a wide variety of anticipated operational occurrences, design basis accidents and severe accidents. This includes disturbances or initiators (of a sequence) resulting from equipment failures or human actions within the plant as well as from hazards that originate outside the plant.

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

The focus of the Committee's assessment was on safety aspects relevant for preventing the impact outside a plant, i.e. prevention of a radioactive release. Having a series of design features that would assure prevention and if not successful, mitigation of accidents 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 a plant.

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 seeing massive investments in safety enhancements, for the second report the Committee concluded that it needed to redefine the originally proposed key features as well as define new key features. The following set of features was considered in the second report:

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

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 acci- dents 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	-

To be able to better distinguish between plants, sub-features were added to the key features. For every feature and sub-feature, a 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 were performed during the development of the methods and their results appropriately considered. The Benchmark committee believes that this new methodology is more accurate in reflecting safety of nuclear power plants as defined within the Committee mandate (i.e. impact on the surroundings).

The methodology as defined for the second report was adopted for this third report.

3.2.1 Redundancy and Diversity

Redundancy and diversity are the major design features that strongly impact capabilities of a plant to mitigate consequences of events that otherwise might lead to reactors not being cooled and in danger of overheating, which could result in "core damage" and radioactive release (from fuel).

Redundancy refers to the multiplication of critical components or systems with the intention of increasing the reliability of a system e.g. having two, three or even four parallel pumps or trains 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 of a given function under all circumstances, e.g. electric, steam or diesel driven pumps.

Redundancy and diversity principles are deployed in all nuclear power plant designs. With the same goal of mitigating consequences

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

REDUNDANCY AND DIVERSITY		
Core cooling system		
I	PWR	2 x 100% or less ECCS redundancy, no diversity in AFWS
	BWR	No redundancy in HPCI; 2 x 100% or 3 x 50% LPCI; 1 x 100% CS
II	PWR	More than 2 x 100% ECCS redundancy, no diversity in AFWS OR 2 x 100% ECCS redundancy, diversity in AFWS
	BWR	Redundancy, no diversity in HPCI; 4 x 50% or 3 x 100% LPCI; 1 x 100% CS OR No redundancy in HPCI; 4 x 50% or 3 x 100% LPCI; 2 x 100% CS
III	PWR	More than 2 x 100% ECCS, diversity in AFWS
	BWR	Redundancy and diversity in HPCI; 4 x 50% or 3 x 100% LPCI; 2 x 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*		

of events, practical solutions deployed 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 practice, actual design solutions regarding redundancy and diversity differ between PWRs/PHWRs and BWRs, reflecting different concepts of each of those.

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 matrix (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 and that the degrees of redundancy and diversity could be defined differently.

3.2.2 Containment

The confinement of radioactive material in a nuclear power reactor, including the control of discharges i.e. minimization of external 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 a nuclear reactor – is a major factor in achieving the objectives of the third and fourth levels of defence-in-depth. The containment structure also serves as protection of the reactor against external hazards.

In the Committee's design benchmarking methodology in 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, for the second report 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 exvessel retention of molten core, external reactor vessel cooling and containment filtered venting. With this approach, the containment and its additional features were considered as one key feature, which allows for safety considerations that go beyond the original design basis. Table 3-3 shows the containment function, including its sub-features.

3.2.3 Bunkered systems

Hazards of internal or external origin, such as explosions, fires, flooding, earthquakes, or malevolent acts, all have the potential to initiate a sequence of 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 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 other deficiencies (e.g. inadequate spatial separation, one of the most important protective features for internal and external hazards).

Initially, bunkered systems were meant to be 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 showed that additional hazards needed to be considered.

Table 3-3 | *Definition for 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

Manmade hazards, such as external explosions caused by nearby industrial facilities or aircraft crashes were also considered in some designs. After 2001 terrorist attacks in the USA, other hazards of human origin were considered, e.g. crashes of big commercial aircraft. 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 offered resistance 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 e.g. the "hardened safety core" (HSC), 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 offering additional level of protection. 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 for 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 actions by 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 (SAMGs) to be used by operators. SAM encompasses both the equipment and the actions taken by the plant operating staff during a severe accident, to support:

- 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.

In the late nineties individual plants started to introduce SAMGs and/or some dedicated components to manage severe accidents. After the Fukushima Daiichi accident (Like most Japan's reactors Fukushima units did not have SAMGs), the attitude towards severe accident management changed significantly, and now practically all plants have SAMGs or are in the process of introducing them.

Based on the post-Fukushima safety consideration the key design feature "SAM" was defined

Table 3-5 | *Definition for 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	

to reflect the increased focus on severe accidents as well as the wide-scale deployment of mobile equipment for SAM. Also, the availability of dedicated or qualified instrumentation and control (I&C) for severe accidents was added to SAM category. 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 for the SAM category.

3.2.5 Design of spent fuel pool

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

In some plants, the spent fuel pool is located outside the containment, making it vulnerable to external hazards (e.g. aircraft crash, earthquake). Spent fuel pools that are located within the containment are better protected; should SNF 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 for 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

Table 3-7 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.

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

REDUNDANCY AND DIVERSITY			
Core cooling system			
I	PWR	2 x 100% or less ECCS redundancy, no diversity in AFWS	**1**
	BWR	No redundancy in HPCI; 2 x 100% or 3 x 50% LPCI; 1 x 100% CS	
II	PWR	More than 2 x 100% ECCS redundancy, no diversity in AFWS OR 2 x 100% ECCS redundancy, diversity in AFWS	**3**
	BWR	Redundancy, no diversity in HPCI; 4 x 50% or 3 x 100% LPCI; 1 x 100% CS OR No redundancy in HPCI; 4 x 50% or 3 x 100% LPCI; 2 x 100% CS	
III	PWR	More than 2 x 100% ECCS, diversity in AFWS	**4**
	BWR	Redundancy and diversity in HPCI; 4 x 50% or 3 x 100% LPCI; 2 x 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 with-standing 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**

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

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

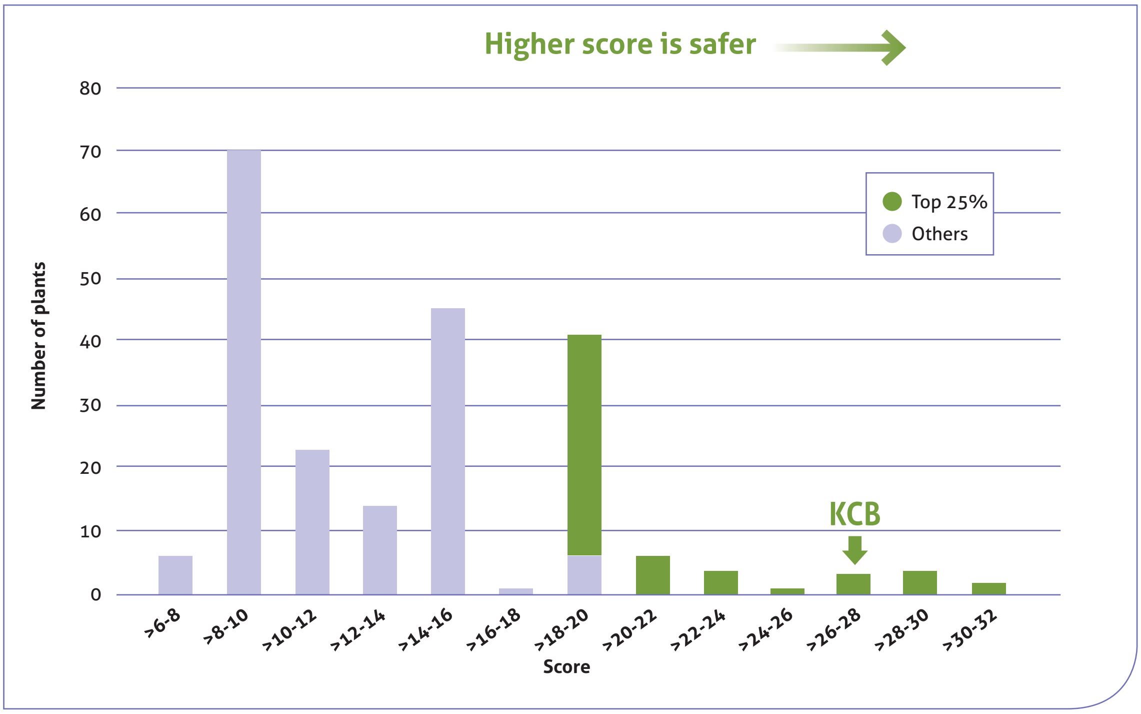
SPENT FUEL POOL		
I	Spent fuel pool located outside the containment	0
II	Spent fuel pool located inside the containment	2

3.3 Results and conclusions

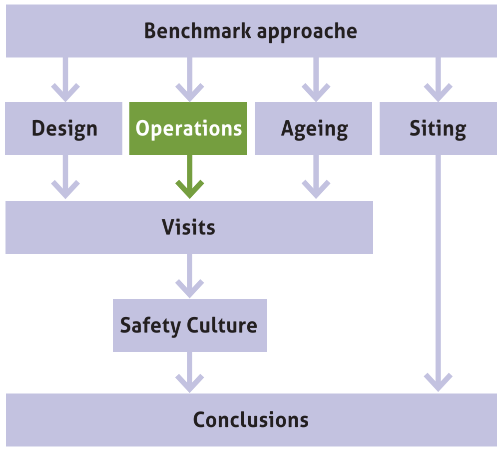
Figure 3-1 provides a graphical representation of the distribution of reactors among the score ranges. Scores assigned to the reactors cover the range of 6.5-30.5 in steps of 0.25. The wide score distribution shows that the benchmarking methodology, including post-Fukushima safety considerations, discriminates well the 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 9% of the reactors score above 20. With a score of 27.25 KCB ranks well within the top 25%.

Figure 3-1 | 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 and subsequently used in the second reporting period up to 2018. 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 a 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 220 the Committee had to consider. 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.

With a score of 88.6 KCB is well within the top 25% best-performing reactors based on performance indicators.

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.

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

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 plants. Plants 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 plants 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. In the past, the OSART report was customarily derestricted ninety days after its issuance, unless the host country requests otherwise. Lately, and that was an issue for the Committee in the 2023 evaluation, the OSART reports are in most cases not made available, even for the regulators in the IAEA member states. What is available are the recommendations and the suggestions that the OSART team made in its report. As explained below, that was enough for the Committee to undertake the operational safety assessment.

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 12 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 and repeated in the second 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 12 peer plants 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 13 plants were considered by the Committee:

- Management, organization and administration;
- Training and qualification;
- Operations;
- Maintenance;
- Technical support;
- Radiation protection;
- Chemistry;
- Operating experience;
- 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;
- 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 (recommendations and suggestions) was objectively categorized, in the same way as for the first and the second BBC report in 2013 and in 2018, 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.

In this area, there was a slight difference in the operational safety evaluation within the BBC benchmarking in 2023 as compared to 2013 and 2018. Namely, the OSART reports are no longer available to the regulatory bodies of the IAEA member states. The information from OSART missions that remain available are the recommendations and suggestions, but not the notes, that were in the past extracted from the textual part of the OSART reports. Therefore, the BBC concluded that there is the need to undertake the operational safety evaluation without considering the notes, for all of the plants in the peer group. To assure that there is no distortion of the assessment, the BBC undertook a sensitivity analysis by excluding the notes for the evaluation of the 2018 report and observing the outcome. The results were that all plants scored lower, but that was relatively proportionally to their previous scores, factually not affecting the outcome. Out of 10 peer plants only two swapped places, and the remainder stayed in the same position as in the initial evaluation. Borssele's position among the peers stayed exactly the same as before. Considering the outcome of this analysis the BBC decided that the methodology may remain as it is even with no notes being available.

The Committee also considered that from the point of view of their potential impact, the issues for which recommendations and suggestions were made can vary in significance.

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.

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
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
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
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
Group V Insignificant/out of scope issues	**0**				

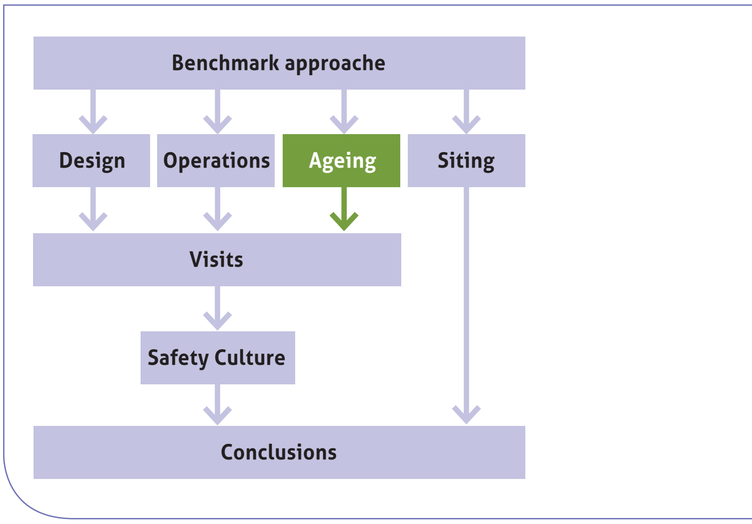
4.4 Results and conclusions

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 4.2 (highest operational safety), to 19.35 (lowest operational safety). KCB is situated in the middle of the range, with seven plants of the peer group being better and five being worse than KCB. 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 keep the ageing degradation and wear of systems, structures and components within acceptable limits.

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 used the same method as for the second report, based on the relevant IAEA safety standards. These standards were slightly modified in the last years, but the Committee came to the conclusion that revision of the method was not needed.

The methodology considered safety aspects of ageing management for long-term operation assessed in IAEA SALTO missions. 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, 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 have a high score on operational performance, preferably ranking in the top 25% based on performance indicators (see Chapter 3).
- Plants should be in or in preparation for LTO.
- The peer group should include different types of reactors.
- The peer group should include plants geographically spread over the benchmark area: European Union, USA, and Canada.
- Reports on IAEA SALTO missions should be available.

To the extent possible, peer plants should have had a relatively recent SALTO review. Because KCB did not have a recent SALTO review, the Committee decided to organise a similar review following the methodology of the IAEA OSART guidelines. The IAEA initiated the SALTO mission programme fifteen years ago. Although more SALTO missions are performed, due to the COVID epidemic no missions were performed for several years, limiting the number of available recent SALTO reports. In the Committee's view, plants having undergone a SALTO mission indicates that ageing management has a relatively high priority at the plant, which is the reason to prefer such plants in the peer group. The final selection of five peers, besides KCB, was based on the expert opinion of the Committee. All selected plants score within the top 25% best-performing plants in the operational safety benchmark (see section 4.2).

5.3 Methodology

The methodology to benchmark ageing aspects was the same as in the second report. The methodology considered the safety aspects of ageing management for LTO that were assessed in the IAEA SALTO peer review service. The ageing management benchmark methodology involved consideration from two points of view: the on-site evaluation of the SALTO team during the mission as reported, 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. 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. Three levels of safety significance were considered: High (100% score), Medium (80%) and Low (60%).

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

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.

The ageing management review methodology was tested in the second reporting period 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 ageing management review methodology.

5.4 Results and conclusions

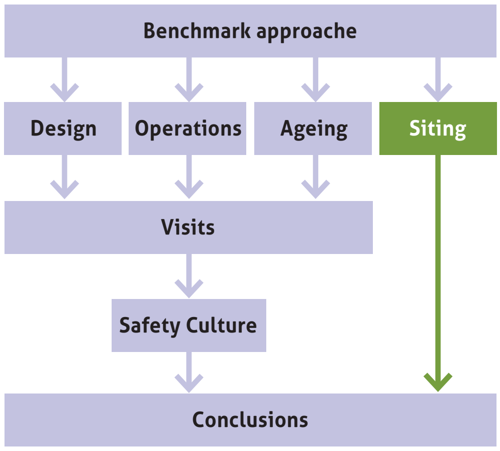
The methodology discussed in 5.3 was used to analyse the six plants in the ageing management peer group. Figure 5-1 shows the total score for each plant, with lower scores indicating better ageing management programmes.

The results show that overall KCB was the second 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 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 designed 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.

In the second report, the Committee focused on KCB and evaluated whether the siting risks at KCB are assessed in line with good international practices and considered in the design, and whether these external hazards pose a risk to KCB. The Committee concluded that the siting risks at KCB were well investigated in line with modern international good practices and requirements for existing nuclear power plants, and considered the findings of the Fukushima Daiichi accident. As such, the Committee was confident that siting did not negatively impact the overall safety ranking of KCB.

6.2 Methodology

The Committee is of the opinion that revisiting the same evaluation of the siting risks as in the second report is of limited added value and would not result into any new insights. However, after the stress test following the Fukushima Daiichi accident, WENRA (Western European Nuclear Regulators Association) recommended that external hazards should be more systematically reviewed in the periodic safety review of nuclear power plants (*WENRA Position paper on Periodic Safety Reviews (PSRs) taking into account the lessons learnt from the TEPCO Fukushima Dai-ichi NPP accident*). These periodic safety reviews are performed at least every ten years by all European and Canadian nuclear power plants and are meant to identify possible safety gaps or points for improvements. A similar periodic analysis (Regulatory Compliance Program) is in place in the USA.

In this context, the Committee decided to investigate how systematically external hazards are reviewed in the periodic safety review of nuclear power plants in various countries in the EU, USA and Canada. The goal was to assess if KCB treats external hazards in the periodic safety evaluation according to the state-of-the-art, and if this is on a similar level as nuclear power plants in other countries within the scope of the benchmark.

6.3 Evaluation of KCB

For KCB, siting aspects are within the scope of the periodic safety review and are fully covered in a systematic way. This includes an evaluation of all external hazards and the level of protection of the safety relevant structures, systems and components against these hazards. Potential future developments (e.g. sea-level rise caused by climate change) are included. The review includes an evaluation if the knowledge on the external hazards is complete, up-to-date and according to the current state of technology, as well as an evaluation of the adequacy of the protection of the plant against these hazards now and in the foreseen future, identifying any possible points of improvement.

In the periodic safety review, KCB checks whether the key environmental characteristics of the KCB site are up to date. It also checks whether the set of external threats considered in the design basis assessment framework is complete in relation to the applicable national and international regulations. Foreseeable developments with regard to the analytical methods of external threats are assessed, including the analysis of the risks arising from these threats. In addition, the manifested external threats at other nuclear power plants are also examined, to determine if they can be used to improve the design.

6.4 Evaluation of other countries

In the majority of investigated countries, siting aspects are fully covered by the PSR in a systematic way. The review covers all external hazards and the protection of the plant against these hazards, with the goal to identify potential points of improvement. The review includes an evaluation of the completeness of the hazards considered and if they are up-to-date, taking into account current analytical methods, safety standards and the latest knowledge of science and technology.

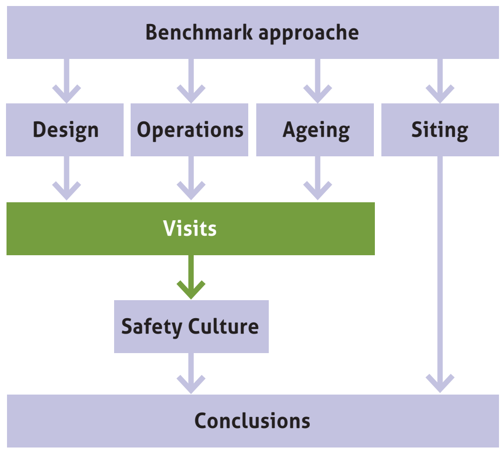
Some countries do not include siting related aspects in the PSR. There is no systematic and complete evaluation of external hazards and the level of protection against these hazards compared to the current state of technology, to identify possible knowledge gaps or points of improvements of the protection against external hazards. Only if there is a specific reason to reconsider a specific siting aspect or there is a concern about the plant's capability to withstand a specific hazard, is that specific hazard included in the PSR. Alternatively, in case of specific events (e.g. the Fukushima accident) or a significant change of the license (e.g. life-time-extension) would these aspects be reevaluated.

6.5 Results and Conclusions

The Committee concludes that KCB treats siting aspects in their periodic safety review in line with modern international good practices. The Committee concludes that the way KCB treats siting aspects in the periodic safety review is similar to most plants in the benchmark and better than some. 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, besides KCB, were visited.

The site visits were carried out after finalizing the desktop analyses.

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
  - Systems, Structures and Components - specific Ageing Management Programs
  - Ageing Management Programme scope for Long-Term Operation
  - Validity of Time Limited Ageing Analyses for the planned period of Long-Term Operation

- **Safety Culture (see chapter 8)**
  - Individual Commitment to Safety
  - Management Commitment to Safety
  - Management Systems

- **Stress test, Post-Fukushima modifications and other upgrades.**
  - Main results of the Post-Fukushima stress test and resulting modifications
  - Other safety related major upgrades

During the plant tour the Committee experts aimed at obtaining an impression regarding issues such as:

- Main Control Room operations and the status of the Reserve/Emergency Control Room.
- Material conditions and housekeeping.
- Maintenance working places (maintenance shops as alternative).
- Specific areas to observe the equipment dedicated to accident management.
- Conditions of safety related systems, in particular 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

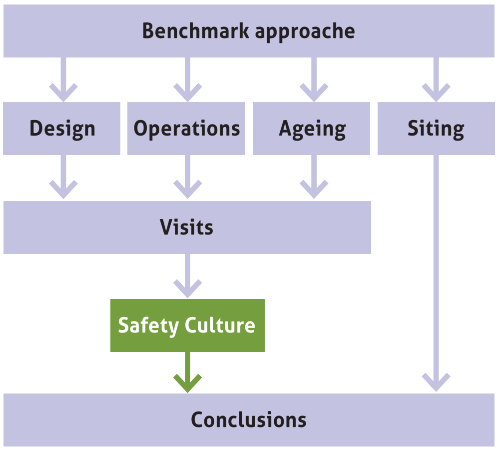
From the overall result of the site visits, the Committee concluded that their impressions were in line with the results from the desktop reviews and that KCB is in line with international best practices and requirements in terms of the items examined.

Below are 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 a continued increase in 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. However, as in the second report, some differences were noticed 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 kept up to date with the current state of the plant and how operators were trained.

8 Evaluation of Safety Culture



8.1 Introduction

As part of our site visits the Committee focused on the programs in place to create a healthy nuclear safety culture. 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 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, developed a custom tool based on eleven indicators of safety culture quality and gave Safety Culture a prominent place during the plant visits.

For this report, the Committee developed a questionnaire based on the World Association of Nuclear Operators (WANO) Principles document, *Traits of a Healthy Nuclear Safety Culture, 2013.* WANO uses this document during its peer review visits to evaluate each site, so it provides a common set of criteria for the benchmark. The questionnaire (see Table 8.1) was provided to the sites prior to the visit to focus the discussion.

8.2 Methodology

It is recognised that safety culture is not all or nothing, instead it is constantly moving along a continuum. The WANO Principles document defines nuclear safety culture as the values and behaviours resulting from a collective commitment by leaders and individuals to emphasise safety over competing goals, to ensure protection of people and the environment.

Experience has shown that the personal and organisational traits described in the Principles document are present in a positive safety culture. Conversely, shortfalls in these traits and attributes are a significant contributor to plant events.

Each site was requested to address the traits described in the WANO Principles document during their presentation, including any areas they are working on to improve the nuclear safety culture at their facility. The Committee evaluated the responses and made observations during the presentation and the plant tour of the effectiveness of each plant's programs to foster a healthy safety culture.

The traits are divided into three categories. The categories and their primary traits are as follows:

- Individual Commitment to Safety
  - Personal Accountability (PA)
  - Questioning Attitude (QA)
  - Safety Communication (CO)
- Management Commitment to Safety
  - Leadership Accountability (LA)
  - Decision-Making (DM)
  - Respectful Work Environment (WE)
- Management Systems
  - Continuous Learning (CL)
  - Problem Identification and Resolution (PI)
  - Environment for Raising Concerns (RC)
  - Work Processes (WP)

8.3 Results and conclusions

Safety culture is a multi-faceted and multi-layered concept. Nevertheless, using the safety culture questionnaire 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 questionnaire, a meaningful comparison among peer plants could be made.

The Committee noted that at all the visited plants, safety culture receives significant attention. However, differences in methodology and ways of implementation of the WANO Principles continue to exist from plant to plant.

The Committee noted that KCB continues to be 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.

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

Individual Commitment to Safety
Is there a vision/mission statement/policy that addresses the responsibility and authority for nuclear safety? (CO)
How is the importance of nuclear safety communicated to all levels of the organisation? (CO)
Do individuals have the authority to carry out their work safely, including stopping work when in doubt about the safety of evolution? (PA & QA)
How do you measure that employees take personal responsibility for nuclear safety? (PA)
How are safety communications incorporated in work activities? (CO)
How do individuals and work groups communicate and coordinate their activities within and across organisational boundaries to ensure nuclear safety is maintained. (CO)

Management Commitment to Safety
How do leaders throughout the organization demonstrate a commitment to nuclear safety? (LA)
When a situation arises that requires a choice between nuclear safety and production how is the decision handled and who decides? (DM)
How is trust fostered among individuals and work groups throughout the organization? (WE)
How are individuals encouraged to voice concerns, provide suggestions and raise questions? (WE) *Is there a process for resolving differing professional opinions? (WE)*

Management Systems
What is the process for the collecting, evaluating and implementing lessons learned from operating experience information? How are lessons learned communicated to the staff? (CL)
Are there regular/recurring nuclear safety initiatives or programmes planned or recently completed as the result of self-assessments or benchmarking? (CL)
How does your corrective action programme work? (PI) *a) How can any employee submit an issue to the programme?* *b) How are the evaluations and resolutions prioritized?* *c) How are issues from safety audits and regulatory inspections addressed?*
Is there a policy that supports individual rights and responsibilities to raise safety concerns and does not tolerate harassment, intimidation, retaliation, or discrimination for doing so? (RC)
Is there a process for an employee to raise a concern that is independent of line management? How are these concerns addressed? (RC)
How does the process of planning, controlling and executing work activities ensure that nuclear safety is the overriding priority and that nuclear safety risk is commensurate with the work to be performed? (WP)