The Koeberg Risk Assessment Report

A Report for PSA Analysis
Number: PSA-R-T19-01
Revision: 10

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CONTENTS

LIST OF FIGURES

LIST OF TABLES

Page

EXECUTIVE SUMMARY

One aim of this report is to assess the off-site radiological risk due to accidental releases of radioactive materials at Koeberg Nuclear Power Plant (KNPP) and to compare those risks to the criteria of the National Nuclear Regulator (NNR) as prescribed by RD-0024 [2].

It integrates the following modifications considered during the replacement of the steam generators:

- Replacement of the steam generators currently installed on the plant by 60/19 T steam generators,
- Abandonment of the ORT temperature programme for the primary coolant,
- Implementation of the reference core approach (flexible and multiple fuel vendors).

The risks from all accidents and their percentages of the NNR criteria are summarised in Table 0-1 below.

Table 0-1: Risks for KNPP (for Internal and External Events)

Criteria	Annual Risk (Station)	% of NNR Criteria
Peak Public Risk (fatalities year^{-1})	1.17E-7	2.35
Average Public Risk (fatalities person^{-1} year^{-1}) Using 2008 national population[1]	1.03E-9	10.32
Average Public Risk (fatalities person^{-1} year^{-1}) Using 2011 national population[2]	9.71E-10	9.71
Average Public Risk (fatalities person^{-1} year^{-1}) Representative of year 2025[3]	2.37E-9	23.75
Peak Site Personnel Risk (fatalities year^{-1})[4]	7.56E-6	15.12
Average Site Personnel Risk (fatalities person^{-1} year^{-1})[4]	4.07E-06	40.66

1. The national population in 2008 was 48 687 000. This was included to use the same population year as per the Level 3 PSA.
2. The national population in 2011 was 51 770 560.
3. 2008 average public risk scaled by 2.3 to be representative of the year 2025 (see Section 9.1.7).
4. Scaling of the peak and average site personnel risks by a factor of 10 are included (see Section 5.6).

Several conservative assumptions were made in the compilation of this assessment, which implies that the final results are also conservative. For example, no emergency plan countermeasures, other than the food ban, have been assumed. The public are therefore assumed to be outdoors at the time of the accident and to remain outdoors for the duration of the exposure period. Operating experience such as the Fukushima Daiichi NPP accident indicates that Emergency Planning can be very effective at reducing public doses.

To ensure that no single accident results in an unacceptably high number of deaths, the NNR have included a bias against larger accidents. This is illustrated in Figure 0-1 below.

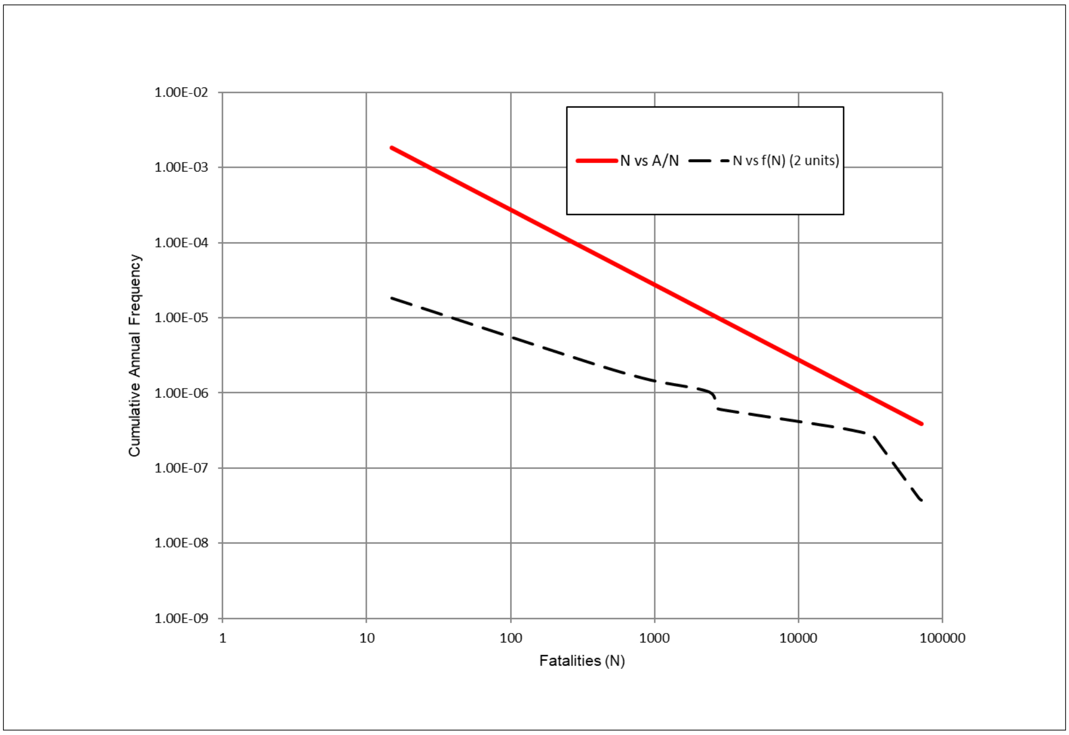

Figure 0-1: Bias Against Large Accidents (For Internal and External Events)

This study therefore indicates that all the criteria set by the National Nuclear Regulator for public risks and site personnel risks have been met. However, as discussed in the Koeberg Licence Basis Manual referenced from NNR NIL-1 [1], ALARA must be applied and so efforts to maintain, or even reduce, this risk must continue.

1. INTRODUCTION

The Koeberg Nuclear Installation License [1] requires Eskom to assess nuclear risk and demonstrate compliance with the safety criteria of the NNR as specified in RD-0024 [2]. Eskom demonstrates compliance with these licensing requirements by maintaining a living Probabilistic Safety Assessment (PSA), the results of which are summarised in this Risk Assessment Report.

The Risk Assessment Report summarises the PSA results and lists the main conclusions and insights drawn from the PSA. This risk assessment reflects the physical plant following the replacement of the steam generators.

Note that Revision 10 of the Risk Assessment Report (RAR) does not invalidate Revision 9 of the RAR. Revision 9 will continue to be applicable to the relevant unit(s) until the steam generators on both units have been replaced.

1.1 KOEBERG SAFETY CASE

Accidents can be assessed from a probabilistic or deterministic perspective. Deterministic safety assessment is generally used for the Design Basis Accident (DBA) analyses contained in the Safety Analysis Report (SAR). The SAR also contains deterministic safety assessments of certain Beyond Design Basis Accidents (BDBAs), such as the chapters dealing with complementary accidents and miscellaneous accidents. These BDBA analyses include, for example, Station Blackout (SBO) and Loss of Ultimate Heat Sink (LUHS). PSA is used for a wider scope of accidents, not only the DBAs and BDBAs defined in the SAR, and the results are summarised in the Risk Assessment Report (RAR).

The RAR was originally part of the SAR when Koeberg NPP was commissioned. Thus, Chapter 17 of the Intermediate Safety Analysis Report (ISAR) summarised the Probabilistic Safety Assessment. However, during the Koeberg Safety Re-Assessment (KSR) project, the RAR was removed and became a stand-alone document. The RAR still forms part of the Koeberg NPP overall Safety Case, demonstrating that the public risks are within acceptable limits.

1.2 PSA APPLICATIONS

The most prominent PSA application, addressed in Chapter 13 of this report, is the demonstration of compliance to the risk criteria in RD-0024 [2]. In addition, the Precursor Analysis application is presented in Chapter 14 of this report as it can influence compliance to the risk criteria in RD-0024 [2]. The Precursor Analysis is also presented to Koeberg management twice annually to enable any risk insights to be acted upon. An example of this was the Precursor Analysis highlighting the significance of fire doors in the electrical building.

Many other processes at Koeberg NPP are risk informed. Examples include the Safety Evaluation process, (KAA-709), Operating Technical Specifications (OTS), Maintenance Bases, Component Classification (331-94), Work Management (KAA-721) and Severe Accident Management (SAM). Discussion of these risk informed processes, however, falls outside the scope of this document.

1.3 PEER REVIEWS AND INSPECTIONS

In 2010 a peer review of the Koeberg PSA against the ASME PSA Standards was performed. A follow-up focussed peer review was performed in 2015. The Koeberg PSA was found to substantially comply with the ASME PSA Standards at Capability Category II, and can therefore be used to support risk-informed applications.

The peer review resulted in a number of recommendations for improvements to parts of the Koeberg PSA which were subsequently implemented except in a few select cases where justified. The most significant change was updating the Human Reliability Analyses with specific focus on human errors occurring during maintenance, surveillances and testing prior to the accident initiator.

2. SCOPE

The PSA considers all major accidents involving nuclear fuel and the possible release of radionuclides. This includes accidents in the reactor vessel, spent fuel pool and spent fuel storage casks, as well as accidents during fuel handling in the reactor and fuel buildings.

The overall risk to site personnel and the public, due to the release of radionuclides, is quantified in the PSA. These risks are compared to the regulatory limits imposed by the NNR in RD-0024 [2].

The change in nuclear risk over time is tracked in the Risk Profile. This risk may either decrease (*e.g.* due to safety-related modifications) or increase (*e.g.* due to emergent plant issues or on-line maintenance).

Changes to the plant's nuclear risk are typically compared to the internally-imposed Eskom risk limits defined in reference [3]. Where regulatory guidance is absent, Koeberg is benchmarked against the risk limits such as those from the IAEA.

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4. DEFINITIONS AND ABBREVIATIONS

4.1 DEFINITIONS

4.1.1 Core Damage Frequency: Probability of having a core damage event within a certain time frame, typically per year.

4.1.2 Severe Accident: A severe accident is defined by plant conditions that are indicative of fuel damage that result in the release of a significant portion of the fission products from the fuel rods.

4.2 ABBREVIATIONS

Abbreviation	Description
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Basis Accident
BOR	Boil-off Rate
CDF	Core Damage Frequency
CGS	Council for GeoScience
CSB	Cask Storage Building
DBA	Design Basis Accident
EdF	Électricité de France
EDG	Emergency Diesel Generator
EERT	External Hazards Review Team
EOP	Emergency Operating Procedure
DevonWay	The Corrective Action Management software
EPRI	Electric Power Research Institute
ASG	Auxiliary Feedwater System
FDF	Fuel Damage Frequency
HEP	Human Error Probability
HHSI	High Head Safety Injection
HRA	Human Reliability Analysis
ICRP	International Commission on Radiological Protection
IEF	Initiating Event Frequency
ISAR	Intermediate Safety Analysis Report
ISLOCA	Interfacing Systems Loss of Coolant Accident
KSR	Koeberg Safety Re-Assessment
LERF	Large Early Release Frequency
LLW	Low Level Waste
LOCA	Loss of Coolant Accident
LOSP	Loss of Off-Site Power
LTOP	Low Temperature Over-Pressure

Abbreviation	Description
LUHS	Loss of Ultimate Heat Sink
MAAP	Modular Accident Analysis Program
MCCI	Molten Core Concrete Interaction
MPC	Multi-Purpose Container
NNR	National Nuclear Regulator
NM	Normal Mode
NPP	Nuclear Power Plant
OM	Outage Mode
OTS	Operating Technical Specifications
ORT	Operating at Reduced Temperature
PAZ	Precautionary Action Zone
PGA	Peak Ground Acceleration
PDS	Plant Damage State
PORV	Power Operated Relief Valves
POS	Plant Operating State
PSA	Probabilistic Safety Assessment
PSV	Pressuriser Safety Valve
PTR	Reactor Cavity and Spent Fuel Pit Cooling System
PZR	Pressuriser
RC	Release Category
RCP	Reactor Coolant System
RRA	Residual Heat Removal System
RRI	Component Cooling System
SAM	Severe Accident Management
SAMG	Severe Accident Management Guideline
SAR	Safety Analysis Report
SBO	Station Blackout
SF	Spent Fuel
SFC	Spent Fuel Cask
SFD	Spent Fuel Damage
SFP	Spent Fuel Pool
SG	Steam Generator
SGR	Steam Generator Replacement
SGTR	Steam Generator Tube Rupture
SMA	Seismic Margin Assessment
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
TISF	Temporary Interim Storage Facility
TSC	Technical Support Centre

Abbreviation	Description
TPU	Thermal Power Uprate
UPZ	Urgent Protection Zone

5. MAIN CHANGES INCLUDED INTO THE CURRENT REVISION

The main changes made from the previous version of the Koeberg PSA are listed below.

5.1 KOEBERG SPECIFIC DATA

The last major Koeberg-specific component reliability data review was performed in February 2013 [41]. Since then, 6-monthly reviews of component failures recorded on DevonWay are done. The most recent 6 monthly review was performed in December 2021.

Independently of this, AC power-related component failures are reviewed more frequently due to their dominant impact on the PSA results. Thus, an update using observed data of the failure rates related to the 400 kV and 132 kV off-site power supplies and the Emergency Diesel Generators are recorded in separate PSA Notebooks.

The overall PSA results were not meaningfully impacted by these failure-rate updates, which essentially imply that the observed failures and unavailabilities are within their expected ranges.

5.2 LEVEL 1 PSA

The purpose of the Level 1 PSA is to determine the Core Damage Frequency after the replacement of the steam generators. The following changes were introduced in the SGR Level 1 PSA model update, as detailed in [4]:

- Change in PORV/PSV PZR LOCA success criteria to credit an early cooldown and depressurisation using ES-1.2, within 35 minutes of the accident initiator, in case of HHSI failure;
- Need for controlling pressure with pressurizer sprays, PORVs or PSVs opening in case of initiating events leading to a decrease in heat removal;
- Change in ASG success criteria to credit one turbo-pump to two SGs instead of one turbo-pump to one SG;
- Change in Loss of RRA success criteria to require manual opening of all three LTOP PORVs in case of RRA-RCP suction line not isolated;
- Change in some HEP values following evolutions in time available to perform actions. The HEPs that were changed are documented in [50] and [51];
- Change in recovery times of LHx switchboards;
- Increase in loss of RRI flooding initiating event frequency. According to the frequency and flood rate analysis provided in Table 3 of [44], the current RRI pump rooms flooding initiating event frequency of (9.44E-02/y)×2= 1.89E-01/r/y is revised to reflect the recalculated frequency of (1.32E-01/y)×2= 2.64E-01/r/y.

5.3 SPENT FUEL POOL PSA

Spent fuel pool accidents remain a significant contributor to the overall estimated public risks, although the model conservatively discounts the early phase Emergency Plan actions such as evacuation which could be very effective given the substantial time generally available.

PSA18-0044 [42] compared the source terms for the pre- and post-SGR cases and concluded that "continued use of the pre-SGR SFP source term as an input parameter and reference in the SFP PSA for the post-SGR without TPU is justified" [42].

5.4 LEVEL 2 PSA

The thermal hydraulic studies supporting the assumptions related to severe accident phenomenon and the human reliability assessments have been revised with the SGR configuration. The SG modelling has been adapted to reflect the new design and the residual heat and the source term have been revised according to the reference core approach [32].

As shown in reference [11], the SGR modification does not induce any changes to the Koeberg containment split fractions. However, the following modelling improvements have been implemented during the SGR project even though they are not directly related to the replacement SGs:

- Introduction of a release category RC-6,
- Update of split fractions to consider KNPS specific fragility curve,
- Change in early hydrogen burn, DCH and vessel rocketing split fractions,
- Update of error factors.

To determine the impact of these modelling improvements, a new pre-SGR Level 2 baseline was developed that takes into account only these changes. The Level 2 PSA was subsequently recomputed to take into account Level 1 changes (due to SGR) and the improvements in the Level 2 PSA baseline itself. All changes provided to the RiskSpectrum model are documented in reference [33].

5.5 LEVEL 3 PSA

The thermal hydraulic studies used to assess the Release Categories, the energy of release and release fractions have been revised with the SGR configuration. The Level 3 calculations were adapted to reflect the new design, the residual heat and the new source term based on the reference core approach. Moreover, the extraction methodology of energy of release from MAAP calculations has been revised [32].

Some MAAP parameters are used to evaluate the energy of release contributing to plume rise. The PC Cosyma manual states that 'energy of release' in the code refers to the heat release contributing to plume rise. Since decay heat does not contribute to plume rise, a different methodology is applied, taking into account the thermodynamic energy of the release, for most cases:

$WRBj \times HGRBc$, where

WRBj is the flow rate of the release through junction j,

HGRBc is the enthalpy of the fluid in the upstream compartment of the release.

Such a methodology gives a more correct evaluation of the energy contributing to the plume rise at the time of the release, and would therefore improve the accuracy of the Level 3 PSA results.

For cases in which the containment does not fail but is bypassed, such as SGTR or ISLOCA cases, the formula given above does not apply as the released fluid is not released through a junction. Consequently, the following formulas have been applied instead:

For SGTR cases: $WGBST \times HGSB$, where

WGBST is the flow rate out of the broken SG,

HGSB is the gas enthalpy of the broken SG.

For ISLOCA cases: $WGBB \times HGBB$, where

WGBB is the gas flow rate out of the broken loop break,

HGBB is the gas enthalpy of the broken loop flow.

The Level 3 PSA has thus been recomputed to take into account Level 2 changes and the changes provided to the characterization of release categories (because of SGR impact and because of change in methodology as explained above). All changes provided to the PC Cosyma model are documented in [12].

Additionally, the following way forward proposed in Section 5 of the report titled: "Impact Evaluation of Outdated Population Data Used in PSA Studies", PSA-R-T16-22, Revision 4 [49], is implemented as an interim arrangement for all PSA applications and projects while Section 5.4 of the Duynefontein Site Safety Report is being updated to consider the Census 2011 population data:

- For existing Koeberg Level 3 studies where the average public risk results have already been calculated using the total national population for Census 2011 value of 51770560, a scaling factor value of 1.0633 should be applied to the calculated average public risk results (i.e., multiply the average public risk results by 1.0633) to adjust the average public risk results to be representative of the year 2008.
- A further scaling factor value of 2.3 determined for the year 2025 should be applied to the calculated average public risk results (i.e., multiply the calculated average public risk results by 2.3) to adjust the average public risk results to be representative of the year 2025.

Lastly, a scaling factor of 10 to the peak and average risks to site personnel from accidents related to releases from core damage, the spent fuel pool and casks, as proposed in PSA-R-T18-03 [47] were applied where applicable, to compensate for the limitations of PC Cosyma when used at near-field distances.

The results of the quantification are detailed in reference [12].

5.6 SITE PERSONNEL PSA

The fuel handling accidents initiating event frequencies were revised to integrate the change in fuel management strategy following SGR [13].

A bounding source term applicable for SGR with TPU and SGR without TPU configurations has been used to update the fuel handling accident risk [13].

It should be noted that in the Pre-SGR baseline PSA, the whole body effective and organ doses due to inhalation were calculated using the LUDEP computer code. As the LUDEP code was obsolete, in the frame of SGR and for this analysis, the whole body effective and organ doses due to inhalation were calculated using ICRP Publication 68 dose conversion factors from Radiological Toolbox Version 3.0.0 [43]. This method gives similar results to the LUDEP calculations previously used to perform these calculations.

A scaling factor of 10 was applied to the peak and average site personnel risks arising from accidents related to releases from core damage, the spent fuel pool and casks, as proposed in PSA-R-T18-03 [47]. The purpose of this scaling factor is to compensate for the limitations of PC Cosyma when used at near-field distances.

5.7 SPENT FUEL CASK PSA

The scope of the Spent Fuel Cask (SFC) PSA is to assess the risk to site personnel and the public associated with accidents involving spent fuel assembly metal casks [29]. Such accidents could occur during the cask loading, transfer and storage phase at the Koeberg site. To the extent practical, the SFC risk assessment is based on guidelines from recent studies performed by EPRI as detailed in Section 10

Accidents occurring during the cask loading phase have been shown to be a significant contributor to the overall station risk and therefore cask rigging operations are critical to nuclear safety and need to be appropriately controlled. Accidents occurring during the storage phase are not a significant contributor to the overall station risks.

5.8 EXTERNAL HAZARD PSA

The following external events were considered in the analysis of the external events PSA:

- Seismic events
- Aircraft Crash
- Jellyfish Events
- Oil Spill Events
- SHY Plant

Internal Fire and Internal Flood were previously indicated as external events and have now been included in the calculation of Internal Events as per the ASME PSA Standards.

6. THE KOEBERG LEVEL 1 PSA

6.1 INTRODUCTION

The Level 1 PSA quantifies the likelihood of having a core damage event in the reactor. It covers all plant states where nuclear fuel is present in the vessel.

The Koeberg PSA model is a large and complex model, whereby operating principles, maintenance regimes, thermal hydraulic, electrical, mechanical, procedural and accident phenomena are modelled to provide a best estimate of core damage frequency.

The quantification of the Level 1 PSA for the configuration when the steam generators have been replaced has been documented in [4]. Note that the Level 1 PSA was developed considering SGR with TPU configuration as it is bounding for SGR without TPU configuration.

In this assessment, all applicable requirements of NNR RD-0024 [2] were met. Where RD-0024 does not give instruction, ASME PSA standards were used as guides in the development of the PSA with the intent that the PSA should meet “Capability Category II” in areas impacting risk informed decisions. Applicable ASME standards are:

- ASME/ANS RA-Sa-2009, Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications [5],
- ANSI/ANS-58.22 (2009), Low Power and Shutdown PRA Methodology (Draft) [6].

6.1.1 New Model Changes

The following changes were introduced in the SGR Level 1 PSA model update, as detailed in [4]:

- Change in PORV/PSV PZR LOCA success criteria to credit an early cooldown and depressurisation using ES-1.2, within 35 minutes of the accident initiator, in case of HHSI failure;
- Need for controlling pressure with pressurizer sprays, PORVs or PSV opening in case of initiating event leading to a decrease in heat removal;
- Change in ASG success criteria to credit one turbo-pump to two SGs instead of one turbo pump to one SG;
- Change in Loss of RRA success criteria to require manual opening of all three LTOP PORVs in case of RRA-RCP suction line not isolated;
- Change in some HEP values following evolutions in time available to perform actions. The HEPs changes are described in references [50] and [51];
- Change in recovery times of LHx switchboards;
- Increase in loss of RRI flooding initiating event frequency. According to the frequency and flood rate analysis provided in Table 3 of [44], the current RRI pump rooms flooding initiating event frequency of (9.44E-02/y)×2= 1.89E-01/r/y is revised to reflect the recalculated frequency of (1.32E-01/y)×2= 2.64E-01/r/y.

6.1.2 Methodology

The Level 1 PSA develops success criteria to model the accidents sequence mitigation. It also leads to performing human reliability assessment to model the operator actions in the accident sequence mitigation.

In the baseline PSA, these were mainly performed using MAAP studies. The main baseline MAAP studies [32] were recomputed to revise both success criteria and human reliability analyses.

Note: The modification consisting of the replacement of Steam Generators was performed following all contractual quality standards, ensuring a high quality at design, manufacturing, and installation stages. In addition, the use of Alloy 690 for the SG tubes should be beneficial with regards to the frequency of the SGTR event. Despite this, SGTR, LOCA Initiating Events frequency which are currently defined using NUREG/CR-5750 [46] have not been modified to maintain conservatism and also to align with international references.

6.2 RESULTS

As documented in [4], the total core damage frequency, after combining internal initiators, internal flood, aircraft crash and fire initiators, is 6.78E-06 per reactor per year. These initiators are the scope of the formal Koeberg PSA Level 1 model.

The most dominant contributors to the core damage frequency are listed below. These dominant initiating event groups contribute 89% (6.04E-06/r.y) to the overall core damage frequency.

- Internal fire with 34.4% (2.34E-06/r.y).

  Redacted information provides detailed information on the plant vulnerabilities, key equipment and their locations which can be exploited by persons with malicious intent. NNR act.

- LOCA event group with 26.9% (1.82E-06/r.y), dominated by the following initiating events
  - Draindown LOCA contributes 7.5% (5.12E-07/r.y)

    Redacted information provides detailed information on the plant vulnerabilities, key equipment and their locations which can be exploited by persons with malicious intent. NNR act.
  - Small LOCA (PRZ, RCP) contributes 11.5% (7.80E-07/r.y). Specifically the sequences with failure of the HHSI automatic injection (signal elaboration and spurious SI stop) dominate.
- Loss of off-site power with 11.5% (7.78E-07/r.y).
- Loss of electrical boards with 8.5% (5.76E-07/r.y).
  - Dominated by the CCF LLHx (loss of both LHx boards event) which contributes 6.2% (4.18E-07/r.y).
- Secondary transients with 7.7% (5.22E-07/r.y)
  - Dominated by the loss of main feedwater event which contributes 2.09% (1.96E-07/r.y).

A more detailed overview of the dominant contributors to CDF is shown in Figure 6-1. Although aircraft crash is classified as an external hazard, it is included here because the PSA assumes that it leads directly to core damage without any mitigation being possible. Chapter 12 of this report provides a more comprehensive discussion on the impact of external hazards on accident modelling in the PSA.

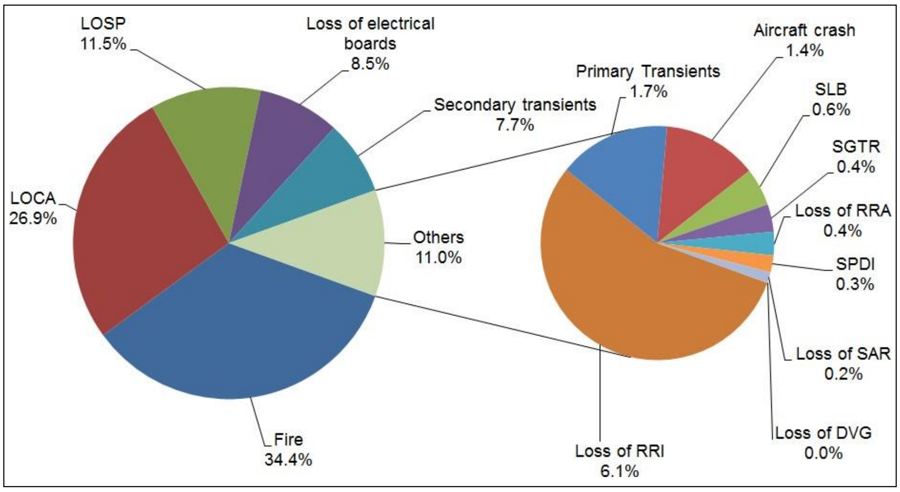

Figure 6-1: Dominant contributors to the Level 1 CDF

6.2.1 Internal Flooding

The Level 1 model includes the estimation of core damage resulting from an internal flooding event. After implementation of plant modifications recommended by the **PSA** (such as moving the PTR Level sensors), internal flooding risk is now dominated by flooding of the RRI pump rooms. The internal flooding report [44] impacting the flooding zone, has been incorporated into the PSA. The core damage frequency from flooding leading to the loss of RRI alone results in 4.14E-07/r.y (6.1%).

Redacted information provides detailed information on the plant vulnerabilities, key equipment and their locations which can be exploited by persons with malicious intent. NNR act.

The most dominant contributor to the core damage > 24 hours results from a single initiator, namely, loss of RRA cooling with 97.9% (5.01E-04/r.y). Mitigations for loss of RRA cooling with cavity full are not modelled, since the recovery time available is generally greater than 100 hours approximately.

6.2.2 Plant Operating States

A penalizing assumption in the PSA model is that the decay heat load is the same when returning to power as when shutting down for refuelling. These heat loads vary considerably. This, together with the conservative assumption to combine the accidents during shutdown together with accidents for start-up, using the worst-case scenario in each case, adds to the high risk contribution from human error events. It is recognized that more time is available for manual mitigation actions during startup.

The contributions of different plant operating states (POS) to the overall CDF are shown in Table 6-1. Also shown are the average time fractions spent in each plant operating state, as determined in [7]. The Level 1 PSA does not include any accidents where the fuel is removed from the reactor building.

Table 6-1: Contribution of POS to CDF

Plant Operating State	Time Fraction Spent in POS	Contribution to CDF	
		%	CDF (yr^{-1})
At power	84.4%	71.6%	4.88E-06
Shutdown with RRA valved-out	4.0%	10.2%	7.02E-07
Shutdown with RRA valved-in	5.8%	18.2%	1.24E-06
Reactor completely defuelled	5.8%	-	-

The main contributors to CDF in each of the plant operating states are shown in Figure 6-2, Figure 6-3 and Figure 6-4.

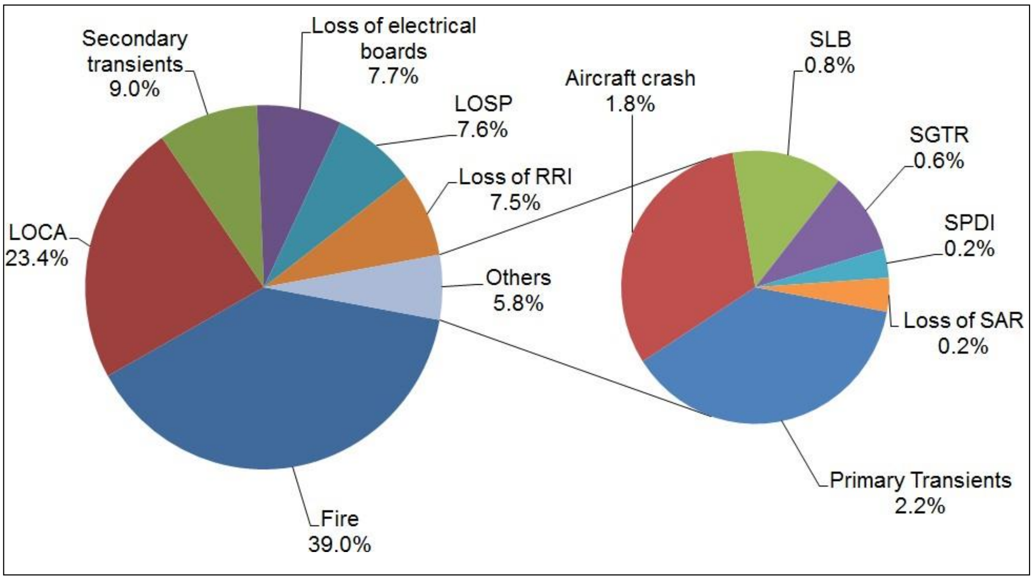

Figure 6-2: Dominant CDF Contributors at Power

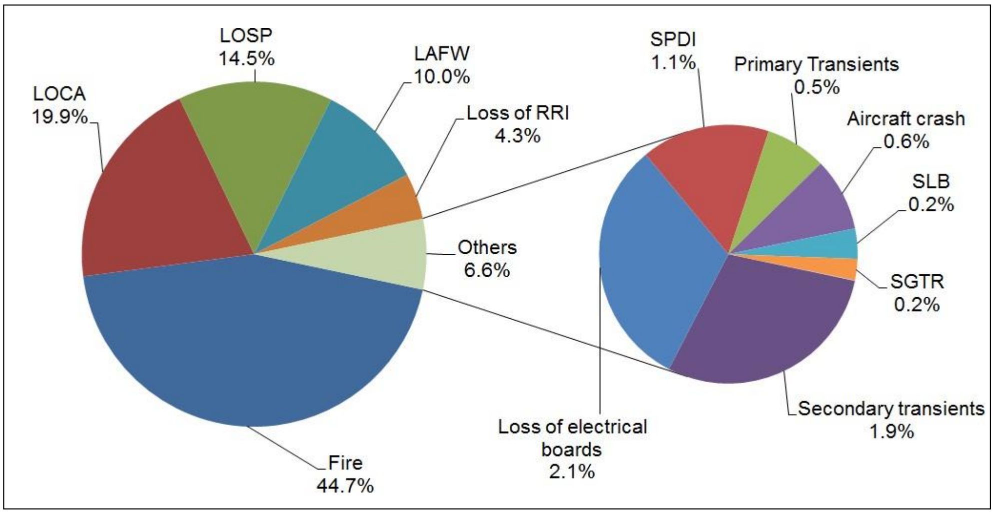

Figure 6-3: Dominant CDF Contributors at Shutdown – RRA Not Connected

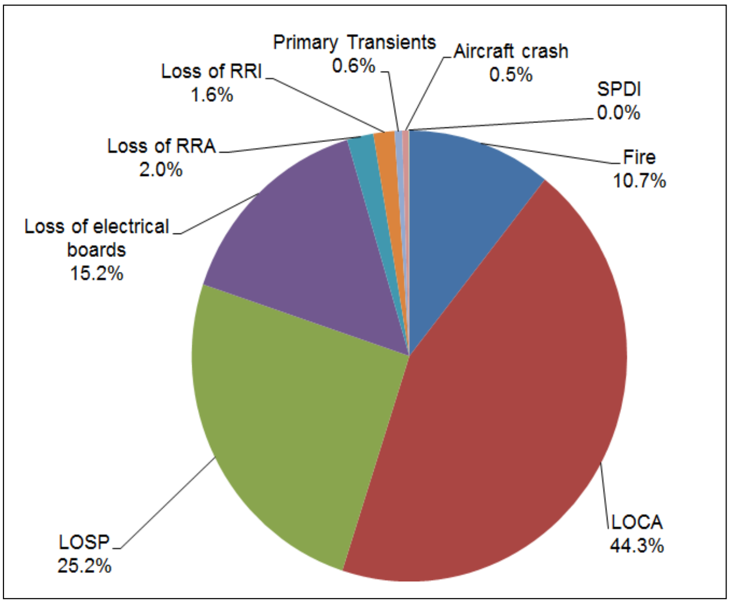

Figure 6-4: Dominant CDF Contributors at Shutdown – RRA Connected

6.2.3 Human Reliability Analysis

The human reliability analysis (HRA) results show that the operator actions listed in Table 6-2 are the top 10 dominant operator action basic events impacting CDF. [4].

Table 6-2: Main Operator Action Contributions to the CDF

ID	Description	FV	RIF
HEP-FB\LX402	Fire Brigade Fails to Suppress Fire in FZ LX 402 Following Automatic Detection	3.28E-01	1.00
HEC-LGX-ECA00-01H	Operator Fails to Swap Over from Unit to Station Supply	1.30E-01	10.1
HEP-MUHSI-ECA02-<2HF	Operator Fails to Perform HHSI Injection (<2 hours)	9.94E-02	5.39
HHD-CDD-ES12-PH2	Operator Fails to Perform ES-12 Cooldown & Depressuration (LHHSI)	7.47E-02	1.07
HEC-SIT-PREMAT	Operator Terminates Prematurely SI	6.13E-02	452
HEB-RRI-FLOOD	Operator Fails to Mitigate Flooding causing loss of RRI (zone 7)	5.91E-02	4.22
HEA-PTR_MN-HJD	PTR 018-020 MN Tank Level Transmitters Miscalibrated (Human Joint Dependency)	4.83E-02	1060
HEA-RCP_MP1-HJD	RCP 005/ 006/ 013-015 MP PRZR Pressure Transmitters Miscalibrated (Human Joint Dependency)	4.04E-02	142
HEC-HSI\D-E0-PH1	Operator Fails to Perform Safety Injection (LOCA\S\AP_ / SD_)	4.02E-02	11.5
HEC-CH\R-IRRA3-03H	Operator Fails to Increase Charging	3.94E-02	43.3

6.3 INSIGHTS AND CONCLUSIONS

The importance of the "At Power" POS is quite evident. The plant is expected to be in this operating domain on average for 84.4% of the year and the contribution to CDF is 71.6% (4.86E-06/r.y). In addition to fire in zone (39%), the dominant "At Power" risk is due to PZR LOCA with 6.67E-07/r.y (13.7%). Redacted information provides detailed information on the plant vulnerabilities, key equipment and their locations which can be exploited by persons with malicious intent. NNR act.

The plant is expected to be in RRA conditions on average for 5.8% of the year and the contribution to core damage frequency is 18.2% (1.24E-06/r.y). The main initiating events contributing to the core damage frequency in RRA conditions are draindown LOCA with 5.11E-07/r.y (41%), followed by LOSP with 3.11E-07/r.y (25%).

Thus, the risk is significant when operating with RRA connected. This could possibly be explained by the large reliance on operator action to mitigate accidents occurring under these conditions. This emphasizes the need to investigate automated activation of systems under these operating conditions.

This model conservatively includes the estimation of core damage resulting from internal fire. The core damage frequency from this initiator results in 2.34E-06/r.y (34.4%). Specifically, the primary contributor to CDF (2.23E-06/r.y) is due to unsuppressed severe fire in the fire zone ;

Redacted information provides detailed information on the plant vulnerabilities, key equipment and their locations which can be exploited by persons with malicious intent. NNR act.

Following the update of the Level 1 PSA [4], fire curtains and an additional fire door were installed to separate the electrical boards These modifications greatly decrease the contribution of zone to the fire risk. The fire curtains and the additional fire door were not included in the fire model for the calculation of CDF because an updated Level 1 fire model has not been developed.

7. THE KOEBERG SPENT FUEL POOL PSA

7.1 INTRODUCTION

The objective of this analysis is, firstly, to determine the frequency of Spent Fuel Damage (SFD) of the Koeberg Spent Fuel Pool (SFP) using PSA techniques and considering the impact of the Steam Generator Replacement (SGR) project. The detailed Koeberg SFP PSA is contained in reference [8]. In this analysis initiating events that lead to spent fuel damage were identified and quantified using detailed event trees developed from fault trees. The fault trees in turn were developed through component data analysis and HRA. Additionally, the site personnel and public radiological risk associated with the initiating events is determined and compared against the NNR licensing criteria in RD-0024 [2].

In this assessment, all applicable requirements of the NNR in RD-0024 [2] were met. Where RD-0024 does not give instruction, ASME PSA standards were used as guides in the development of the PSA with the intent that the PSA should meet “Capability Category II” in areas impacting risk informed decisions. Applicable ASME standards are:

- ASME/ANS RA-Sa-2009, Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications [5],
- ANSI/ANS-58.22 (2009), Low Power and Shutdown PRA Methodology (Draft) [6],
- ANS/ASME-58.24 (2010), Severe Accident Progression and Radiological Release (Level 2) PRA Methodology to Support Nuclear Installation Applications (Draft) [9],
- ASME/ANS-58.25 (2010), Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications (Draft) [10].

In the case of the Spent Fuel Pool PSA where no specific PSA standard exists, the above standards were used to the extent practical.

7.1.1 Modelling Assumptions, Limitations and Boundary Conditions

The following assumptions were used in the SFP PSA model:

- For the purposes of ascertaining milestone times for the SFP from the onset of the initiating event to the point of fuel uncovery, the initial temperature of the SFP was assumed to be 50°C (I-PTR starting temperature).
- Make-up from the PTR system to the SFP was not considered due to its limited capacity and because of OTS restrictions.
- Although the seals are qualified for an operating temperature of 80°C, it has been shown that the seals are able to withstand a temperature of 100°C. The seals are therefore assumed to fail at 100°C.
- The Spent Fuel Pool (SFP) heat removal capacity was increased by the modification of the PTR additional cooling loop and therefore the additional cooling loop was used in the analysis
- For the purposes of this analysis, it is assumed that the decay heat in the SFP during normal operation is 4.06 MW. This is the maximum decay heat during normal operation per cycle.
- The maximum heat to be removed during refuelling outage operation is 11.5 MW. During outage operation, any one pump and one heat exchanger are sufficient to prevent boiling when the decay

heat in the pool is 11.5 MW. Thus the accident analysis success criterion is operation of one pump and one heat exchanger.

- As the pump installed for the 3rd train is identical to PTR 001 and 002 PO, it was assumed that this pump and its motor are exposed to common cause failure modes.
- During normal plant operation:
  - PTR Train A is assumed to be in service with Train B on standby.
  - RRI Train A is assumed to be in service with Train B on standby.
  - SEC Train A is assumed to be in service with Train B on standby.
  - 48 VDC Train A and common relaying is supplied by LCA.
  - 48 VDC Train B relaying is supplied by LCB.
- Recovery of the four diesels and recovery of Acacia is not modelled in order to be consistent with the Level 1 PSA model. However, due to the long times available for recovery of SFP cooling, these recoveries could be considered if deemed appropriate at a later stage.
- Due to limited data on grid recoveries; for those milestone times over 24 hours, the non-recovery probability is cut off at 3.9E-05. This assumption is conservative.

7.1.2 Event Timing

In this section the time to boiling and the time to fuel damage is presented using maximum pool heat loads during outages and heat load outside of outages.

Times were calculated for:

- Normal Mode (NM). The post-SGR decay heat load in the SFP is 3.45 MW. For the Level 1 and Level 2 PSA aspects of this study, the post-SGR heat load is bounded by 4.06 MW which is the heat load of the pre-SGR configuration.
- Outage Mode (OM). This mode represents the situation where fuel unloading has been completed. The heat load is assumed to take its maximum value of 11.5 MW.

The calculations determine the times to heat the SFP from 50°C to 100°C, from 100°C (boiling) to the level of 17 m (restarting PTR is not allowed below this level); from 100°C (boiling) to the level of 15.5 m (loss of PTR suction) and from 100°C to the level of 9.85 m (fuel uncovery). These results, which were calculated for various initial pool levels, directly influence the probability of recovery actions.

The change in the normal mode of operation for the post-SGR case will not have any negative effect on the SFP heat-up rate or heat-up and boil-off times. The results of the SFP heat-up rate and boil-off times are presented in Table 7-1. These results remain unchanged when compared to the SFP pre-SGR case.

Table 7-1: Heat-up Times for SFP Operating Modes

11.5 MW									
SFP level [m]	**Total SFP heat capacity [kJ/°C]**	**HUR for 11.5 MW [°C/h]**	**Time from 50°C to 100°C [h]**	**BOR [kg/h]**	**Boil-off Time to 17m [h]**	**Boil-off Time to 15.5m [h]**	**Boil-off Time to 13.25m [h]**	**Boil-off Time to 12.2m [h] (Fuel Uncovery)**	**Boil-off Time to 9.85m [h] (Fuel Damage)**
19.3	4902023	8.45	5.9	18342.9	13.2	21.9	34.8	39.6	49.9
17	3887970	10.65	4.7			8.6	21.6	26.4	36.7
15.5	3226630	12.83	3.9				12.9	17.7	28.0
13.25	2234621	18.53	2.7					4.8	15.1
12.2	1802272	22.97	2.2						9.1
9.85	860755	48.10	1.0						
4.06 MW									
SFP level [m]	**Total SFP heat capacity [kJ/°C]**	**HUR for 4.06 MW [°C/h]**	**Time from 50°C to 100°C [h]**	**BOR [kg/h]**	**Boil-off Time to 17m [h]**	**Boil-off Time to 15.5m [h]**	**Boil-off Time to 13.25m [h]**	**Boil-off Time to 12.2m [h] (Fuel Uncovery)**	**Boil-off Time to 9.85m [h] (Fuel Damage)**
19.3	4902023	2.98	16.8	6475.9	37.5	61.9	98.6	112.2	141.3
17	3887970	3.76	13.3			24.4	61.1	74.7	103.8
15.5	3226630	4.53	11.0				36.7	50.2	79.4
13.25	2234621	6.54	7.6					13.6	42.8
12.2	1802272	8.11	6.2						25.6
9.85	860755	16.98	2.9						

7.1.3 Initiating Event Analysis

The following initiating events were assessed:

- Loss of PTR cooling,
- Loss of PTR inventory (rupture and flow diversion),
- Gate seal failure,
- LOSP,
- Local Boron Dilution,
- Global Boron Dilution,
- Multiple misplacement of fresh fuel assemblies.

The main changes made to the Risk Spectrum SFP PSA model to take into account the impact of SGR are as follows:

- SGR will lead to an increase in the annual frequency assigned to basic event IEC MULTI_\SFP\NM\1 which represents misplacing multiple new assemblies into Region 2 racks. Following SGR there will be 60 new fuel assemblies loaded as opposed to the current fuel loading plan where 53 fresh fuel assemblies are loaded. As such, the conditional Initiating Event Frequency (IEF) of misplacing a new assembly into Region 2 racks will be 3.99E 08/r.y × 60 / 53 or 4.52E-08/ r.y [8].
- The probability parameter assigned to the basic event MIS-N-FUEL; which represents misplacing a single new fuel assembly and is calculated using a probability of 1.13E-09 per assembly movement. Following the SGR, there will be 60 new fuel assemblies loaded. As such, the probability of misplacing a single new fuel assembly directly into the SFP Region 2 racks will be 1.13E-09 × 60 or 6.78E-08 [8].

The accident sequences were quantified for the two operating modes defined. The initiating event frequencies (IEF) are given in Table 7-2. Note that the conditional IEFs assume a full year in the specified operating mode.

Table 7-2: Spent Fuel Pool Initiating Event Frequencies

Initiator	Time Fraction	Conditional IEF	IEF
Catastrophic Gate Failure (NM)	9.42E-01	5.37E-03	5.06E-03
Loss of SFP Cooling (NM)	9.42E-01	9.52E-04	8.97E-04
Loss of SFP Cooling (OM)	5.80E-02	1.17E-02	6.79E-04
Loss of SFP Inventory due to diversion (NM)	9.42E-01	3.57E-02	3.36E-02
Loss of SFP Inventory due to diversion (OM)	5.80E-02	3.57E-02	2.07E-03
Loss of SFP Inventory due to PTR Pipe Rupture (NM)	9.42E-01	1.30E-08	1.22E-08
Loss of SFP Inventory due to PTR Pipe Rupture (OM)	5.80E-02	1.30E-08	7.54E-10
LOSP (NM)	9.42E-01	4.50E-01	4.24E-01
LOSP (OM)	5.80E-02	1.54E+00	8.93E-02
Local Boron Dilution (NM)	9.42E-01	1.36E-04	1.28E-04

Initiator	Time Fraction	Conditional IEF	IEF
Local Boron Dilution (OM)	5.80E-02	2.30E-03	1.33E-04
Global Boron Dilution (NM)	9.42E-01	1.82E-05	1.71E-05
Global Boron Dilution (OM)	5.80E-02	3.09E-04	1.79E-05
Multiple misplacement of fresh assemblies (NM)	9.42E-01	4.52E-08	4.26E-08

7.2 SOURCE TERM

For the SFP Level 3 PSA [8], it was demonstrated that the post-SGR without Thermal Power Uprate (TPU) SFP source term is not significantly different from the pre-SGR SFP source term [42]. It is therefore acceptable to continue using the pre-SGR SFP source term with associated decay heat for the post-SGR without TPU case.

7.3 RESULTS

The Spent Fuel Pool PSA [8] concluded that a conservative estimate of the frequency of Spent Fuel Damage post-SGR is 5.03E-08/r.y. The results in this section exclude external hazards which are assessed separately in Section 12 and then included in the overall results presented in Section 13 and the Executive Summary. The fuel damage frequencies for the initiating events considered in the SFP PSA for both Normal Mode and Outage Mode are shown in Table 7-3.

Table 7-3: Fuel Damage Frequency for Normal Mode and Outage Mode

Initiating Event	Outage Mode		Normal Mode	
	Events/Year	% of FD	Events/Year	% of FD
Multiple misplacement of fresh assemblies	-	-	4.26E-08	84.60
Catastrophic door seal failure	-	-	2.46E-10	0.49
SFP LOSP	1.35E-09	2.69	4.81E-09	9.56
Loss of PTR inventory due to flow diversion	4.77E-10	0.95	6.46E-10	1.28
Local boron dilution	5.69E-11	0.11	5.48E-11	0.11
Global boron dilution	7.65E-12	0.02	6.19E-12	0.01
Loss of SFP cooling due to PTR system failure	9.16E-11	0.18	4.17E-12	0.01
Loss of SFP inventory due to PTR pipe rupture	1.69E-21	0.00	1.69E-21	0.00
Operating Mode Frequency	**1.99E-09**	**3.94**	**4.83E-08**	**96.06**
Overall Fuel Damage Frequency	**5.03E-08**			

The peak and average public and site personnel risks for the station are shown in Table 7-4 using both the MAAP (best estimate) and NRC (conservative) release fractions. Note that the public risk results are a subset of the overall Level 3 public risks reported in Section 9, and the site personnel risks are a subset of the overall site personnel risks reported in Section 11.

Table 7-4: Station Spent Fuel Pool Risks

Risk Criteria	MAAP RF		NRC RF	
	Annual Risk	% of NNR Criteria	Annual Risk	% of NNR Criteria
Peak Public Risk (fatalities year^{-1})	5.04E-09	0.10%	8.36E-09	0.17%
Average Public Risk (fatalities person^{-1} year^{-1}) Using 2008 national population[1]	6.87E-11	0.69%	3.15E-10	3.15%
Average Public Risk (fatalities person^{-1} year^{-1}) Using 2011 national population[2]	6.46E-11	0.65%	2.96E-10	2.96%
Average Public Risk (fatalities person^{-1} year^{-1}) Representative of year 2025[3]	1.58E-10	1.58%	7.25E-10	7.25%
Peak Site Personnel Risk (fatalities year^{-1})	8.40E-08	0.17%	1.19E-07	0.24%
Average Site Personnel Risk (fatalities person^{-1} year^{-1})	8.40E-08	0.84%	1.19E-07	1.19%

1. The national population in 2008 was 48 687 000. This was included to use the same population year as per the Level 3 PSA.
2. The national population in 2011 was 51 770 560.
3. 2008 average public risk scaled by 2.3 to be representative of the year 2025 (see Section 9.1.7).
4. Scaling of the peak and average site personnel risks by a factor of 10 are included (see Section 5.6).

7.4 INSIGHTS AND CONCLUSIONS

When using the more conservative NRC release fractions, the Koeberg SFP PSA study [8] considering the impact of SGR results in a station average public risk representative of year 2025 of 7.25E-10 fatalities person-1 year-1 (7.25% of the NNR criterion) and a peak public risk of 8.36E-09 fatalities per year (0.17% of the NNR criterion) . Thus, the SFP risk remains within the NNR limit in the SGR configuration.

The most dominant contributions to the fuel damage frequency result from Multiple Misplacement of Fresh Assemblies (NM) and LOSP (NM & OM).

The accident Multiple Misplacement of Fresh Assemblies is essentially where the new assemblies are loaded into Region 2 racks side-by-side. It is assumed that this will lead to criticality occurring without any dilution of the boron in the spent fuel pool water. There are a number of measures taken to prevent this misplacement from occurring so it is an unlikely event but it remains one of the dominant spent fuel pool accidents.

In the case of LOSP leading to SBO the JPS pump and motor (JPS 001 PO and JPS 001 MO) and the SED valve (SED 981 VD\P) is essential to avoid Spent Fuel Damage. It is therefore important to maintain these systems and keep the equipment in a good, operable condition.

In the event of a Loss of 400 kV grid and Loss of Acacia, the 9 LKK board supplying the SED system, which is the preferred source for SFP make-up, is not available. JPP and JPS then becomes the only source of SFP make-up emphasising the importance of maintaining these systems in a good operable condition and having spares available.

The FDF from internal events is expected to be significantly lower than CDF since the decay heat in the SFP is much lower than the heat load in the reactor. Therefore, it is not surprising that the FDF is in the order of 100 times less than the estimated CDF. However, without a containment structure to prevent fission product releases, the consequences can be much worse making the overall public risks comparable.

In the present study it is conservatively assumed that no Emergency Plan countermeasures are applied after or during the accident other than a ban on irradiated foodstuffs. In practice it is required that the Precautionary Action Zone (PAZ) of 5 km and the Urgent Protective Action Planning Zone (UPZ) of 16 km are evacuated within 4 hours and 16 hours respectively. Both these evacuation times are less than the time to spent fuel uncovery and spent fuel damage in most sequences modelled.

8. THE KOEBERG LEVEL 2 PSA

8.1 INTRODUCTION

The objective of the Koeberg Level 2 PSA is to determine the frequency and magnitude of releases from severe accidents occurring in the reactor. The detailed assessment is contained in reference [11].

In this assessment, all applicable requirements of the NNR in RD-0024 [2] were met. Where RD-0024 does not give instructions, ASME PSA standards were used as guides in the development of the PSA with the intent that the PSA should meet “Capability Category II” in areas impacting risk informed decisions. Applicable ASME standards are:

- ASME/ANS RA-Sa-2009, Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications [5],
- ANS/ASME-58.24 (2010), Severe Accident Progression and Radiological Release (Level 2) PRA Methodology to Support Nuclear Installation Applications (Draft) [9].

The Koeberg Level 2 PSA is an assessment of the challenges to the containment, a description of the possible containment responses and their estimated probabilities, and an assessment of the consequent releases to the environment. This assessment includes the inventory of material released, its physical and chemical characteristics, and information on the time, energy, duration and location of the release.

8.2 MODULAR ACCIDENT ANALYSIS PROGRAM

Modular Accident Analysis Program (MAAP) Version 4.0.7 is used extensively in the Level 2 PSA. MAAP treats the spectrum of physical processes that could occur during a severe accident (e.g. core heat-up, hydrogen evolution, vessel failure, ignition of combustible gases and fission product release, transport and deposition). MAAP treats all the important engineered safety systems such as emergency core cooling, containment sprays, and power operated relief valves. In addition, MAAP allows operator interventions and incorporates these in a flexible manner, permitting the user to model operator behaviour. Specifically, the user models the operator specifying a set of variable values and/or events, which are the operator intervention conditions, combined with associated operator actions.

8.3 LEVEL 1 / LEVEL 2 INTERFACE

The interface between the Koeberg Level 1 Systems Analysis and the Level 2 Containment Analysis consists of a set of Plant Damage States (PDS). The plant damage states are defined by a set of functional characteristics for system operation which are important to accident progression, containment failure and source term definition. Each PDS contains Level 1 sequences with sufficient similarity in functional characteristics that the containment accident progression for all sequences in the group can be considered to be essentially the same. Each PDS defines a unique set of conditions regarding the state of the plant, the containment building systems, the physical state of the core, the Primary Coolant System and the containment boundary at (approximately) the time of core damage / vessel failure.

All the new Level 1 PSA sequences that lead to core damage in SGR configuration have been characterized with a PDS to allow appropriate interface with the Level 2 PSA.

8.4 CONTAINMENT BUILDING FAILURE MODES

The Level 2 PSA also defines containment failure modes and associates, depending on the accident progression, a probability (or split fraction). These split fractions have been reviewed to ensure they are still applicable in the SGR configuration. The containment failure modes considered in the Level 2 PSA are:

- Induced SGTR
- Containment Bypassed
- Hydrogen Burn Prior to Vessel Failure
- In Vessel Steam Explosion
- "Vessel Rocketing"
- Direct Containment Heating
- Ex-Vessel Steam Explosion
- Ex-Vessel Hydrogen Burn
- Containment Failure Due to Long Term Over Pressurisation
- Containment Failure Due to Long Term Hydrogen Burn
- Containment Failure Due to Basemat Melt Through.

8.5 CONTAINMENT STRENGTH UNDER STATIC PRESSURE LOADS

The Koeberg specific containment fragility curve with a mean containment failure pressure of 6.73 bar abs [11] is used in the updated baseline. This leads to revised split fractions for early hydrogen burn and direct containment heating (DCH).

The fragility curve was expressed in the form of a log-normal distribution relating the internal pressure to the probability of containment failure at less than or equal to that pressure. The parameters for the updated fragility curve are

$Median\ [bar] = \exp(\alpha)$ and $Mean\ [bar] = \exp(\alpha + \frac{1}{2}\varphi^2)$

- $\alpha = 1.90$
- $\varphi = 0.092$
- Error Factor = 1.163
- Median = 6.70 bar abs
- Mean = 6.73 bar abs
- 5 percentile = 5.76 bar abs
- 95 percentile = 7.79 bar abs.

Using the above parameters, the following containment fragility curve **Figure 8-1** has been considered to perform the Level 2 PSA quantification with SGR configuration.

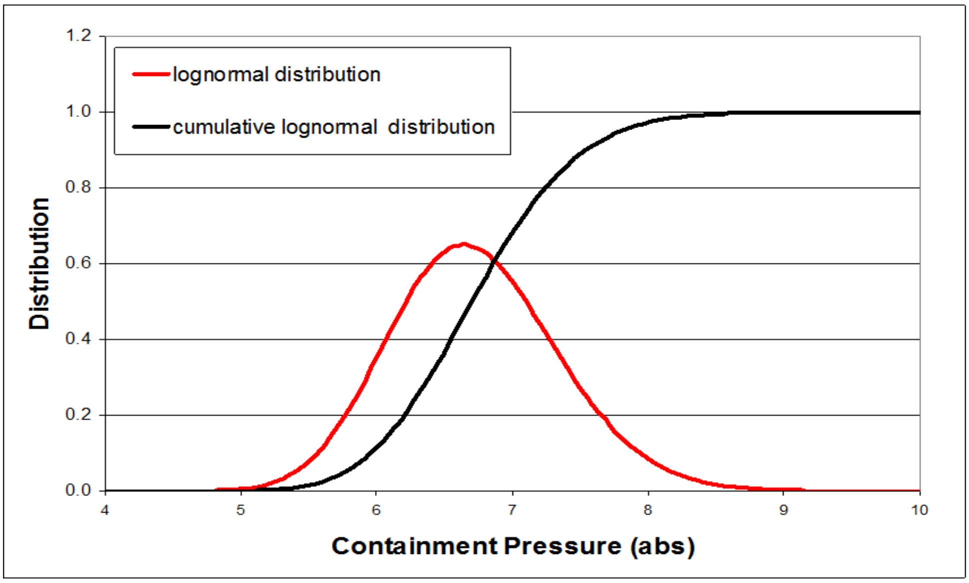

Figure 8-1: Containment Fragility Curve

8.6 RELEASE CATEGORIES

Accidents in a Release Categories (RC) are expected to have similar radiological characteristics and potential off-site consequences. The RCs are defined on the basis of attributes that affect fission product releases and accident consequences. Accidents with the same Release Category are similar in terms of the Source Term characteristics. These characteristics include the magnitude, timing, elevation and energy of the release of radioactivity. Table 8-1 lists the release categories used in the Level 2 PSA.

Table 8-1: Release Categories

Release Category	Definition of Release Category
RC-1	Containment integrity is maintained. Containment spray system operable. This results in a slow release of radioactivity into the external environment at the containment design leakage rate.
RC-2	Containment is not isolated, (CI Failure). Containment spray system is operable. This results in a slow filtered release of radioactivity into the external environment.
RC-3	Containment is by-passed or not isolated (SGTR or ISLOCA). The containment remains unisolated, resulting in a continuous radioactive release into the external environment. Containment Spray function is irrelevant in by-pass accidents, but Containment Spray is impaired in CI failure accidents resulting in an unfiltered release.
RC-4	Early containment failure. This corresponds to the total and early loss of containment integrity and a direct release of radioactivity into the external environment within a few hours of the start of the accident.
RC-5	Late containment failure. Containment Spray System operable. This results in a direct release of radioactivity into the external environment and is considered to occur approximately one day into the accident.

RC-6	Late containment failure. Containment spray function impaired. This results in a direct release of radioactivity into the external environment and is considered to occur approximately one day into the accident.
RC-7	Basemat melt-through. Containment Spray System operable. This results in a ground level release of radioactivity into the external environment and is considered to occur several days into the accident. (Koeberg's basemat is 6,7 m thick but the area below the basemat is open to the external atmosphere.)
RC-8	Basemat melt-through. Containment spray function impaired. This results in a ground level release of radioactivity into the external environment several days into the accident.

8.7 SEVERE ACCIDENT MANAGEMENT GUIDELINES

The following SAMGs make a significant impact on the Koeberg Level 2 PSA Study. Further, even in these guidelines credit is only given to equipment which is appropriately designed, maintained, inspected and tested.

SCG-1: Mitigate fission product releases

SCG-2: Depressurise Containment

SCG-3: Mitigate MCCI

SAG-2: Depressurise the RCP

SAG-3: Inject into the RCP

SAG-5: Control Containment Conditions

By comparison to Control Room operator training in the procedures used prior to core damage, the training for the Technical Support Centre (TSC) personnel is not as extensive and so the reliability of the TSC personnel using SAMGs is generally not expected to be as high as the Control Room operators when they are using the Emergency Operating Procedures (EOPs).

8.8 INITIAL INVENTORY

The steam generator modelling has been adapted to reflect the new design and the residual heat and the source term have been revised according to the reference core approach (see reference [32]).

8.9 RESULTS

The isotopic release fractions, times of release and energy of release of the most likely accident sequences for each Release Category for Koeberg are determined. This source term will then serve as input into the Level 3 PSA.

A summary of the Release Frequency results for internal events, including internal flooding and internal fire, and aircraft crash is presented in Table 8-2.

Table 8-2: Koeberg Release Category Results (Internal Events and Aircraft Crash)

RC	Release Category Definition	Frequency (/r.y) [11]	Percentage (%)	Error Factor
1	Containment integrity is maintained	5.94E-06	86.6	2.9
2	Containment is not isolated.	1.90E-08	0.3	4.2
3	Containment is not isolated/by-passed	3.42E-08	0.5	2.7
4	Early Containment Failure	1.74E-07	2.5	3.5
5	Late containment failure (with containment sprays)	- [1]	-	-
6	Late containment failure (no containment sprays)	4.53E-07	6.6	10.8
7	Basemat melt-through (with containment sprays)	7.63E-08	1.1	12.9
8	Basemat melt-through (no containment sprays)	1.64E-07	2.4	9.2

The results show that the most frequent release category is the one in which the containment integrity is maintained (RC-1: 86.6%). This means that in case of core damage, the containment response is appropriate in more than 86% of the cases. The second most frequent release category is the one where the containment fails because of overpressure in the containment (RC-6: 6.6%). This is mainly due to the loss of containment spray system.

Table 8-3 presents an overview of the Level 2 PSA results for internal events, including internal flooding and internal fire, and aircraft crash.

Table 8-3: Containment Failure Results (Internal Events and Aircraft Crash)

	No Containment Failure	Early Containment Failure	Late Containment Failure
Frequency per reactor year [11]	5.94E-06	2.27E-07	6.93E-07
Percentage Frequency per reactor year	86.6%	3.3%	10.1%

The Large Early Release Frequency (LERF) contributes to 3.3% of the overall core damage frequency.

Redacted information provides detailed information on the plant vulnerabilities, key equipment and their locations which can be exploited by persons with malicious intent. NNR act.

The principal contributors to LERF are Aircraft Crash (43.5%), Fire in Zone [redacted] 19.5%), Steam Generator Tube Rupture (12.4%) and Loss of electrical switchboards accidents (7.0%). A more detailed overview of the dominant contributors to LERF is shown in Figure 8-2.

[1] The frequency of RC-5 is zero due to the installation of hydrogen passive auto-catalytic recombiners which eliminates the threat of late hydrogen burn.

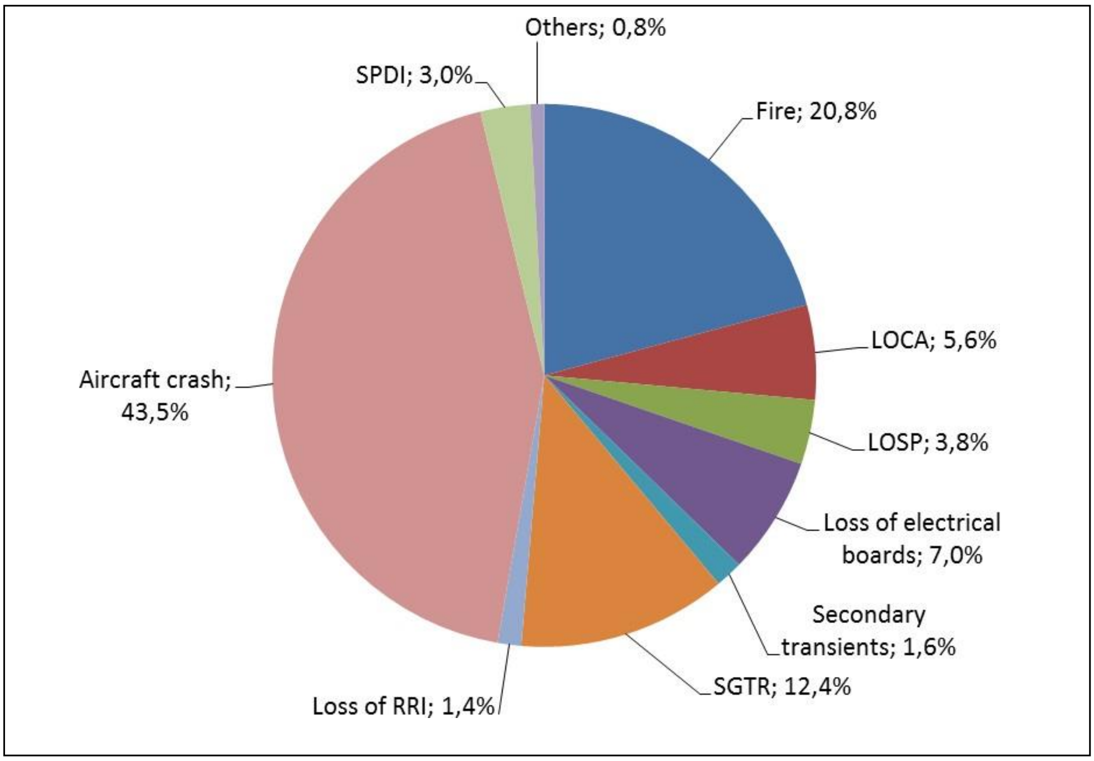

Figure 8-2: Dominant Contributors to Large Early Release Frequency (LERF)

9. THE KOEBERG LEVEL 3 PSA

9.1 INTRODUCTION

The objective of the Level 3 PSA is to assess the off-site radiological consequences of accidental releases of radioactive materials into the atmosphere from Koeberg NPP and to compare those consequences to the safety standards of the NNR to determine compliance. Specifically, only the health impact of such releases on the population in an area lying within 60 km of site will be evaluated. No attempt is made in this study to assess the economic or social consequences of accidents. Operating experience from the Fukushima Daiichi NPP accident suggests that these consequences can be more significant than health impacts from exposure to radiation.

In this assessment, all applicable requirements of the NNR in RD-0024 [2] shall be met. Where RD-0024 does not give instruction, the draft ASME Level 3 PSA standard shall be used as a guide in the development of the PSA with the intent that the PSA should meet "Capability Category II" in areas impacting risk informed decisions. In this case, the applicable ASME standard is:

- ANS/ASME-58.25 (2010), Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications (Draft) [10].

9.1.1 NNR RD-0024 Risk Requirements

The Koeberg Nuclear Installation License [1] requires the licensee to demonstrate that its operation does not expose the public and site personnel to unacceptably high risks in the event of accidental releases of radioactivity into the environment. It must demonstrate its safety by meeting certain safety criteria set by the NNR in RD-0024 [2]. These risk criteria are given below:

- The average annual mortality risk to a member of the public due to accidents must not exceed 10^{-8} deaths per person per year.
- The peak individual risk to a member of the public must not exceed 5×10^{-6} deaths per year.
- A bias against larger accidents is imposed by the requirement that the annual average frequency *f(N)* of accidents resulting in more than N fatalities must be less than $A(N^{-1} - N_p^{-1})$, where A is a constant.
- The average annual mortality risk for site personnel must not exceed 10^{-5} deaths per person per year.
- The peak individual risk to site personnel must not exceed 5×10^{-5} deaths per year.

9.1.2 Level 3 PSA Scope

The Koeberg Level 3 PSA [12] is concerned with the transport of radioactive materials through the environment and the effect that the exposure to radiation will have on the population. The scope of this analysis is limited to the analysis of the following endpoints in the absence of countermeasures:

- Number of early fatalities;
- Number of latent cancer fatalities;

- Short-term individual doses;
- Long-term individual doses;
- Individual risk of early effects;
- Individual risk of late effects.

9.1.3 Health Effects

The exposure of individuals to radiation can lead to deterministic or stochastic health effects.

Deterministic effects result from exposure of the whole or part of the body to high doses of radiation. Their severity is observed to increase with dose and there is usually a threshold dose below which the effects are not induced.

Stochastic effects of radiation include increased incidence of cancer among the exposed population and of hereditary disease in their descendants. For stochastic effects the probability of occurrence, but not severity, depends on the radiation dose.

Deterministic effects and stochastic effects are often referred to as 'early' effects and 'late' effects, respectively.

9.1.4 PC Cosyma

The Koeberg Level 3 PSA uses PC Cosyma Version 2.02 to derive the radiological consequences of severe accidents in the reactor vessel and in the Spent Fuel Pool.

9.1.5 Main Assumptions

The two major assumptions that have been made in this analysis are:

- Further irradiation of the exposed population through ingestion of food is not considered, and
- No emergency plan (countermeasures), other than the food ban, to mitigate the accident has been assumed.
- Cumulative population data (sum of the permanent resident and tourist populations) based on the 2001 population census projected for the year 2008 has been used. Population projections for the end of plant life is not considered and is justified given that the Koeberg PSA is maintained as a living PSA and updated periodically.

Explanation and justification for these assumptions can be found in the Koeberg Level 3 PSA [12].

9.1.6 Meteorological Data

The meteorological conditions attributed to the site were obtained from 2 years of hourly measurements taken at the meteorological station at Koeberg. The meteorological data for the years 2006 and 2007 were used in the analysis.

9.1.7 Population Data

The population data is based on the 2001 population census projected for the year 2008 as documented in the latest demography section of the Duynefontein Site Safety Report and takes into account the transient / tourism figures within 60 km of site, i.e., it represents the 2008 projected cumulative population (sum of the permanent resident and tourist populations).

The population data used in this study will be updated when an updated Duynefontein Site Safety Report containing revised population data based on the 2011 census data is issued. This is to ensure that configuration control between the two documents is maintained.

The population data section of the Duynefontein Site Safety Report also provides projected cumulative population data for 2016 which is evaluated as part of the sensitivity analysis presented in the Level 3 PSA [14].

9.2 METHODOLOGY

9.2.1 The Peak and Average Public Risks

The peak individual risk refers to the highest risk that a member of the public may be subjected to as a result of a nuclear accident. The risk will be greatest closest to the plant. In the case of Koeberg, the point closest to the site at which the public may be found is the site boundary. The site boundary at Koeberg extends from 1,3 km to 2,5 km from site. Therefore, the peak public risk will be calculated at the distance band from 1km to 2,5 km. Thus, the risk at 1.75 km, which is the most representative distance where dose consequences are calculated in PC Cosyma, is given as annual fatalities at 1.75 km divided by the number of people in that distance band.

If $D_{i(1.75\ km)}$ is the total mean number of fatalities at 1,75 km due to release category RC_i, f_i is the frequency of the release category (RC_i) and $P_{(1,75\ km)}$ is the total population at 1,75 km then the peak public risk is given by:

$$\textit{Peak public risk} = \frac{\sum_{i=1}^{8}(D_{i(1.75\,km)} \times f_i)}{P_{(1.75\,km)}}$$

The average public risk to an individual in the national population is defined as the number of annual fatalities divided by the total national population. The total national population figure is 51.77 million from the 2011 census data.

Therefore, if D_i is the total mean number of fatalities summed over all distances due to release category RC_i and f_i is the frequency of the release category (RC_i), then the average risk to the public is given by:

$$\textit{Average Public Risk} = \frac{\sum_{i=1}^{8}(D_i \times f_i)}{\textit{Total National Population}}$$

In accordance with the NNR accepted interim arrangement for addressing the use of outdated population data in PSA Level 3 studies, the average public risk result was adjusted further as follow:

- Firstly, scaled by a factor of 1.0633 to be representative of the year 2008, to ensure that the total national population and the small area projected cumulative population data (i.e. the number of people in the 1.75 km distance band) are both for the year 2008;

- Then scaled further by a factor of 2.3 to be representative of the year 2025.

9.2.2 Peak and Average Site Personnel Risks

The individual risk is defined as the maximum risk inside the 1 km radius. Since all the personnel are exposed to this risk, this corresponds to the peak and average site personnel risks due to severe accidents.

Using PC Cosyma to estimate doses on site (e.g., close to the site of the accident) can lead to considerable uncertainty. The site personnel risks from severe accidents can therefore only be regarded as a poor approximation but that this approximation is far from the allowable limits that a considerable factor of increase could be imposed without effecting the conclusions of the report.

In this regard Eskom acknowledges the NNR's acceptance communicated in letter k27997N [48] of the proposed application of a scaling (or risk increase) factor of 10.0 derived in the report entitled "Comparison of PC Cosyma and ARCON2 Near-Field Atmospheric Dispersion Factors", Number: PSA-R-T18-03, Revision 2 [47], to be applied to the PC Cosyma results at 500 m that are used to approximate the peak and average risks to site personnel from accidents related to releases from core damage, the spent fuel pool and casks accidents, in order to appropriately adjust the PC Cosyma results at 500 m to be more representative and bounding of the results expected at a distance of 100 m. Accordingly, the peak and average risks to site personnel from accidents related to releases from core damage, the spent fuel pool and casks accidents were adjusted / scaled by a factor of 10.

The majority of severe accidents take several hours to develop into a release from the containment building. This allows for the implementation of the Emergency Plan, which involves the evacuation of non-essential personnel and the sheltering of the rest. Given these protective actions the site personnel doses should be limited.

9.2.3 Bias Against Larger Accidents

A bias against larger accidents is imposed by the requirement that the annual average frequency $f(N)$ of accidents resulting in more than N fatalities be less than $A(N^{-1} - N_p^{-1})$, where A is a constant determined by limiting the mean number of fatalities per person per annum to 10^{-8} in the range $1 < N < N_p$, where N_p is an acceptable projection of the national population.

Mathematically, it is as follows:

$$f(\geq N) < A\left(\frac{1}{N} - \frac{1}{N_p}\right)$$

$$= \frac{A}{N} \qquad \text{for } N_p >> N$$

where $f(\geq N)$ = Annual frequency of any accident in which N is equalled or exceeded.

The probability density function, *F(N)*, for having *N* fatalities per annum is chosen with the following form in accordance with reference [2]:

$$F(N) = \frac{A}{N^2}$$

The parameter *A* is independent of *N*. The number of fatalities, *N*, serves as a measure of the magnitude of large accidents. *F(N)* effects the bias against larger accidents by suppressing *F(N)* for large *N*.

The average annual population risk is then given by:

$$\langle N \rangle = \frac{1}{N_p} \int_1^{N_p} F(N) N dN$$

$$= \frac{1}{N_p} \int_1^{N_p} \frac{A}{N^2} N dN$$

$$= A \frac{\ln(N_p)}{N_p}$$

The quantity *A* is determined by the condition:

$$\langle N \rangle = C$$

where C is average population risk criterion of 10^{-8} y^{-1} per site.

Therefore, $A = C \frac{N_p}{\ln(N_p)}$

From the 2011 census data the population figure is 51.77 million.

$$A = 10^{-8} \times \frac{51.77 \times 10^6}{\ln(51.77 \times 10^6)} = 0.0291$$

From the 2008 census data the population figure is 48.687 million.

$$A = 10^{-8} \text{x} \frac{48.687 \text{ x } 10^6}{\ln \ (48.687 \text{ x } 10^6)} = 0.0257$$

It can therefore be stated that the frequency of accidents resulting in N or more fatalities must conservatively be less than 0.0275 N^{-1}.

9.3 RESULTS

The Level 3 PSA results [12] are presented in the tables below. These results are for aircraft crash initiator and Internal Events initiators (including internal Fire and internal Flood initiators as per the ASME PSA Standards.) for all plant operating states. Note that aircraft crash and internal events initiators are also included in the Level 1 PSA and so carried through to the Level 2 and Level 3 PSA.

The determination of the overall (total) peak and average public and site personnel risk results from all internal and external events are discussed and presented in Chapter 12 of this report. The bias against larger accidents is discussed in Chapter 13 of this report.

Table 9-1: Risk Comparisons of Release Categories (for one Unit)[12]

Release Category	Frequency	Average Public Conditional Risk[1]	Average Public Risk	% Risks	Peak Public Conditional Risk	Peak Public Risk	% Risks
RC-1	5.94E-06	2.87E-07	1.71E-12	4.97%	2.93E-04	1.74E-09	8.56%
RC-2	1.90E-08	5.21E-05	9.89E-13	2.88%	4.07E-02	7.73E-10	3.80%
RC-3	3.42E-08	3.65E-04	1.25E-11	36.35%	9.69E-02	3.31E-09	16.28%
RC-4	1.74E-07	4.63E-05	8.06E-12	23.45%	3.82E-02	6.65E-09	32.64%
RC-6	4.53E-07	7.50E-06	3.40E-12	9.89%	4.51E-03	2.04E-09	10.03%
RC-7	7.63E-08	6.25E-05	4.77E-12	13.88%	3.60E-02	2.75E-09	13.48%
RC-8	1.64E-07	1.80E-05	2.95E-12	8.58%	1.89E-02	3.10E-09	15.21%
TOTAL (2011)[1]	**6.86E-06**	**5.52E-04**	**3.44E-11**	**100.00%**	**2.35E-01**	**2.04E-08**	**100.00%**
TOTAL (2008)[2]		**-**	**3.65E-11**				
TOTAL (2025)[3]		**-**	**8.40E-11**				
NNR Criteria			**1.00E-08**			**5.00E-06**	

[1] Average public risk based on 2011 national population.
[2] 2011 average public risk scaled by 1.0633 to be representative of the year 2008.
[3] 2008 average public risk scaled by 2.3 to be representative of the year 2025.

Table 9-2: Annual Risks For The Station From Reactor Accidents [12]

Criteria	Annual Risk (per Unit)	Annual Risk (Station)	% of NNR Criteria
Peak Public Risk (fatalities year^{-1})	2.04E-08	4.07E-08	0.81%
Average Public Risk (fatalities person^{-1} year^{-1}) (using 2011 national population)	3.44E-11	6.87E-11	0.69%
Average Public Risk (fatalities person^{-1} year^{-1}) (representative of year 2008)	3.65E-11	7.31E-11	0.73%
Average Public Risk (fatalities person^{-1} year^{-1}) (representative of year 2025)	8.40E-11	1.68E-10	1.68%
Peak Site Personnel Risk[1] (fatalities year^{-1})	5.45E-07	1.09E-06	2.18%
Average Site Personnel Risk[1] (fatalities person^{-1} year^{-1})	5.45E-07	1.09E-06	10.90%

[1] Scaling of the peak and average site personnel risks by a factor of 10 are included.

Table 9-3: RC-3 30-day Organ Doses at 3,75 km[12]

		Pathway % Contribution		
Organ	**Dose (Sv)**	**Cloudshine**	**Groundshine**	**Inhalation**
Lung	8.31E-01	1.81	46.26	51.92
Thyroid	2.84E+00	0.60	15.00	84.37
Eye Lens	4.26E-01	3.97	96.03	0.00
Ovaries	3.67E-01	3.24	93.70	3.05
B. Marrow	3.93E-01	3.55	92.45	4.00
GI-Tract	4.83E-01	2.65	68.40	28.95
Effective	5.78E-01	2.58	63.67	33.74

Table 9-4: RC-3 50-year Organ Doses at 3,75 km[12]

		Pathway % Contribution		
Organ	**Dose (Sv)**	**Cloudshine**	**Groundshine**	**Inhalation**
B. Marrow	1.66E+00	0.84	95.77	3.38
B. Surface	1.91E+00	0.94	88.09	10.95
Breast	1.66E+00	0.99	97.45	1.56
Lung	2.59E+00	0.58	65.00	34.39
Stomach	1.55E+00	0.88	96.96	2.15
Colon	1.62E+00	0.79	89.13	10.08
Liver	1.57E+00	0.87	95.30	3.82
Pancreas	1.43E+00	0.89	96.86	2.25
Thyroid	4.43E+00	0.39	42.22	57.37
Gonads	1.59E+00	0.82	97.29	1.89
Remainder	1.71E+00	0.90	96.85	2.25
Effective	1.91E+00	0.78	84.60	14.61

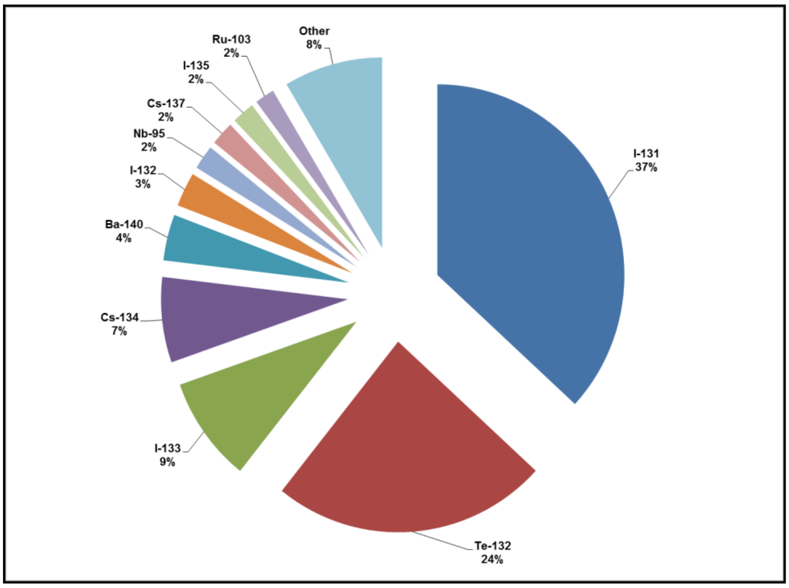

Figure 9-1: Dominant isotope contribution to the RC-3 30 day dose at 3,75 km

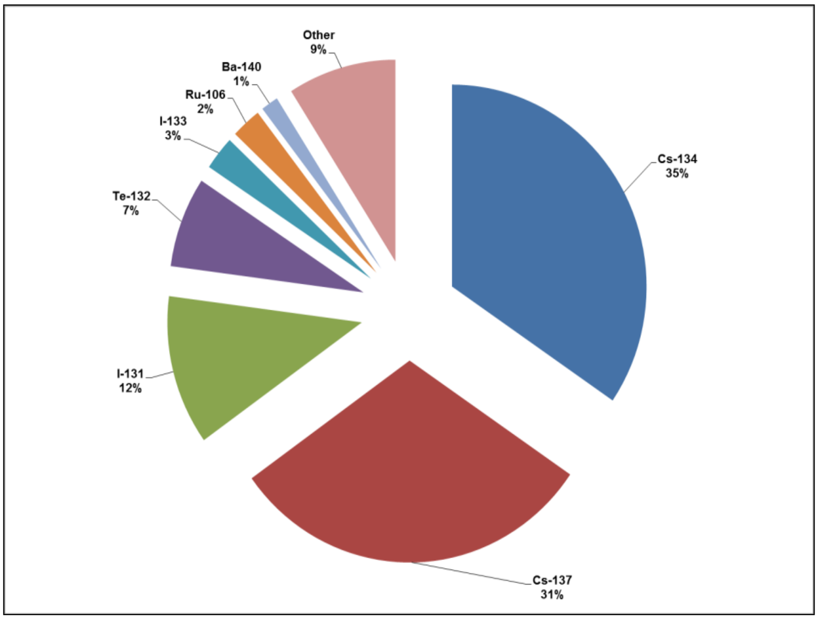

Figure 9-2: Dominant isotope contribution to the RC-3 50 year dose at 3,75 km

10. THE KOEBERG SPENT FUEL CASK PSA

The Koeberg cask project envisages that at most 7 HOLTEC casks will be loaded, transferred and stored per year, resulting in 161 casks being stored at the TISF including the 4 CASTOR X/28F casks from the CSB. The risk assessment for additional metal casks [29] considered a comprehensive and generic list of initiating events developed by EPRI, and screens out seismic events since it was assumed the161 casks will be stored in the TISF.

The cask loading campaign in 2021 resulted in 7 HOLTEC casks being stored in the CSB which brings the total number of casks in storage to 11 (including the 4 CASTOR X/28F casks) and further cask loadings are planned. A re-analysis of the risk assessment was conducted in PSA18-0043 [34] which also assumes 7 cask movements per year, resulting in eighteen (18) casks being stored in the CSB and screened in seismic event as an initiator. The scope of the assessment considers cask handling activities in the SFP building and casks being stored in the CSB.

In the assessment, all applicable requirements of NNR RD-0024 [2] were met. Where RD-0024 does not give instruction, ASME PSA standards were used as guides in the development of the PSA with the intent that the PSA should meet “Capability Category II” in areas impacting risk informed decisions. In this case, the applicable ASME standards are:

- ASME/ANS RA-Sa-2009, Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications [5],
- ANSI/ANS-58.22 (2009), “Low Power and Shutdown PRA Methodology, Draft” [6],

- ANSI/ANS/ASME-58.24 (2010), “Severe Accident Progression and Radiological Release (Level 2) PRA Methodology to Support Nuclear Installation Applications, Draft” [9], and
- ANS/ASME-58.25 (2010), Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications (Draft) [10].

Eskom acknowledges the NNR’s acceptance communicated in letter k27997N [50] of the proposed application of a scaling (or risk increase) factor of 10.0 derived in the report entitled “Comparison of PC Cosyma and ARCON2 Near-Field Atmospheric Dispersion Factors”, Number: PSA-R-T18-03, Revision 2 [49], to be applied to the PC Cosyma results at 500 m that are used to approximate the peak and average risks to site personnel from accidents related to releases from core damage, the spent fuel pool and casks accidents, in order to appropriately adjust the PC Cosyma results at 500 m to be more representative and bounding of the results expected at a distance of 100 m. Accordingly, the peak and average risks to site personnel from seismic events related to releases from cask accidents were adjusted / scaled by a factor of 10.

Table 10-1 present the respective overall peak and average public and site personnel risk post-SGR results for cask accidents including internal events, aircraft crash induced fire and seismic events, and compares cask storage risks to the NNR criteria.

Table 10-1: Post-SGR Total Annual Risk to the Station from Cask Accidents

Description	Cask Seismic and Aircraft Crash	%of NNR Criteria	Station Annual Risk from Cask Accidents (Internal and Seismic and Aircraft Crash)	% of NNR Criteria
Peak Public Risk (fatalities year ^{1})	2.74E-08	0.55%	5.23E-08	1.05%
Average Public Risk (fatalities person^{-1} year^{-1}) (representative of year 2008)	1.86E-10	1.86%	7.56E-10	7.56%
Average Public Risk (fatalities person^{-1} year^{-1}) (representative of year 2025)	4.28E-10	4.28%	1.74E-09	17.40%
Peak Site Personnel Risk (fatalities year ^{1})	6.21E-07	1.24%	9.40E-07	1.88%
Average Site Personnel Risk (fatalities person^{-1}	6.21E-07	6.21%	8.94E-07	8.94%

The cask PSA does not include an assessment of the risks associated with the transportation of the casks to a permanent off-site facility. Risks associated with the normal operation of the casks are outside the scope of this document. Sabotage has been excluded from consideration as this challenge is addressed by separate and more confidential studies as recommended by Electric Power Research Institute (EPRI) and is normally carried out by the regulatory authority [36]. Considering that the Koeberg PSA is maintained as a “living” PSA, this study will be updated periodically to reflect any changes in the dry cask storage system.

11. THE KOEBERG SITE PERSONNEL PSA

11.1 INTRODUCTION

In compliance with the Koeberg Nuclear Installation License [1], an evaluation of the risk to site personnel from nuclear accidents is performed and compared to the licensing risk criteria specified by the NNR in RD-0024 [2] (see Table 11-1). The detailed site personnel risk assessment may be found in [13].

Table 11-1: NNR Licensing Criteria For Site Personnel Risk

Personnel Risk	Accidents	Normal Operation
Average	1E-5 fatalities/person/annum	100 mSv per 5 years
Peak	5E-5 fatalities/annum	50 mSv per year

The NNR, in reference [2], defines accident conditions as events with expected mean frequencies of occurrence of less than 0.01 per year. The Risk Assessment Report considers site personnel risk arising only from nuclear accidents. Radiological exposure due to normal operation is controlled in accordance with the KNPP radiation protection framework, and falls outside the scope of this report.

The site personnel risk assessment is used to identify the dominant nuclear accidents that contribute to site personnel risk. Insights from these accidents may be used to limit site personnel exposure during accident conditions.

Eskom acknowledges the NNR’s acceptance [48] of Eskom’s proposal, documented in reference [47], to apply a scaling (or risk increase) factor of 10.0 to the PC Cosyma results at 500 m, that are used to approximate the peak and average risks to site personnel from accidents related to releases from core damage, the spent fuel pool and spent fuel casks accidents. The aim of this proposal is to appropriately adjust the PC Cosyma results at 500 m to be more representative and bounding of the results expected at a distance of 100 m.

11.2 METHODOLOGY

11.2.1 PSA Standards and Requirements

In this assessment, all applicable requirements of NNR RD-0024 [2] were met. Where RD-0024 does not give instruction, ASME PSA standards were used as guides in the development of the PSA with the intent that the PSA should meet “Capability Category II” in areas impacting risk informed decisions.

Applicable ASME standards are:

- ASME/ANS RA-Sa-2009, Standard for Level 1 / Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications [5],
- ANSI/ANS-58.22 (2009), Low Power and Shutdown PRA Methodology (Draft) [6],
- ANS/ASME-58.24 (2010), Severe Accident Progression and Radiological Release (Level 2) PRA Methodology to Support Nuclear Installation Applications (Draft) [9],
- ANS/ASME-58.25 (2010), Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications (Draft) [10].

The specific methodologies used in producing the site personnel risk assessment are:

- PSA-P-T18-02, Site Personnel Risk Methodology, Revision R1 [14],
- PSA-P-T18-01, Accident Screening Process for Site Personnel Risk, Revision R1 [15].

11.2.2 Deterministic and Stochastic Risk

Mortality risk has two contributors:

- Deterministic (short-term) risk, where radioactive exposure may result in organ failure and eventually death due to the absorption of excessive energy by the organ; and
- Stochastic (long-term) risk, where radioactive exposure may cause cancer effects.

Stochastic risk is calculated from the whole body effective dose, which is the total whole body dose due to external irradiation (cloudshine, groundshine and beta skin radiation) for the duration of the exposure plus the 50 year committed effective dose due to inhalation (internal irradiation) of the isotopes of interest. Since the exposure is received over a relatively short period, a risk conversion factor of 5% per Sv is used in accordance with ICRP Publication 60. The Koeberg Site Personnel Risk Assessment [13] makes use of the Linear No-Threshold model, which assumes that no threshold dose exists below which no effects will be detectable and that a linear dose-response relationship applies.

Deterministic risk is calculated from dose received during the first 30 days after exposure by the most radiosensitive organs, *i.e*. bone marrow, lungs and gastro-intestinal tract, noting that these organs have different degrees of radiosensitivity. The thyroid has not been considered since damage can be reduced by the prompt administration of stable iodine and because, with adequate medical intervention, destruction of the thyroid should not be fatal.

11.2.3 Accidents that result in the exposure of site personnel

Accidents that could contribute significantly to site personnel risk are:

- Core damage accidents,
- Spent fuel pool cooling accidents,
- Accidents where core damage is avoided, but the fission product inventory of the primary system water leads to site personnel exposure,
- RCV break accidents,
- Fuel handling accidents, and
- Waste treatment accidents.
- Cask accidents.

Following a screening analysis, the following accident sequences were identified to be analysed in detail in the Koeberg Site Personnel Risk Assessment [13]:

- Fuel handling accidents
  - Fuel handling accident above the reactor,
  - Fuel handling accident inside containment but not above the reactor,
  - Fuel handling accident inside the spent fuel building during an outage and
  - Fuel handling accident inside the spent fuel building not during an outage.
- Waste treatment related accidents
  - Safety valve premature opening on tanks 1 TEP 001 BA and 2 TEP 008 BA.
- Chemical and volume control system breaks
  - RCV break inside the NAB with prompt intervention.
- Spent fuel pool accidents.
- Core damage accidents.
- Cask accidents.

As explained in [13] the whole body effective and organ doses due to inhalation were calculated using the LUDEP computer code in the Pre-SGR baseline PSA. For this analysis, the whole body effective and organ doses due to inhalation were calculated using ICRP Publication 68 dose conversion factors from Radiological Toolbox Version 3.0.0 [43]. This method gives similar results to the LUDEP calculations.

The fuel handling accidents, waste treatment related accidents, chemical and volume control system breaks risk assessment were re-evaluated in [13]. The site personnel risk assessment following core damage accidents has been re-evaluated in reference [12] whilst the spent fuel pool and cask accidents have been analysed in [8] and [34] respectively.

11.2.4 Definition of Peak and Average Site Personnel Risk

It is generally assumed that different site personnel would be exposed in the event of fuel handling, waste treatment and RCV break accidents and that all site personnel are exposed in the event of core damage and spent fuel pool accidents. The peak and average site personnel risks are determined as shown in Table 11-2 below.

Table 11-2: Determination of Overall Peak And Average Site Personnel Risks

Peak Site Personnel Risk	Average Site Personnel Risk
The highest risk value of either: • Fuel Handling Accidents in Containment / 3 • Fuel Handling Accidents in Spent Fuel Building / 3 • Waste Treatment Accidents • RCV Break Accidents +Core Damage Accidents +Spent Fuel Pool Accidents + Cask Accidents	Core Damage Accidents + Spent Fuel Pool Accidents + Cask Accidents + Fuel Handling Accidents in Containment × (3 × 100) / 1000 + Fuel Handling Accidents in Spent Fuel Building × (3 × 15) / 1000 + Waste Treatment Accidents / 1000 + RCV Break Accidents / 1000

For the peak risk, the highest risk value of the risk arising from fuel handling, waste treatment or RCV break accidents are added to the risks from core damage and spent fuel pool cooling failure accidents.

The average risk is the sum of all the accidents, taking into account the number of site personnel involved in each accident relative to the total number of on-site personnel, which is assumed to be 1000.

The overall peak and average site personnel risk are then expressed as a percentage of the NNR criteria for site personnel risk due to accidents shown in Table 11-1.

11.3 RESULTS

Table 11-3 summarises the results of the analysis contained within [13] and compares the peak and average site personnel risks to the NNR criteria for site personnel risk. The determination of the overall (total) peak and average site personnel risk results from all internal and external events are discussed and presented Chapter 12 of this report. The accidents that contribute significantly to site personnel risk are:

- Fuel Handling Accidents,
- Core Damage Accidents and
- Cask Accidents.

Table 11-3: Summary of Site Personnel Risks Due to Accidents [13]

Accident Type	**Peak Risk** (fatalities year^{-1})	**Average Risk** (fatalities person^{-1} year^{-1})
Fuel Handling in Containment Building	5.00E-06	1.50E-06
Fuel Handling in Spent Fuel Building	1.19E-06	5.36E-08
Waste Treatment	4.14E-09	4.14E-12
RCV Break	5.90E-08	5.90E-11
Core Damage Accidents[1] (includes internal events, internal fire, internal flood and aircraft crash events)	1.09E-06	1.09E-06
Spent Fuel Pool Accidents[1] (includes internal events only)	8.40E-08	8.40E-0
Cask Accidents[1] (includes internal events, aircraft crash and seismic events)	9.40E-07	8.94E-07
Total	**7.11E-06**	**3.62E-06**
RD-0024 Risk Limit	**5.00E-05**	**1.00E-05**
% of RD-0024 Risk Limit	**16.58%**	**35.46%**

[1] Scaling of the peak and average site personnel risks by a factor of 10 are included.
Note that the determination of the total peak and average site personnel risks used formulae presented in Table 11-2.

11.4 INSIGHTS AND CONCLUSIONS

As indicated in Table 11-2, the peak site personnel risk is determined by summing the risk arising from Fuel Handling in containment building, Core Damage and Spent Fuel Pool and Cask accidents. The average site personnel risk is calculated as the sum of all the accidents listed in Table 11-3, taking into account the number of site personnel involved in each accident relative to the total number of on-site personnel, assumed to be 1000[13].

The calculated peak and average site personnel risks shown in Table 11-3 are within allowable limits. However, RD-0024 [2] also states that the risk must be as low as reasonably achievable and so efforts to improve site personnel safety should continue.

In conclusion it was found that the largest contribution to the overall peak site personnel risk is due to fuel handling accidents in the containment building. The analysis of fuel handling accidents contains a number of bounding assumptions. The final results are therefore conservative. The main conservative assumptions are:

- Assembly unloading starts at 100 hours.
- Assembly re-loading starts at 9 days.
- All the fuel rods in a damaged assembly break releasing all the gap gases instantaneously into the water.
- There is no warning of impeding difficulties prior to the accident.
- Although a linear dose/risk relationship is assumed for the induction of stochastic effects, there is little evidence that low doses of radiation cause harmful effects.

The most conservative assumption is that all the fuel rods of the dropped assembly lose their integrity. World experience has shown that PWR fuel assemblies are relatively robust.

12. THE KOEBERG EXTERNAL HAZARD RISKS

12.1 INTRODUCTION

To the extent practical, the quality assurance programme for the assessment of external hazards is based on the IAEA guideline in [17]. More details can be found in [18]. An External Hazards screening report has been completed (see Reference [19]) using the guidance provided in ANSI/ANS methodology in [20].

Extreme external hazards have large uncertainties associated with them and so their assessment tend to contain more uncertainty and more conservatism than the assessment of internal events which are much better understood. Further, internal events are to a greater extent under management control and so are more easily addressed and the associated risk reduced. Thus, improving the risk from internal events has received considerable focus in the past resulting in plant modifications such as those implemented within the CP1 project. Based on the OE from the Fukushima accident, more focus has been placed on external hazards and changes to be implemented to reduce these risks.

The NNR in reference [52] raised concerns about a number of External Hazards such as coastal erosion and silting of the intake basin. These were generally addressed in the External Hazards Safety Re-assessment [53].

The External Hazards screening report [19] recommended further analysis on the following external hazards:

1. Seismic
2. Tsunamis
3. Extreme Winds
4. Oil Spills

The assessment of these hazards is largely based on expert judgment through comparison to similar plants. Note that Internal Fire and Internal Flooding are accounted as internal hazards as per the ASME PSA Standards, and are assessed accordingly in the PSA Level 1 [4]

12.1.1 Seismic

The seismic hazard assessment for the Stress Test included a review of the site hazard characteristics of the Koeberg site. These were based on original studies conducted by Dames & Moore in 1976. Subsequent studies conducted by the Council for GeoScience (CGS) in 2004 and by Paul C. Rizzo Associates in 2008 (using current methodologies) [39] have confirmed that a design peak ground acceleration (PGA) of 0.3 g defining the seismic ground motion for the design of Koeberg is appropriate. The value is the current licensed design basis for Koeberg and is considered to be adequately conservative. It represents the expected ground motion from an event of magnitude 7 from a postulated fault line occurring 8 km from the Koeberg site.

International experience shows that nuclear installations can cope with seismic events well in excess of their original design. In order to obtain a more realistic picture of the actual capacity of the plant, a seismic margin assessment (SMA) was performed on those SSCs that are essential to a safe and prolonged shutdown as identified in a needs analysis.

The purpose of the SMA was to obtain a more realistic indication of the capability of the plant and to identify any vulnerabilities and improvements to the plant’s capability to cope with a large seismic event. The functional requirements diagram was used to illustrate the plant’s capability to withstand a beyond-design-basis 0.5 g seismic event. The full report is contained in reference [21].

The seismic PSA for the metal casks for post-SGR[34] assessed the seismic risk for cask handling activities in the SFP building and casks being stored in the CSB and the results of which is summarized in Chapter 10. The results were used in the derivation of the total annual station risk for internal and External Events presented in Table 12-2

12.1.2 Tsunami

The assessment investigated the potential sources and risks of a tsunami at Koeberg NPP. The robustness of the plant’s design to mitigate a design-basis, as well as a beyond-design-basis tsunami was evaluated for different levels of flooding to identify vulnerabilities and associated cliff edge effects.

Based on current evidence, the intra-oceanic South Sandwich Island subduction zone (SSSZ) situated in the south-western Atlantic Ocean represents the closest source of high-frequency, high-intensity sub-sea seismicity which could threaten the west coast of southern Africa. The design basis tsunami for Koeberg, which is based on this, is appropriate and the height of the terrace provides adequate safety margin. The full report is contained in reference [22].

12.1.3 High Winds & Tornadoes

The assessment of high winds considered the effect on buildings and structures and the associated effects of missiles. A review of latest meteorological data confirmed that the design basis for Koeberg for winds was adequate and there is adequate safety margin between the design basis winds for safety and non-safety related buildings and the highest recorded wind speed. Wind-induced missiles were not considered in the design of Koeberg and no analysis exists on the impact of wind-borne missiles on the plant. The assessment identified that many non-safety related structures, particularly the mobile office units, would not survive a beyond-design-basis high wind and some could be expected to become the source of wind-borne missiles. The full report is contained in reference [23].

Whilst tornadoes were not considered in the design of Koeberg, the effects of tornado-generated missiles and tornado-induced atmospheric pressure changes from tornadoes were assessed. A probabilistic assessment was also performed. The functional requirements diagram was used to highlight the impact of tornadoes on the plant for the Enhanced Fujita Scale of tornadoes, up to EF3. Some vulnerabilities were identified and proposals made to address them in a manner similar to addressing the effects of high wind. The assessment showed that the plant is resilient to the tornadoes that have been experienced in the area, up to EF2. The full report is contained in reference [24].

12.1.4 Oil & Jellyfish Ingress

The assessment was made of oil and jellyfish ingress and the impact on the intake basin cooling water and the equipment that is on standby for mitigating such ingress. Proposals were made to address identified vulnerabilities, the main proposal being a modification to provide an alternative heat sink. The full reports are contained in references [25] and [26]. The sum of the CDF and FDF for oil spill events is 9.07E-07 / yr and for jellyfish ingress is 9.69E-07 / yr. These values were obtained by determining the conditional CDF upon loss of RRI (not due to flooding) from the Level 1 PSA model using the analysis case CD-NORMAL&FIRE-ALL.

This was calculated as follows:

The Fussell Vesely (FV) values for the different plant states for the loss of RRI (not due to flooding) were obtained from the results of the Level 1 analysis.

Plant State	FV
IEC-LRRI-APH	1.82E-03
IEC-LRRI-APL	4.10E-05
IEC-LRRI-R1D	5.51E-05
IEC-LRRI-R2D	3.94E-05
IEC-LRRI-RHD	6.19E-06
IEC-LRRI-SDH	1.12E-04
IEC-LRRI-SDL	1.79E-05
SUM	**2.09E-03**

The sum of the FV values was then multiplied by the total CDF for post-SGR to determine the CDF due to loss of RRI (no flood), i.e. 2.09E-03 X 6.78E-06 = 1.42E-08. The initiating event frequency for the loss of RRI from the PSA model is 1.83E-04. Therefore, the conditional CDP upon loss of RRI was calculated to be 1.42E-08 / 1.83E-04. = 7.75E-05.

For oil spills, a conservative assessment [60], has shown that with no prevention or mitigation, the likelihood of getting some oil in the intake basin is 2.6E-02 per year. A best estimate case assumes that half the oils spills are of sufficient magnitude to lead to total loss of SEC / RRI and the booms are effective in 1 out of 10 cases.

Therefore, for **oil spill events** due to shipping accidents, the CDF will be
7.75E-05 X 2.6E-02 X 0.5 X 0.9 = **9.07E-07/r.y.**

For jellyfish incursions, numerous incidents have occurred where jellyfish have entrained into the cooling water intake basin. However, they have not posed problems to the SEC system. Since the SEC / RRI system was never threatened, the likelihood of jellyfish leading to total loss of RRI / SEC can be calculated to be 0.5 in 36 years of operation. (Note that the value of 0.5 is obtained using Jeffries non-informative prior which assumes a gamma distribution with shape parameter equal to 0.5 and scale parameter equal to 0.0.) As in the case of oil spills, it is also assumed that the booms are effective in 1 out of 10 cases.

Therefore, the CDF for jellyfish incidents will be
7.75E-05 X 0.5/36 X 0.9 = **9.69E-07/r.y**

For each release category RCj, j=1, 2….8 of the Jellyfish events, its frequency was calculated as follows:

$$\left(\frac{Frequency\ of\ LRRI(no\ flood)\ for\ RCj}{Sum\ of\ Frequencies\ of\ LRRI\ (no\ flood)\ for\ RC1\ to\ RC\ 8}\right) X(CDF\ for\ the\ Jellyfish\ incidents)$$

Similarly, the frequency for each release category for oil spills events was calculated.

12.2 RESULTS

The figures below present the estimated risks including External Events.

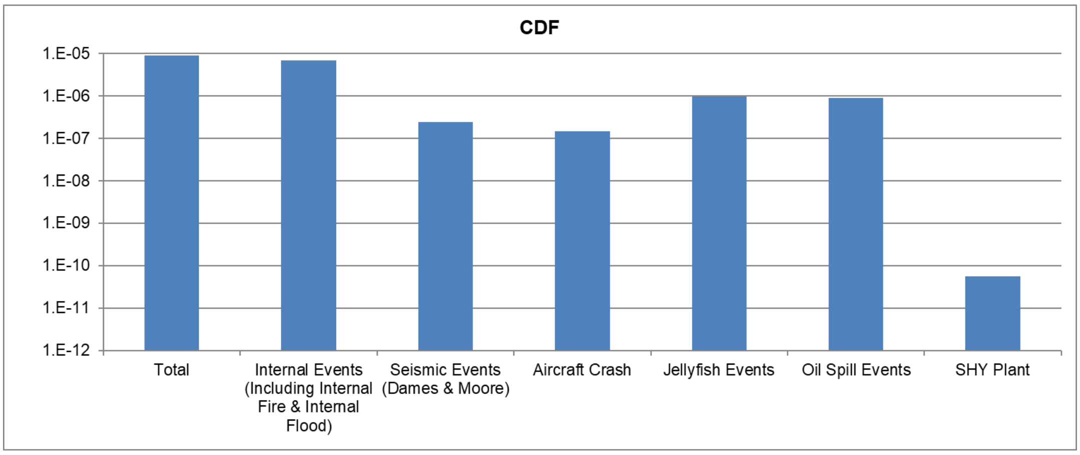

Figure 12-1: CDF Comparison for Internal and External Events

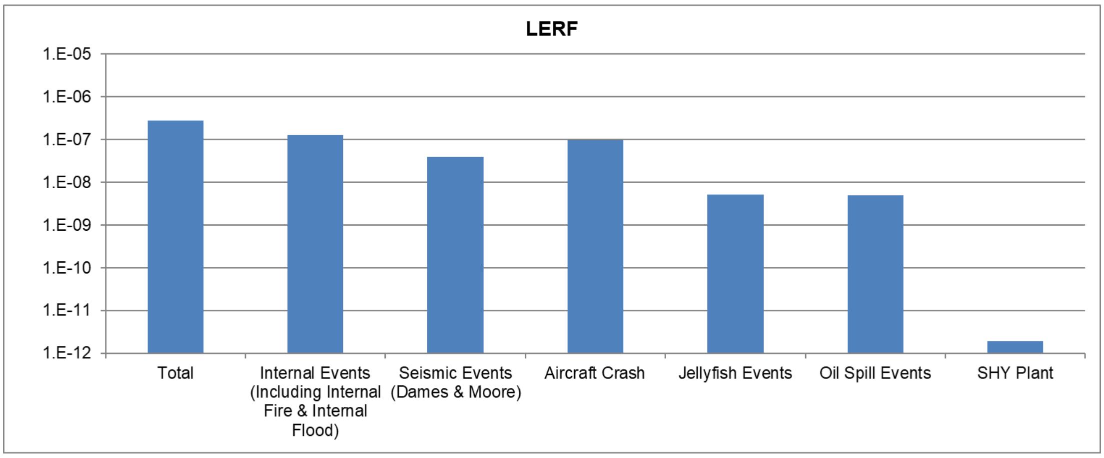

Figure 12-2: LERF Comparison for Internal and External Events

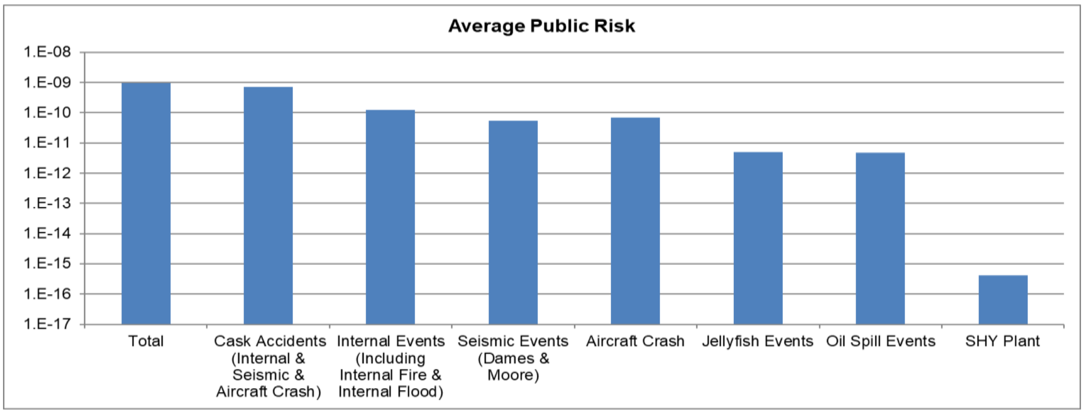

Figure 12-3: Average Public Risk Comparison of Internal and External Events

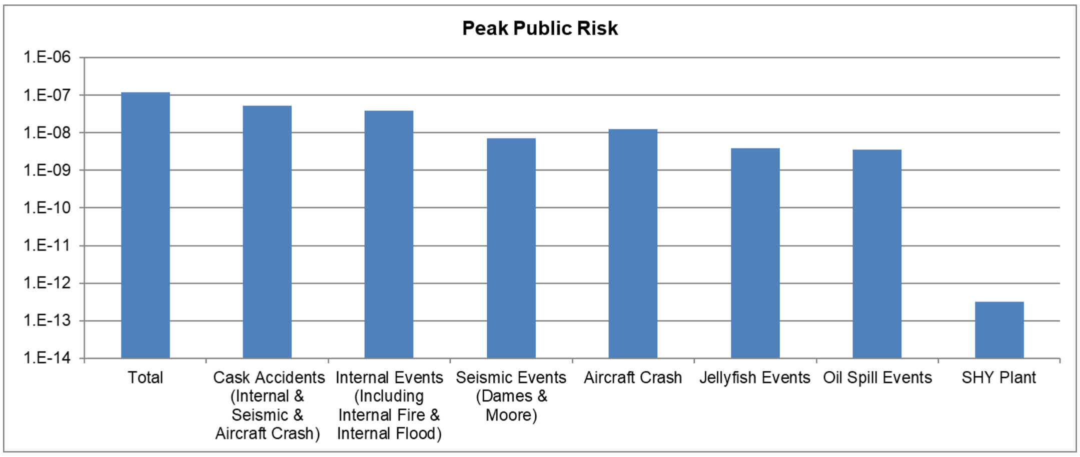

Figure 12-4: Peak Public Risk Comparison for Internal and External Events

Table 12-1 Internal and External Events Risk Results by Release Category

Release Category	Release Category Frequency[a] / yr	Internal Events (Including Internal Fire & Internal Flood) / yr	Seismic Events[b] (Dames and Moore)/ yr	Aircraft Crash[c] / yr	Jellyfish Events[d] / yr	Oil Spill Events[e] / yr	SHY Plant[f] / yr
RC-1 (No Containment Failure)	7.88E-06	5.94E-06	1.80E-07	0.00E+00	9.08E-07	8.50E-07	4.29E-11
RC-2 (LERF)	2.35E-08	1.90E-08	0.00E+00	0.00E+00	2.33E-09	2.18E-09	2.18E-13
RC-3 (LERF)	3.45E-08	3.42E-08	0.00E+00	0.00E+00	1.50E-10	1.40E-10	1.01E-14
RC-4 (LERF)	1.99E-07	7.60E-08	2.00E-08	9.80E-08	2.78E-09	2.60E-09	1.69E-12
RC-5 (Late Containment Failure)	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RC-6 (Late Containment Failure)	4.53E-07	4.53E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.62E-12
RC-7 (Basemat Failure)	1.18E-07	7.63E-08	0.00E+00	0.00E+00	2.14E-08	2.00E-08	1.22E-13
RC-8 (Basemat Failure)	2.31E-07	1.64E-07	0.00E+00	0.00E+00	3.48E-08	3.25E-08	1.34E-12
RC-9 (SFP Outage)[g]	5.95E-09	1.99E-09	1.16E-09	2.80E-09	0.00E+00	0.00E+00	0.00E+00
RC-10 (SFP Normal) [g]	1.13E-07	4.83E-08	1.88E-08	4.54E-08	0.00E+00	0.00E+00	0.00E+00
RC-11 (Dual Unit - Outage)[h]	1.16E-09	0.00E+00	1.16E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RC-12 (Dual Unit - Normal)[i]	1.88E-08	0.00E+00	1.88E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total (CDF + FDF)	N/A	6.81E-06	2.40E-07	1.46E-07	9.69E-07	9.07E-07	5.59E-11
Total (LERF)	N/A	1.29E-07	4.00E-08	9.80E-08	5.26E-09	4.92E-09	1.92E-12
Total (Average Public Risk)	N/A	1.24E-10	5.51E-11	7.09E-11	5.06E-12	4.73E-12	4.19E-16
Total (Peak Public Risk)	N/A	3.83E-08	7.14E-09	1.23E-08	3.82E-09	3.57E-09	3.20E-13

[a] Obtained from the PSA Level 1 Post-SGR Model [45].
[b] The frequency of an earthquake of magnitude 0.3 g is 1E-06 [40]. Section 9 of the External Events Risks report [18] explains that the Koeberg seismic CDF would be approximately one fifth of SSE frequency. Therefore, the best estimate seismic CDF using the Dames and Moore curve is 1E-06 / 5 = 2E-07 / yr. Only RC1 and RC4 will be impacted by a seismic event and were allocated 90 % and 10 % of the seismic CDF respectively. These percentages were used because the containment is considered to be robust and the containment building is not expected to fail in seismic scenarios.
[c] The frequencies were taken from Koeberg Aircraft Crash Risk Assessment study [31]. The frequencies in RC-4 and RC-9 are based on the time period (94.2 %) when fuel assemblies will be present inside containment and the time period (5.8 %) during outage when fuel assemblies will be present in the SFP.
[d] and [e] The calculation of jellyfish and oil spill events is explained in 12.1.4.

[f] The value of 5E-11 / yr for the total CDF and FDF was obtained from [30].
[g] Based on the seismic event frequency for RC-4 (i.e., early containment failure) and has been adjusted to consider the time period (94.2% of the time) when fuel assemblies are present inside containment and the time period during outage when fuel assemblies are present in the Spent Fuel Building (5.8% of the time).
[h] The consequence (dose) of an early containment failure and SFP uncovery due to a seismic event during an outage is determined as the sum of the consequences from RC-4, RC-9 and RC-10
[i] The consequence (dose) of an early containment failure and SFP uncovery due to a seismic event during an outage is determined as the sum of the consequences from RC-4, RC-9 and RC-10.

Table 12-2: Total Annual Post-SGR Station Risk (for Internal and External Events)

	Total Annual Station Risk (i.e., 2 units)		
	External Events + Internal Events	**Station Internal Events (Including Internal Fire and Internal Flood)**	**Station External Events**
Peak Public Risk(fatalities per year)	1.17E-07	6.31E-08	5.43E-08
Average Public Risk (fatalities person^{-1} year^{-1}) Using 2008 National Population	1.03E-09	7.02E-10	3.31E-10
Average Public Risk (fatalities person^{-1} year^{-1}) Using 2011 National Population	9.71E-10	6.60E-10	3.11E-10
Average Public Risk (fatalities person^{-1} year^{-1}) Using 2025 National Population	2.37E-09	1.61E-09	7.60E-10
Peak Site Personnel Risk (fatalities per year)	7.56E-06	6.31E-06	1.25E-06
Average Site Personnel Risk (fatalities person^{-1} year^{-1})	4.07E-06	2.82E-06	1.25E-06

13. RISK CRITERIA COMPLIANCE DEMONSTRATION

13.1 COMPLIANCE TO NNR RISK CRITERIA

One aim of this report is to assess the off-site radiological risk due to accidental releases of radioactive materials at Koeberg NPP and to compare those risks to the criteria of the National Nuclear Regulator (NNR) as prescribed by RD-0024 [2]. The risks for KNPP from all accidents and provided as percentages of the NNR criteria are summarised in Table 13-1 below. These risks consider the internal and external events affecting the reactors, the spent fuel pools, the casks, and fuel handlings.

Table 13-1: Total Annual Post-SGR Risks for KNPP (for Internal and External Events)

Criteria	Annual Risk (Station)	% of NNR Criteria
Peak Public Risk (fatalities per year)	1.17E-07	2.35%
Average Public Risk (fatalities person^{-1} year^{-1}) Using 2008 national population	1.03E-09	10.32%
Average Public Risk (fatalities person^{-1} year^{-1}) Using 2011 national population	9.71E-10	9.71%
Average Public Risk (fatalities person^{-1} year^{-1}) Using projected population for 2025	2.37E-09	23.75%
Peak Site Personnel Risk (fatalities per year)	7.56E-06	15.12%
Average Site Personnel Risk (fatalities person^{-1} year^{-1})	4.07E-06	40.66%

*Scaling of the peak and average site personnel risks by a factor of 10 are included.

Several conservative assumptions were made in the compilation of this assessment, which implies that the final results are also conservative. For example:

- No emergency plan countermeasures, other than the food ban, have been assumed. Therefore, the public are assumed to be outdoors at the time of the accident and to remain outdoors for the duration of the exposure period. The shielding effects of clothing and building structures are not taken into consideration. During this period, we further assume that they have no access to water for decontamination.
- The accident occurs at the end of the cycle consequently maximising the initial core inventory.
- In the event of a fuel handling accident all the fuel rods in an assembly break releasing all of the gap gases instantaneously into the water.

To ensure that no single accident results in an unacceptably high number of deaths, the NNR have included a bias against larger accidents. This is illustrated in Figure 13-1.

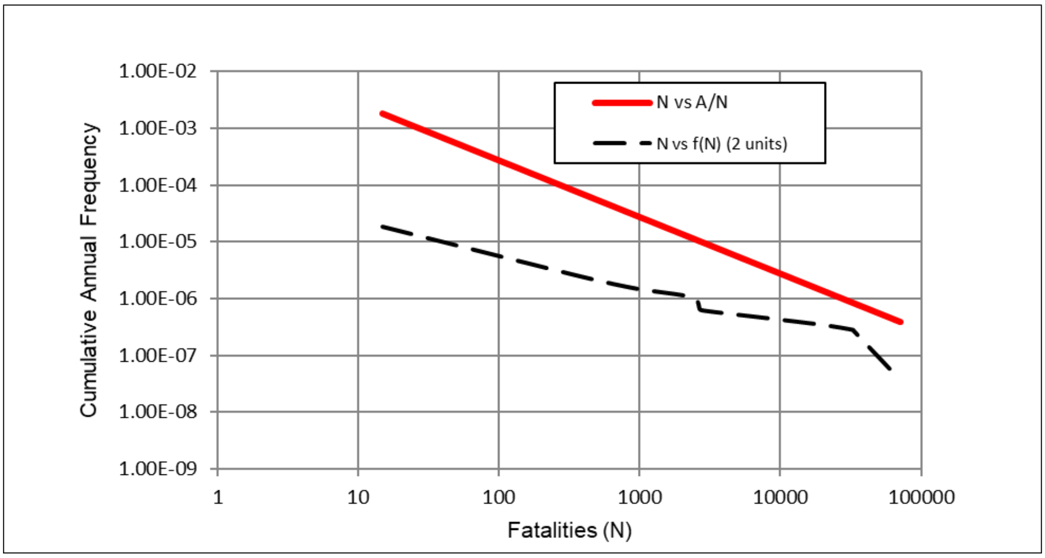

Figure 13-1: Bias against Large Accidents (with External Events)

Figure 13-2 below indicates how the baseline peak public risk including risk associated with cask activities has changed over time, relative to the peak public risk limit defined by the NNR in RD-0024 [2].

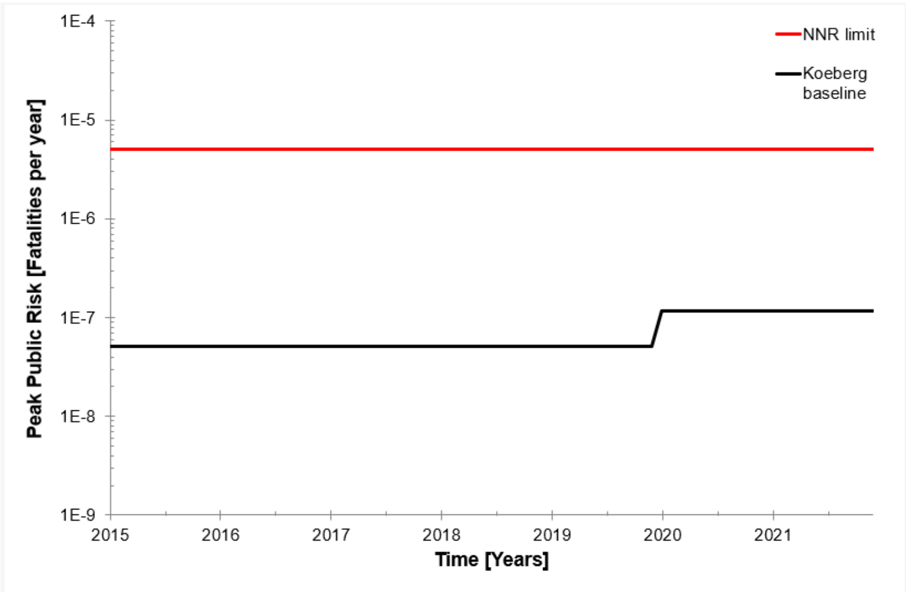

Figure 13-2: Baseline Peak Public Risk (for Internal and External Events)

This study therefore indicates that all the criteria set by the National Nuclear Regulator for public risks and site personnel risks in RD-0024 [2] have been met. The increase in the peak public risk seen in 2020 and 2021 is due to cask activities.

13.2 COMPLIANCE TO INTERNATIONAL RISK CRITERIA

Figure 13-3 and Figure 13-4 below demonstrate compliance to the IAEA risk criteria for CDF and LERF given in paragraph 27 of reference [27].

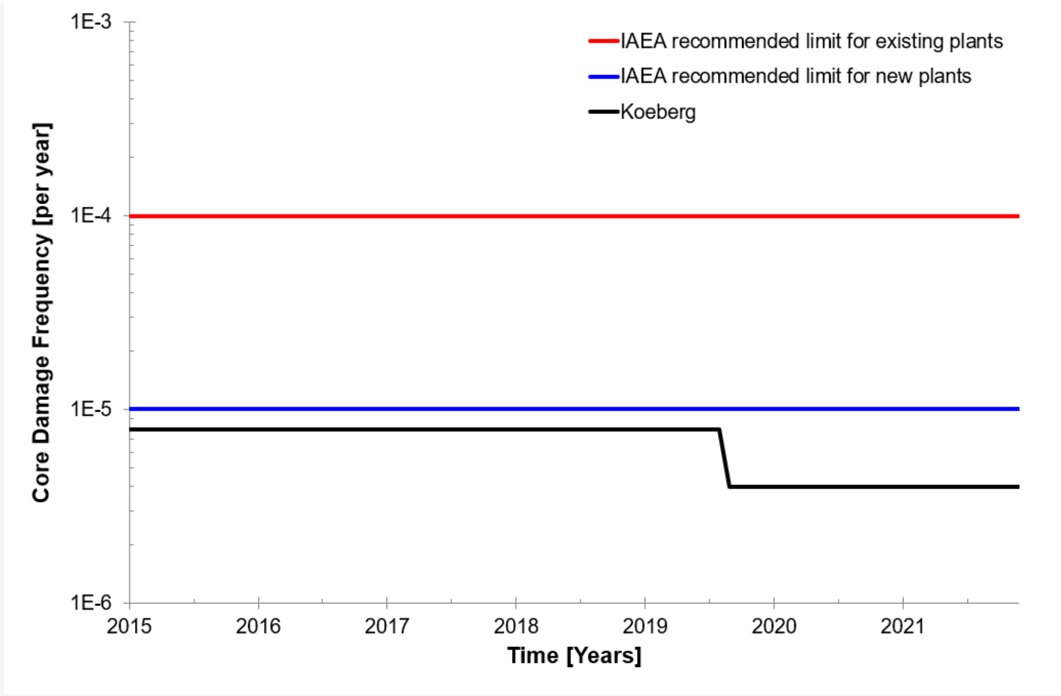

Figure 13-3: Baseline Core Damage Frequency (for Internal Events and Aircraft Crash)

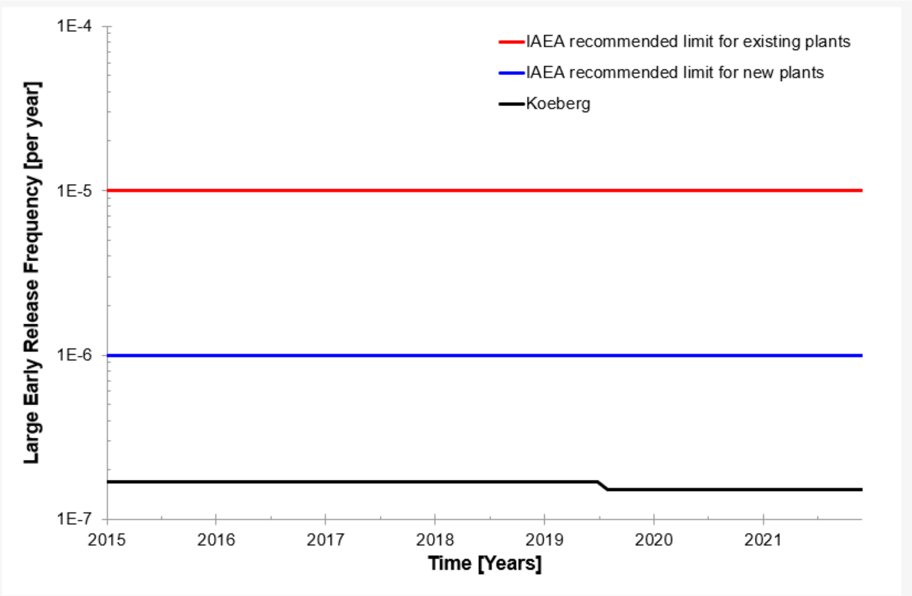

Figure 13-4: Baseline Large Early Release Frequency (for Internal Events and Aircraft Crash)

14. PRECURSOR ANALYSIS

Figure 14-1 below presents the significant accident precursors since 2002. A comprehensive analysis of the accident precursors is given in the Risk Profile report (reference [28]).

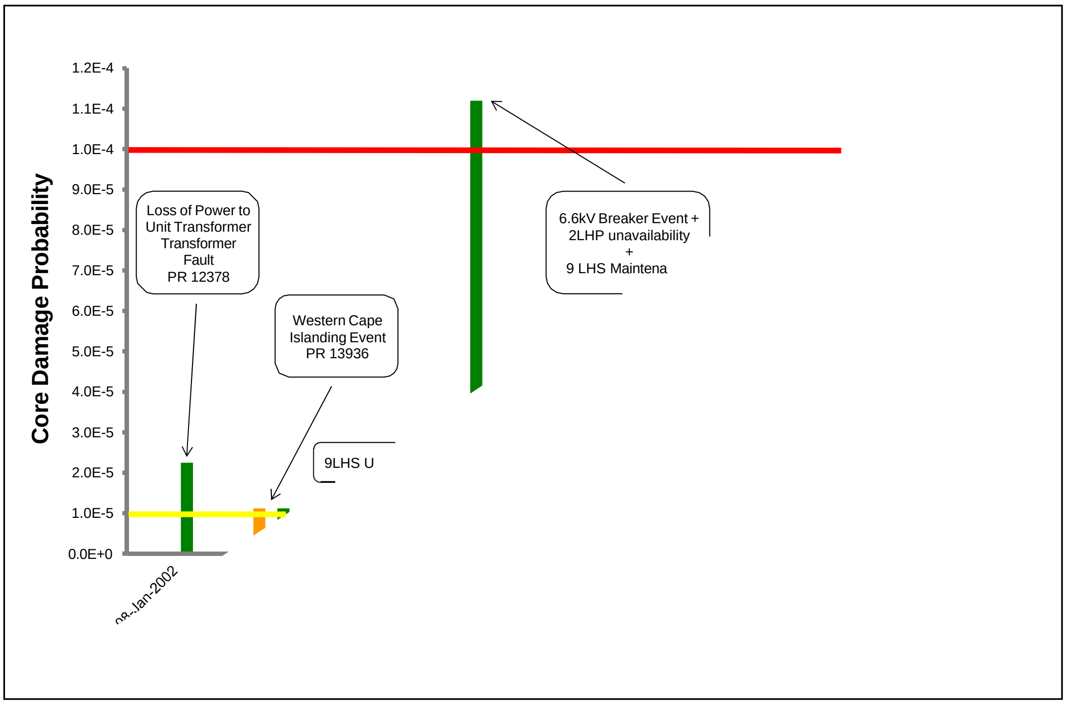

Figure 14-1: Significant Accident Precursors since 2002.

Table 14-1 presents the risk significant events from 2002 until the end of December 2020

Table 14-1: Risk Significant Events since 2002

Date of event	Unit	Event	ΔCDP	ΔLERP	Reference
Dec 2005	1	1 RIS 001 / 002 PO inoperable due to low RRI flow to the motor coolers	2.50E-04	7.50E-06	PR 25939
Aug 2002	2	Common mode failure of 6.6 kV breakers (Together with the 2LHP unavailability and the 9LHS in maintenance)	1.12E-04	1.79E-05	Ops Log
Mar 2012	2	Emergency Seal Injection inoperability (Inadequate ventilation on LLY switchboard)	4.00E-05	2.46E-07	PN 67406
Jan 2002	2	Loss of off site power (Loss of power to unit transformer)	2.25E-05	3.60E-06	PR 12 378
Jul 2002	Both	Loss of off site power (Western Cape islanding event)	1.12E-05	1.79E-06	PR 13 936
Mar 2012	1	Electrical Building Fire Doors (Fire door damaged and others left open)	4.30E-06	2.65E-08	KORC 2911
Aug 2002	Both	9LHS Unavailable (The Unit 9 Diesel Generator unavailable for an extended period))	1.37E-06	2.19E-07	Ops Log
Feb 2006	Both	Loss of off site power (Loss of Grid Event)	1.07E-06	9.23E-08	PR 26 629
Feb 2006	Both	Loss of off site power (Loss of Grid Event)	1.07E-06	9.23E-08	PR 26 629
Feb 2006	Both	Loss of off site power (Loss of Grid Event)	1.07E-06	9.23E-08	PR 26 722
Nov 2005	Both	Loss of off site power (Loss of Grid Event)	1.07E-06	3.41E-08	LOSP Report

The conclusion reached from this historical review is that the plant continues to be operated safely. The review highlighted that:

- Since August 2002 all accident precursors have been below the cut-off value of 1E-05 increase in the CDP. Below this value, issues are not considered “Risk Significant”.
- Events associated with the reliability of electrical supplies dominate the precursor analysis, highlighting the importance to safety of the grid and the alternate AC power sources.

15. DEVELOPMENT TEAM

The following people were involved in the development of this document:

- Danette Dreyer
- Emmanuel Lamprecht
- Guy Blaise Dongmo
- Shivani Fagan
- Thavy Krishna
- Yolanda Combrink

APPENDIX A: REVISION INFORMATION

Date	Rev.	Compiler	Remarks
June 2022	10	D Dreyer / S Fagan	The Risk Assessment Report was updated to reflect the Koeberg PSA following the replacement of the steam generators. A full review of the report was performed. Inputs from the Framatome reports for SGR were included in Revision 10. SPRA improvements were also included in this revision.
June 2015	9	LJ Perryman	In Section 3.2, changed: • PSA-R-T15-01, Spent Fuel Pool PSA from Revision 6 to Revision 7 • PSA-R-T16-01, The Koeberg Level 3 PSA Study from Revision 16 to Revision 17 • PSA-R-T18-01, The Koeberg Site Personnel Risk Assessment from Revision 16 to Revision 17 No technical changes made.
May 2015	8	LJ Perryman	Changes resulted from NNR commitments in K-21122-E and K-21123-E; a main technical change being the revision of the population data used in the off-site consequence assessments. This primarily impacted the off-site consequence results for the Level 3 and Spent Fuel Pool PSA studies.
June 2013	7	HL Bosman	The Risk Assessment report was updated to reflect changes to the Koeberg PSA.