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1 INTRODUCTION

In terms of the provisions of section 21 of the National Nuclear Regulator Act 1999, Act No. 47 of 1999 [7] (hereinafter referred to as the NNRA), the siting, construction, operation, decontamination or decommissioning of any nuclear installation as defined in section 1(xviii) of the NNRA must be authorised by way of a nuclear installation license granted by the National Nuclear Regulator (NNR).

Application for the granting of a nuclear installation licence must be made to the Chief Executive Officer of the NNR in the prescribed format and the *applicant* must provide such information as the Board of Directors of the NNR may require.

The legislation authorises the inclusion in the nuclear installation licence of any conditions deemed necessary by the NNR to ensure the protection of persons, property and the environment against nuclear damage or for the rehabilitation of the *site*.

The principal requirements that must be met to ensure safety in all nuclear installations are presented in the Regulations on Safety Standards and Regulatory Practices published as Regulation No. R388 dated 28 April 2006 (hereinafter referred to as the RSRP) [1]. The *Basic Licensing Requirements (BLR)* for the Pebble Bed Modular Reactor (PBMR), as presented in this Requirements Document (RD), are based on and established to fulfil these requirements.

2 PURPOSE

This RD establishes *Basic Licensing Requirements* of the NNR for the PBMR *facility* in line with the NNRA [7] and RSRP [1] and defines principal stages of the licensing process.

3 OBJECTIVES

The objectives of this RD are:

- In Section 5 to list and define the Licensing Process and Licensing Stages for the PBMR.
- In Section 6 to adopt the principal radiation protection and nuclear safety requirements as formulated in Section 3 of [1] for their application to the PBMR.
- In Section 7 to define the *BLR* for the PBMR (based on the principal radiation protection and nuclear safety requirements) that inter alia include the dose and risk limits applicable to the PBMR.
- In Section 8 to specify the processes which the *applicant*/licensee and the designer of the facility must undertake to demonstrate compliance with the *BLR*.

4 SCOPE

The document is applicable to the design of the PBMR that is planned, under construction, in operation or being decommissioned under a nuclear installation application or licence as per the NNRA.

The *applicant*/licensee responsible for design, construction and operation and the designer of the PBMR are required to develop, implement and maintain a design fulfilling the requirements of this RD.

Additional licensing requirements and regulatory guidelines relevant to the PBMR covering specific areas (e.g. Quality and Safety Management, Fuel Qualification, etc.) are stipulated in additional Regulatory Requirements and Guidance Documents as identified by the NNR.

5 LICENSING STAGES FOR THE PBMR

(1) In accordance with the provisions of section 21 of the NNRA:
"Any person wishing to site, construct, operate, decontaminate or decommission a nuclear installation may apply in the prescribed format to the chief executive officer for a nuclear installation licence and must furnish such information as the Board requires".

The above therefore represents the logical licensing stages that are applicable to any nuclear installation. The *applicant* may however choose to combine individual stages; such combination of stages may be approved by the NNR subject to the *applicant* ensuring that all the necessary safety documentation relevant to the combined stages has been submitted to the NNR.

(2) The combinations of licensing stages need to be established with a view to streamlining and scheduling of the licensing process. Allowance must be made for assessments that may prove to be time-consuming.

(3) The *applicant* is required to produce a *safety case* for each licensing stage or combination of licensing stages of the PBMR.

(4) Based on the applicant's proposal for combination of licensing stages the NNR may impose hold or witness points. The applicant must not proceed beyond an imposed hold or witness point without prior NNR approval.

6 PRINCIPAL RADIATION PROTECTION AND NUCLEAR SAFETY REQUIREMENTS

The Principal Radiation Protection and Nuclear Safety Requirements formulated in section 3 of the RSRP [1] form the basis for the stipulation of the *BLR* for the PBMR which are given in Section 7 of this RD. Although all the relevant requirements formulated in [1] are applicable to the design, construction, operation and subsequent decommissioning of the PBMR, the Principal Radiation Protection and Nuclear Safety Requirements of Section 3 of [1] are further elaborated in terms of their particular application to the PBMR:

(5) The PBMR must be designed, constructed, commissioned, operated, maintained and decommissioned based on *good engineering practice*. Based on the Safety Classification of the Structures, Systems and Components (SSC) and the overall integrated design performance, application of up to the best available practice including qualification must be adequately ensured.

(6) In line with the Principal Radiation Protection and Nuclear Safety Requirements of [1], the principles of Defence-in-Depth (DiD) must be applied to the PBMR in a manner consistent with the DiD processes described in the appropriate international safety standards and related documents (e.g. documents produced by the IAEA) so that there are multiple layers of PBMR Functions provided by the SSC, and procedures (or a combination thereof), to ensure that the *Fundamental Safety Functions (FSF)* of Heat Removal / Reactivity Control / Confinement of Radioactivity are met. Event prevention and event mitigation are natural consequences of the DiD principle. The application of the DiD Principle to the design and operation of the PBMR is further elaborated in Appendix B.

(7) In line with section 3.2 of [1] the ALARA (As Low As Reasonably Achievable) principle must be adopted in all stages of PBMR (siting, design, construction, operation, maintenance, decommissioning) in a manner consistent with the ALARA processes described in the appropriate international guides (e.g. reports produced by the *ICRP*). The application of the ALARA principle is required for selection of design and operational features that provide control of radiological doses and thus the optimum level of safety in terms of radiological risks. Specific guidance for application of the ALARA principle to the design and operation of the PBMR is given in Appendix C.

(8) In line with section 3.1 and 3.3 of [1] the proposed *facility* must comply with the dose and risk limits as defined in Annexure 2 and 3 of [1] respectively during operation by consideration of measures to control the risk of nuclear damage to individuals determined on the basis of *prior safety assessment*.

Note: The additional Principal Radiation Protection and Nuclear Safety Requirements of Section 3 of [1] 'quality management (3.10 of [1])' and 'safety culture' (3.5 of [1]), as well as accident management and emergency planning, emergency preparedness and emergency response (3.8 of [1]), are detailed in additional NNR Requirements Documents [2], [3], [4].

7 BASIC LICENSING REQUIREMENTS FOR THE PBMR

For licensing of the PBMR, it is required to demonstrate that the following *BLR* derived from the Principal Radiation Protection and Nuclear Safety Requirements of Section 6 of this RD are met. Additional licensing requirements and recommendations covering specific areas are stipulated in [1] and in the other Regulatory Requirements and Guidance Documents developed by the NNR.

Normal Operation and *initiating events* (IE) either singly or in combination are grouped into three categories (categories A, B and C) which are defined in terms of annual frequency of occurrence. Combinations of *IEs* or *IEs* defined to cover several *IEs* are referred to as *Postulated Initiating Events* (PIE). The frequency of events either singly or as combined events must be assessed accordingly and allocated to the appropriate category. For each of the three categories, safety requirements and numerical dose and risk limits are stipulated.

7.1 Selection of Events

The *BLR* apply to *IE* or combinations of *IE*, which lead or could potentially lead to exposure of the plant personnel and/or members of the public.

(9) For licensing it must be demonstrated that the *BLR* are met during the life cycle of the PBMR. For this purpose an enveloping set of *PIE* needs to be selected and categorised based on a comprehensive list of *IE*. The approach on *IE* and *PIE* that must be adopted is defined in section 8 of this RD.

(10) The categorisation of the events must be done considering the uncertainty of the event frequency such that events with substantial uncertainty over the predicted event frequency must be categorised into the higher frequency category.

7.2 Category A

Category A comprises exposures from *Normal Operation* as well as potential exposures from events to be expected during operation and defined as *Anticipated Operational Occurrences* (AOO). *AOOs* are *IEs* which occur with a frequency of more than one in one hundred years ($\geq 10^{-2}$ y^{-1}). *Normal Operation* includes exposures resulting from minor mishaps and misjudgements during commissioning, operation, maintenance and decommissioning that do not need individual event identification.

(11) A representative set of *AOOs* must be defined and considered. *AOOs* must not lead to category B events without occurrence of additional control failures of any type.

7.2.1 Radiation Dose to Occupationally Exposed Plant Personnel

(12) In respect of the dose to plant personnel, the following criteria must be applied in consideration of both design and all stages and modes of operation of the *facility*:

(i) Compliance with the *design dose limit* of 20 mSv in any single year, applicable to occupationally exposed personnel in the plant[1].
(ii) In addition, all radiation doses must be optimised by the application of the ALARA principle and ALARA targets for individual and collective dose must be defined.
(iii) The design provisions must be part of the DiD application.
(iv) Compliance with operational dose limits (effective doses and equivalent doses) as defined in [1] for plant personnel, apprentices, students and pregnant women.

7.2.2 Radiation Dose to Members of the Public and Non-occupationally Exposed Plant Personnel

(13) In respect of the dose to members of the public, the following criteria must be applied in consideration of both design and all stages and modes of operation of the *facility*:
(i) Compliance with the annual individual *design dose limit* of 250 μSv for non-occupationally exposed plant personnel as well as for an average member of the most exposed "critical group" [1] considering normal operation conditions as well as a representative set of *AOOs*.
(ii) Dose constraints must be defined for doses resulting from NO and AOO.
(iii) In addition, all radiation doses must be optimised by the application of the ALARA principle and the ALARA target for the individual dose resulting from NO be at a trivial level of about 10 μSv per year.
(iv) The design provisions must be part of the DiD application.

7.3 Category B

Category B events are those which potentially lead to exposure and which could occur with a frequency of less than one in one hundred years ($<10^{-2}$ y^{-1}) and more than one in one million years ($\geq 10^{-6}$ y^{-1}). The category B events lead to consequences and conditions that are considered for the *design basis* of the PBMR but are beyond the range of category A. Category B events are not expected to occur during the life cycle of the *facility*. In the range of Category B staged dose constraints must be defined as part of the deterministic framework.

(14) Category B events must not lead to consequences more severe than the category B criteria without occurrence of failures of active or passive safety functions additional to those already assumed as part or consequence of the category B events.

7.3.1 Radiation Dose to Occupationally Exposed Plant Personnel

(15) In respect of the dose to occupationally exposed plant personnel, the following criteria must be applied in consideration of both design and all stages and modes of operation of the *facility*:

[1] For the operational dose limit reference [1] applies.

(i) Compliance with the radiation *design dose limit* of 50 mSv for occupationally exposed plant personnel per category B event.
(ii) Dose constraints and ALARA targets to be defined for the individual dose resulting from category B events must be staged according to the frequency of category B events commensurate with their risk contribution for plant personnel.
(iii) In addition all radiation doses must be optimised by the application of the ALARA principle and the design provisions for category B events must be part of the DiD application.

Note: For *recovery actions* after category B events, the category A dose limits apply. The doses to plant personnel already received during the category B event must be taken into account in assessing the remaining dose that may be received during recovery.

7.3.2 Radiation Dose to Members of the Public and Non-occupationally Exposed Plant Personnel

(16) In respect of the dose to members of the public, the following criteria must be applied in consideration of both design and all stages and modes of operation of the *facility*:

(i) Compliance with the total individual *design dose limit* of 50 mSv accumulated as a consequence of a category B event for the dose to non-occupationally exposed plant personnel and visitors and an average member of the most exposed "critical group" [1].
(ii) Dose constraints must be defined for the individual dose resulting from category B events and must be staged in the frequency range of category B. The frequency range of category B events may be subdivided for staging of the dose constraints and the design criteria.
(iii) In addition, all radiation doses must be optimised by the application of the ALARA principle and the design provisions for category B events must be part of the DiD application.
(iv) ALARA targets are to be defined for the individual dose resulting from category B events and must be staged in the frequency range of category B. The ALARA targets must be consistent with the defined dose constraints and start with a value of not more than 250 µSv for a frequency of 10^{-2} y^{-1}.
(v) If the dose to any member of the public resulting from a category B event can potentially exceed an effective dose of 1 mSv, emergency measures consistent with subsection 8.6 have to be implemented to keep the resulting dose ALARA.

7.4 Category C

Category C events are all possible events that potentially could lead to exposure, including those which are demonstrated to be beyond the range of category B events. As such, category C events include category A and B events as well as those events that occur with an annual frequency of less than 10^{-6} per year (*beyond category B events*).

(17) In the range of beyond category B, events must be considered as part of the design assessment where there are significant uncertainties on the event frequencies and related probability values. This will include consideration of the required provisions against accidents with larger consequences. Justification must be given for the exclusion of events from the design assessment.

7.4.1 Limitation of Risk to Occupationally Exposed Plant Personnel

(18) In respect of risk limitation to plant personnel, the risk criteria specified in [1] must be applied in consideration of both design and all stages and modes of operation of the *facility*:

(i) 5×10^{-5} peak individual risk per year,
(ii) 1×10^{-5} average risk per year
(iii) Provisions against *beyond category B events* must be considered so that no *cliff edge effects* are to be expected to plant personnel. This must take emergency response actions into account. In addition, risks must be optimised by the application of the ALARA principle.

Note: These risk limits are to be applied to workers occupationally exposed at the *facility*. Where workers are expected to be occupationally exposed due to more than one *authorized action* adequate controls should be in place to ensure that no individual exceeds the risk limit.

7.4.2 Risk to Members of the Public and Non-occupationally Exposed Plant Personnel

(19) In respect of risk limitation to members of the public and non-occupationally exposed plant personnel and visitors, the risk criteria specified in [1] must be applied in consideration of both design and all stages and modes of operation of the *facility*

(i) 5×10^{-6} peak individual risk per year due to all nuclear installations in South Africa,
(ii) 1×10^{-8} average individual risk per year per *facility,*
(iii) Provisions against *beyond category B events* must be considered so that no *cliff edge effects* are to be expected. All sources of exposure potentially leading to risk must be included. In addition, risks must be optimised by the application of the ALARA principle.

(20) For limitation of risks resulting from larger accidents the risk aversion criterion according Appendix B of [2] must be applied.

7.5 Summary

The Basic Licensing Requirements for the PBMR are summarised in the following table:

INITIATING EVENT FREQUENCY	SAFETY REQUIREMENTS	LIMITS
Category A Category A comprises *Normal Operation* as well as *Anticipated Operational Occurrences* (AOO) which are events that potentially lead to exposure and which could occur with a frequency of more than one in one hundred years ($\geq 10^{-2}$ y^{-1}).	The design must be such to ensure that under anticipated conditions or occurrences of *normal operation*, there is no radiation dose to the plant personnel, non-occupationally exposed worker and visitors at the *site* and members of the public above the category A limits. Normal operation exposure must consider exposures from minor mishaps and misjudgement during operation, maintenance and decommissioning, that do not need individual event identification. For dose calculation a representative set of *AOO* needs to be considered. The design provisions for category A must be part of the DiD application. In addition, all radiation doses must be kept ALARA and individual dose from NO at a trivial level of about 10 μSv for the public.	Occupationally Exposed Plant Personnel: - individual *design dose limit* of 20 mSv in any single year - limits of equivalent dose to organs and tissues according to [1] and Members of the Public (critical group) and non-occupationally exposed plant personnel: individual design dose limit of 250 μSv in any single year considering all actions authorised by a nuclear installation licence as well as a representative set of AOOs
Category B Category B events are those which potentially lead to exposure and which could occur with a frequency of less than one in one hundred years ($< 10^{-2}$ y^{-1}) and more than one in one million years ($\geq 10^{-6}$ y^{-1}).	The design must be such to prevent and mitigate potential failures and to withstand externally or internally originating events which could give rise to radiation doses to plant personnel, members of the public, non-occupationally exposed individuals and visitors in excess of the category B limits. The analysis performed to demonstrate compliance with this requirement must be deterministic and demonstrably conservative with respect to the event frequencies and the resulting radiation doses. The range of category B events may be subdivided for staging of the dose constraints and the design criteria. In addition radiation doses must be kept ALARA. ALARA targets must be defined adjusted to the event frequency and starting with a value of not more than 250 μSv for members of the public. Regarding the dose to the public, emergency measures have to be taken into account if the expected dose to the public can exceed 1 mSv per event to keep the resulting dose ALARA. The provisions for category B events must be part of the DiD application.	Occupationally Exposed Plant Personnel: - 50 mSv individual *design dose limit* per event and Members of the Public (critical group) and non-occupationally exposed individuals and visitors at *the site*: - 50 mSv individual design dose limit for the total accumulated dose per event

Category C Category C events are all events that potentially can lead to exposure. As such, category C events covers the frequency range of category A and category B as well as beyond category B events which occur with an annual frequency of less than 10^{-6} ($<10^{-6}y^{-1}$) and which are still relevant for the risk criteria defined in [1]	The design must be demonstrated to respect the Risk Limits for plant personnel and members of the public [2]. The analysis performed to demonstrate compliance with this requirement may use *best estimate data* provided it is supported by an appropriate uncertainty and sensitivity analysis. The analysis must also demonstrate provisions against very low frequency events in the range below 10^{-6} per year. In addition risks associated with these events must be demonstrated to be ALARA. *The risk aversion criterion must be applied for events that are the main risk contributors.*	Limitation of risk to the values set by the risk criteria: Occupationally Exposed Plant Personnel: - $5 \times 10^{-5}\ y^{-1}$ peak individual risk and - $10^{-5}\ y^{-1}$ average risk Members of the Public (critical group), non-occupationally exposed plant personnel and visitors: - $5 \times 10^{-6}\ y^{-1}$ peak individual risk due to all nuclear installations and - $10^{-8}\ y^{-1}$ average risk imposed by the *facility*

Table 1: Basic Licensing Requirements for the PBMR

8 DEMONSTRATION OF COMPLIANCE WITH THE BASIC LICENSING REQUIREMENTS

8.1 Basic Requirements for Demonstration of Compliance

(21) The *applicant* / licensee must provide the NNR with a *Safety Case* in support of the licence application, which must demonstrate the adequacy of the *facility* design and operational procedures against the requirements of this document (which may include others as indicated in Section 4). The scope and content of the *Safety Case* must consider the applicable NNR Licensing Guides and Requirement Documents and demonstrate compliance thereto.

(22) The extent and significance of any design changes after definition of the *Safety Case* must be identified and analysed in order to ensure the consistency and applicability of the *Safety Case* for licensing.

(23) Compliance with the *BLR* must be demonstrated by way of formalised safety analyses with reference to proven technology and in accordance with international practice (INSAG-12 [6] (154)). Such analyses must include both deterministic analyses and *probabilistic risk assessment*. *Conservative safety analyses* must be applied for demonstration of compliance with the design dose limits and constraints for category A and B events, *best estimate analyses* can be applied for demonstration of compliance with ALARA targets and the probabilistic risk assessment of category C events.

(24) The safety analyses must cover all categories of events (A, B and C) in a structured and enveloping way so that all the potential radiological consequences of *PIE* are covered.

8.2 Demonstration of Compliance for Category A

8.2.1 General Requirements

(25) Category A includes *Normal Operation* as well as *AOO.* A prospective design analysis must be performed which:

(i) Demonstrates compliance with the annual dose limits to both the plant personnel and members of the public as required in section 7 and the respective dose constraints to be defined by the applicant for NO and AOO.
(ii) Includes the establishment of ALARA design targets for annual individual and collective dose and must demonstrate compliance with the ALARA principle through quantitative analysis using *best estimate data.*
(iii) Estimates all normal operational annual radiological releases and those resulting from AOOs from the *facility* to the public and associated doses to the critical group from all pathways under conservative assumptions, taking into account all operational modes of the *facility*. This must include provision to demonstrate that radiological releases are ALARA and that adequate defence-in-depth has been included in the design of systems that collect,

store, process, monitor and discharge radiological releases. The entire life cycle, including retrofitting and decommissioning, must be addressed.

(iv) Considers PBMR source term characteristics as well as relevant siting factors including the sources of radioactivity, the nuclide selection, and the transport and distribution mechanisms.

(26) An operational radiation protection programme must be designed and put in place to ensure that all radiological exposures associated with normal operations are identified and quantified with a view to the implementation of control measures to ensure compliance with the dose limits and the application of the ALARA principle. The operational radiation protection programme must be consistent with the design analysis. The entire life cycle, including retrofitting and decommissioning, must be addressed by both the design analysis and the operational radiation protection programme.

8.2.2 Radiation Protection Requirements

With this objective, the following requirements apply:

Control of Radiation Exposure, Radiation Monitoring and Radioactivity Release Management

(27) Radiation exposure must be optimised by using appropriate design, confinement, shielding, radiation monitoring, personal protective equipment and zoning of the *facility*. *Exclusion areas* must be defined for areas where access must be prevented during operation, depending on the operational mode and state of the facility to avoid uncontrolled and over exposures.

(28) Appropriate dose monitoring must be provided for all individuals entering controlled areas of the *facility* where doses are expected to exceed the natural background radiation. Monitoring must allow for dose assessment in terms of the dose limits defined in this RD and [1] and the ALARA targets to be defined by the applicant.

(29) An operational radioactive release management programme must be defined that is consistent with the category A design dose limits, design dose constraints and the ALARA targets. The programme must include a system to quantify the radionuclides discharged to the environment from all pathways.

(30) Limits of direct radiation and radiological discharges must be defined that the *facility* is required to meet. Such limits necessarily imply the use of *conservative data* for the derivation of dose conversion factors.

(31) Appropriate monitoring / instrumentation must be provided at all pathways of radioactive discharge from the *facility*.

(32) A prospective radiological analysis must be performed, which demonstrates compliance with these limits in respect of direct radiation, discharge control, radioactive waste management and transport of radioactive material.

Requirements in respect of dose limitation must be considered both in the design of the SSC and operation, including the execution of operational programmes.

(33) The specifications must include the establishment of ALARA targets for annual individual dose to the plant personnel and the highest exposed group of individual members of the public and must demonstrate compliance with these targets through radiological analysis. *Best estimate data* may be used.

Radioactive Waste Management

(34) A prospective analysis must be performed which classifies and quantifies the normal operational annual radioactive waste quantities from the *facility*, under conservative assumptions, taking into account all operational modes of the *facility*. The lifecycle of the *facility*, including retrofitting and decommissioning, must be addressed.

(35) An operational radioactive waste management programme, including decommissioning waste, must be defined consistent with the prospective analysis and the conditions defined in [1] and with the requirements for disposal of the different kinds of radioactive waste. The programme must include a system to characterize the waste at all stages in the process of waste management and to quantify the radionuclides in all radioactive wastes and it must make particular provision for quantification of radionuclides that are of a long-lived nature.

(36) Design targets must be defined for the annual quantity of radioactive waste produced and compliance with these targets must be demonstrated through quantitative analysis. *Best estimate data* may be used. This must include provision to demonstrate that radioactive waste quantities are kept to the minimum practicable and that confinement of radioactivity is ensured by adequate design of the SSC that collect, store, process, condition and package radioactive waste.

Transport of Radioactive Material

(37) The prospective analysis must demonstrate that, for all radioactive waste and materials that it is intended to ship from the *facility*, the engineered package designs comply with the appropriate requirements of IAEA Regulations for Safe Transport of Radioactive Material.

(38) The operational programme must ensure that the shipment process complies with the requirements of the appropriate IAEA Regulation for Safe Transport of Radioactive Material.

8.3 Demonstration of Compliance for Category B Events

(39) According to the DiD principle, PBMR *safety functions,* separate to the operational control and limitation functions, must be identified and measures provided to cope with the consequences of category B events and to ensure that

the *FSF* are not violated. No credit shall be taken in the analyses for category B events from *early operator actions* or *Event Management*.

(40) Design criteria and appropriate requirements must be established for the category B events to meet the BLR and to balance the design margins with the frequency and the consequences of the category B events taking the risk contribution and the ALARA principle into account. For this purpose the range of category B may be subdivided.

(41) Deterministic analyses must be performed which demonstrate compliance with the *BLR* in design and operation using *conservative assumptions* (pessimistic with regard to the resulting source terms and the *BLR*). These deterministic analyses must cover all *PIE* arising from inside or outside the *facility* and determine the resulting source terms causing exposures to the personnel and the public considering the PBMR source term characteristics as well as relevant siting factors including sources of radioactivity, the nuclide selection and the transport and distribution mechanisms. The *PIE* selection must be based on a comprehensive list of *IE* derived according to subsection 8.4.2.

(42) The *deterministic framework* resulting from the provisions against category B events must be balanced in a way that the staged dose constraints, to be defined in the frequency range of category B, are met. In the lower range of the category B frequencies, where the uncertainties can be high, the degree of conservatism of deterministic analysis may be justified on a case by case basis. For this purpose the range of category B may be subdivided.

(43) The most limiting Single Failure must be applied to the functional systems of *SSC* providing the required safety functions and taken credit for in the analyses. Any exception to the application of the Single Failure Criterion needs detailed and individual justification.

(44) *Exclusion areas* must be defined within the controlled areas to ensure that the plant personnel will not be exposed uncontrolled or in excess of the dose limits.

(45) In order to keep the resulting dose ALARA, emergency measures consistent with section 8.6 have to be defined for events where the potential dose consequences to the public can exceed the annual effective dose of 1 mSv resulting from a category B event.

8.4 Demonstration of Compliance for Category C Events

8.4.1 Design Considerations

Category C envelopes the range of all events (category A, B and beyond category B events) that can potentially lead to exposure down to *IE* frequencies of less than 10^{-6} per year (Beyond category B Events) and which are still relevant for the risk criteria defined in [1].

(46) All events or combination of events, including those with an annual frequency < 10^{-6} (beyond category B events), have to be assessed probabilistically in order to

assess the total risk imposed by the *facility* and to identify those events or combination of events that are major contributors to risk. The demonstration for the *Probabilistic Risk Assessment* of category C events may be carried out using *best estimate analysis* and data.

(47) Design criteria and appropriate design requirements for the category C events need to be established and adjusted to the frequency of the events to meet the different *BLR* for category A, B and C.

8.4.2 Initiating Events

8.4.2.1 Internal Events

(48) All *IE* originating from within the *facility* boundary (and possible combinations of them) must be identified and considered. They must include but are not limited to *IE* leading to consequences such as:

- Failures of pipes, vessels, tanks, pumps and valves,
- Transients (e.g. of the reactor core and the power conversion unit),
- Air and water ingress to the core,
- Loss of power supply,
- Flooding,
- Internal missiles,
- Load drop,
- Internal explosion,
- Internal Fires

Their contribution to the radiological consequences must be included in the *probabilistic risk assessment*.

8.4.2.2 External Events

(49) All *IE* potentially originating from outside the *facility* boundary (and possible combinations of them) must be identified and considered. They must include but are not limited to:

Natural hazards:
- seismic events,
- weather phenomena,
- precipitation and external flooding,
- other natural hazards as water pollution, coastal erosion, tsunami, etc.

Man made hazards:
- aircraft crashes,
- explosion pressure waves,
- toxic, corrosive or combustible gases,
- external fires,
- terrorist attacks

Their contribution to the radiological consequences and risk must be included in the *probabilistic risk assessment*.

8.4.3 Probabilistic Risk Assessment

(50) As indicated above the overall safety analysis must include both a deterministic analysis of category A and B events as well as a *probabilistic risk assessment* (PRA) of category C. With specific emphasis on the *PRA*, this analysis must:

(i) Provide a systematic analysis giving confidence that the design will comply with the *BLR* for category C events,
(ii) Demonstrate that a balanced design has been achieved such that no particular feature or event makes a disproportionately large or significantly uncertain contribution to the overall risk,
(iii) Provide confidence that the design will prevent sudden escalation in the consequences of any event,
(iv) Provide an assessment of the frequency and consequences of internal *IE*,
(v) Provide an assessment of the frequency and consequences of external *IE*, in particular those unique to the *facility*,
(vi) Identify SSC for which design improvements or operational procedures could reduce the frequency of *beyond category B events* or mitigate their consequences,
(vii) Provide an assessment of the beyond category B events with larger consequences,
(viii) Provide input to the determination of the emergency preparedness requirements,
(ix) Verify compliance of the applied data with established probabilistic data.

(51) The *probabilistic risk assessment* must be reasonably balanced and supported through the use of deterministic arguments that allow judgements to be made about the degree of confidence to be given to these estimates and the assumptions.

(52) The applicant must demonstrate compliance with the risk limits in accordance with [1] and the applicable NNR requirements.

8.4.4 Provision against Beyond Category B Events (risk aversion criterion)

(53) The 10^{-8} y^{-1} average risk criterion for category C events for members of the public imposed by the *facility* covers all events. The provision against *beyond category B events* must ensure that the probability of events with larger accidental consequences is more remote than could be allowed by the average population risk criterion. The applicant must demonstrate compliance with the risk aversion criterion in accordance with the applicable NNR requirements (see Appendix 2 of [2]).

8.5 Verification and Validation of Safety Analyses, and Quantification of Uncertainties

(54) All quantitative techniques used for the individual safety analyses like computer codes must be appropriately verified and validated. The definitions and requirements are given in [5].

(55) It must be ensured that the quantitative techniques used for the deterministic and probabilistic analyses take into account all the potential uncertainties that exist so that an estimate can be made of the confidence level to be ascribed to the quantitative results and the demonstration of the level of conservatism that exists in them. Comprehensive and systematic sensitivity studies and uncertainty analyses must be performed to determine those uncertainties that are most important in each case. The definitions and requirements are given in [5].

8.6 Emergency Preparedness

(56) In addition to the engineered safety features of the *facility*, emergency or remedial measures must be considered where there is a potential for the off-site annual individual effective doses to the public to be more than 1 mSv. In selecting the events where this may apply, consideration must be given to the mitigation of the consequences arising from those events that have been identified in the safety analysis. The requirements stipulated in [4] apply.

(57) In the event of an emergency or when responding to an event by *event management*, appropriate measures are to be defined. This may result in a dose in excess of the operational annual dose limit, to plant personnel, for the purpose of saving life or preventing serious injury or if undertaking actions intended to avert a large collective dose or if undertaking actions to prevent the development of catastrophic conditions. The dose limits for such actions are stipulated in [1].

(58) The extent of the emergency response plan must be commensurate with the radiological consequences predicted for these event sequences. In determining the extent of the emergency response plan, the following must be defined:

(i) An Exclusion Zone (EZ), beyond which no evacuation of members of the public would be required arising from the reference event must be determined. Within the boundaries of that zone or within any erven intersecting with that zone there must be no members of the public resident, no recreational activities, no commercial activities, or institutions which are not directly linked to the operation of the *facility*.

(ii) An overall Emergency Planning Zone (EPZ), for implementation of emergency or remedial measures must be defined. The EPZ must cover the area where the potential exists that any members of the public may receive more than an effective dose of 1mSv.

(iii) A Long Term Protective Action Planning Zone (LPZ), extending beyond the EPZ boundaries, where preparations for effective implementation of

protective actions to reduce the risk of stochastic health effects from long term exposure to deposition and ingestion must be determined.

8.7 Decommissioning

(59) In line with section 5.1.1 of [1] a decommissioning strategy must be submitted as part of the prior safety assessment and must be updated throughout the operation of the authorised action as a basis for detailed decommissioning planning.

(60) In line with section 5.1.2 of [1] a decommissioning plan must be submitted as a basis for authorisation of specific actions or phases of decommissioning.

9. References

[1] Regulations in Terms Of Section 36, Read With Section 47 Of The National Nuclear Regulator Act, 1999 (Act No. 47 Of 1999), On Safety Standards And Regulatory Practices (Published in Government Gazette 28755 April 2006)

[2] RD-0024: Requirements on Risk Assessment and Compliance with Principal Safety Criteria for Nuclear Installations.

[3] RD-0034: Quality and Safety Management Requirements for Nuclear Installations

[4] RD-0014: Emergency Preparedness and Response Requirements for Nuclear Installations

[5] RD-0016: Requirements for licensing submissions involving computer software and evaluation models for safety calculations

[6] INSAG-12 "Basic Safety Principles for Nuclear Power Plants 75-INSAG-3 Rev 1"

[7] "Act No. 47 of 1999: National Nuclear Regulator Act, 1999", published in Republic of South Africa Government Gazette, Vol. 414, No. 20760, 23. December 1999.

APPENDICES

Appendix A: TERMS, ABBREVIATIONS AND DEFINITIONS SPECIFIC TO THE PBMR

Overall terms and definitions are given in [1] and [7]. Terms, abbreviations and definitions specific for this RD are given below.

Term / Abbreviation	Definition
Anticipated Operational Occurrences (AOO)	An operational process deviating from *normal operation* which is expected to occur at least once during the operating lifetime of a *facility* but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety nor lead to category B conditions.
Applicant	Applicant for a nuclear installation licence to site, construct, operate and decommission a pebble bed modular reactor.
Basic Licensing Requirements (BLR)	The BLR are the dose and risk limits to be complied with for the plant personnel and the public as defined for the event categories A, B and C. The application of the ALARA and DiD principles are part of the BLR. The BLR are based on the *Principal Safety Requirements*. Other licensing requirements for the PBMR, covering specific areas are stipulated in additional Regulatory Requirements and Guidance Documents as developed by the NNR.
Best estimate analysis / assumptions / data / results	An analysis that is performed on the basis of the mechanistic behaviour of systems and processes, providing most probable assumptions and values where uncertainties exist and avoiding over-conservative assumptions. For such analyses justified representative input data are used with the purpose of arriving at a realistic and representative set of best estimate results.
Beyond category B Events	Beyond category B events are events which give rise to conditions more severe than those anticipated for the *design basis* of the PBMR. They are events (or combination of events) with a frequency < 10^{-6} y^{-1} which extend the design and safety analysis of the facility beyond the category A and B events to demonstrate that no *cliff edge effects* occur. Appropriate design rules and criteria must be set for beyond category B events, which may differ from those for category A and B events. Beyond category B events must be included (along with the category A and B events) in the Safety Analyses of the category C events to demonstrate that the *BLR* are met. .
Cliff Edge Effect	IAEA definition in INSAG-12, [6] (52): Effects which might permit small deviations to precipitate grossly abnormal facility behaviour and cause damage.
Conservative safety analysis/ assumptions / data / results	The deterministic safety approach requires adequate margins. This is achieved through analyses using conservative assumptions and input data without the introduction of a final margin. For such analyses input data pessimistic in terms of the analytical results are used with the purpose of arriving at a set of safety analysis results that are demonstrably pessimistic in comparison with any likely result.

Term / Abbreviation	Definition
Design Basis	The range of conditions and events taken explicitly into account in the design of the PBMR according to established design criteria, such that the facility can withstand them without exceeding authorized limits.
Design dose limit	Dose limit to individuals that must not be exceeded and must be complied with by the facility design provisions.
Deterministic framework	The deterministic framework of the facility is defined by the designed functional provisions that are required to cope with the design challenges presented by the PIE. The functional provisions that are required, are an outcome of a deterministic functional analysis approach.
Early operator actions	Early operator actions are considered to be any intervention by the operator in the facility operation with the aim of event mitigation at a time after the occurrence of an *IE* when reliability of the actions is too limited. According to international practice, no credit must be taken from operator actions for at least the first half of an hour after occurrence of an incident. Early operator actions relate to DiD level 3.
Event management	The taking of a set of operator actions during the evolution of an event: - to prevent the escalation of the event; - to mitigate the consequences of an event; and - to achieve a long term safe stable state. No credit must be taken for Event Management measures for category A and B events in meeting the BLR. Event management relates to DiD level 3.
Exclusion areas	Exclusion areas are those radiologically controlled areas where access must be prevented during operation, depending on the operational mode and state of the facility to avoid uncontrolled and over exposures.
Facility	The totality of technical equipment / installations necessary for operation of a nuclear installation. A *facility* is subject to a single nuclear installation license or application and may consist of one or a number of modules. Note: The definition of the term *facility* in the RD is consistent with the usage of the term *site* in [1], [2] and [7].
Functional systems of SSC	Functional systems of SSC are defined as a specific and complete configuration of structures, systems and components all of which, working together, provide a required specific safety function. A functional system of SSC usually comprises more than one facility system and includes all the SSC, including support functions, that are necessary to achieve and monitor the safety function.
Fundamental Safety Functions (FSF)	The Fundamental Safety Functions to be ensured for a nuclear reactor are defined as - Reactivity Control - Heat Removal - Confinement of Radioactivity The FSF are provided by single or combinations of the PBMR *Safety Functions*.
Good Engineering Practice	Practices or rules usually applied for a purpose by experienced practitioners and verified by frequent successful application in similar situations.
ICRP	International Commission on Radiological Protection
Initiating events (IE)	All event initiators and combination of independent event initiators (occurring at the same time) that might lead to exposure.

Term / Abbreviation	Definition
Internal / External events	Internal and external events are events that originate within or outside the facility with the potential to cause adverse conditions or even damage to safety important structures, systems or components. These effects can potentially lead to a common cause failure within the systems used to reach or to maintain the facility in a safe state. The objective of the design provisions against internal and external events is to ensure that the safety functions of the structures, systems or components, which are required to bring or maintain the facility in the safe shutdown state are not unduly affected to ensure that the *Fundamental Safety Functions* are maintained.
Normal Operation	Operation within specified operational limits and conditions. This includes starting, power operation, shutting down, shutdown, maintenance, testing and refuelling. Category A includes *Normal Operation* and *AOO*.
PBMR Safety Functions	The PBMR Safety Functions are those safety functions (or combinations thereof) specific to the PBMR design that are provided by the SSC that must ensure that the *FSF* are met.
Postulated Initiating Events (PIE)	*PIE* are the enveloping Initiating Events – covering one or several IE and combinations of *IE* – but excluding mitigation activities. Based on justified frequencies – taking uncertainties into account – the *PIE* are to be allocated to the event categories A, B and beyond B. The comprehensive set of PIE forms the basis of the deterministic Safety Analysis.
Principal Safety Requirements	The *Principal Safety Requirements* are overall radiation protection and nuclear safety requirements that are stipulated as having to be met as part of the *Basic Licensing Requirements* (BLR) for the PBMR
Probabilistic Risk Assessment (PRA)	A comprehensive, structured approach for deriving numerical estimates of risk. The PRA must cover the category C events and demonstrate that the BLR are met for category C.
Recovery Actions	Recovery actions are means and/or operator actions following a category B event to bring the facility from a controlled stable state (safe shutdown and controlled heat removal and radioactivity release state) into a long term safe shutdown state. These actions could be associated with large dose burden and are to be considered in design.
Safety Case	Documentation demonstrating the safety of the *facility* against the BLR. The Safety Analysis Report (SAR) is part of the Safety Case.

Appendix B: Application of the Defence in Depth (DiD) Principle for the PBMR

B-1 Safety functions

The DiD approach has to be implemented in respect of the *Fundamental Safety Functions* (FSF):

- Reactivity control
- Heat removal
- Confinement of radioactivity.

Sufficient PBMR *safety functions* shall be provided to ensure that the *FSF* are maintained and to provide the required levels of DiD.

As a result of the adoption of the DiD principle, the PBMR shall be designed so that DiD can be substantiated for the PBMR by the provision of:

- Sufficient independent reactivity control functions
- Sufficient independent heat removal functions
- Sufficient independent barriers for confinement of fission and activation products

B-2 Levels of defence in depth

The defence in depth concept as described in the IAEA documents: e.g. [6].

The DiD principle requires that various lines of defence are provided by design and appropriate procedures to ensure the *FSF*.

Detailed analysis and assessment of the design of the *facility* and the various systems and procedures are required to ensure that the lines of defence or barriers are of satisfactory quality and independence, taking into account all the *facility* provisions and operating procedures.

The safety philosophy is aimed primarily at the prevention of events but also gives attention to the mitigation of the consequences of events that could give rise to radioactive releases. The aim is to reduce both the probabilities of the events and their associated radiological consequences (inside and outside the *facility*).

The use of the following well established principles of defence in depth is required:

- Prevention of deviation from *normal operation*
- Detection of deviations from *normal operation* and provision of means to prevent such deviations leading to category B events.
- Provision of engineered safety features (active and/or passive to control and mitigate the category B events.

- Prevention and mitigation of *beyond category B events* through the consideration of events or combinations of events with an annual frequency $<10^{-6}$. Emphasis shall be put on prevention of beyond cat B events. Realistic assumptions and best estimate methods may be used to analyse these conditions.
- Mitigation of radiological consequences of significant releases of radioactive materials by means of off-site emergency response.

B-3 Barriers

A second complementary aspect of the defence in depth principle is the concept of multiple, independent physical barriers to the uncontrolled release of radioactive material to the environment. The demonstration of the adequacy of these barriers is an important part of the safety analysis.

These barriers shall be designed on the basis of the *facility's* lifetime, both for steady states and transients occurring in any operational conditions and accident conditions.

The *facility* shall be designed so that:

- Sufficient independent barriers for confinement of fission products are provided.
- The confinement of the fission products is ensured by these barriers with sufficient margins for all category A events.
- The integrity of nuclear fuel is maintained for all category A and B events and fuel failures due to accidental conditions are minimised even for *beyond category B events*.
- The integrity of the Primary Pressure Boundary (PPB) is maintained for all category A and B events except for the failure assumptions to be set for the PPB itself.
- The overall radioactivity confinement function of the civil structures forming the confinement functional design shall be ensured with sufficient margins for all category A events.
- The integrity of the civil structures forming the confinement functional design of the building shall be ensured for the category B events. Provisions shall be made to minimise the damage of the civil structures for *beyond category B events*.
- For *beyond category B events* at least one confinement function must be adequately maintained in such a way that no *cliff edge effects* occur.

B-4 Accident prevention

The importance of prevention of accidents as the main basis of the safety is emphasised.

The primary objective of nuclear power *facility* designers is to provide a sound and balanced design. The SSC of the *facility* shall have the appropriate characteristics, specifications and material composition and shall be combined and laid out in such a way as to meet the *facility* specifications. These specifications shall be consistent with the requirement to meet the safety objectives, the specified duty in terms of electrical output, availability, projected lifetime, and the operations necessary to meet system demands. In respect of the principle of defence-in-depth [6] (46-55) and accident prevention [6] (56-62, and 159), the design shall ensure that exposures to the

personnel and the public exceeding the category A dose criteria are unlikely to occur during the lifetime of the *facility*.

Fuel element design, fabrication and inspection, and the conditions under which the fuel is operated shall be such as to ensure a high degree of integrity.

The integrity of the reactor coolant system as well as that of the systems connected to it shall be ensured by the design with adequate margins.

The design shall aim to provide a *facility* that is simple to operate and maintain. At the design stage, consideration shall be given to the performance capabilities of the personnel who will operate and maintain the *facility*. The designer shall supply information and recommended practices for incorporation into operating procedures. The design shall aim for simplicity, adequate margins and forgiving characteristics to minimise the consequences of operator errors.

Experience feedback from nuclear operating power *facilities* and, as applicable, from other industrial *facilities* shall be extensively and systematically used in the design process. Proven components are to be preferred unless alternatives provide clear advantages in one or more specific areas (e.g. safety, cost, reliability) without significantly affecting the others.

Attention shall be paid to the requirements for inspections, testing, on-line monitoring and maintenance, also in their potential to prevent accidents.

The controls shall maintain the reactor within the parameters set for *normal operation*. The objective shall be to reduce the number of challenges to the reactor protection system.

If deviations from normal operation conditions occur which cause specific limits to be exceeded, the operational control systems shall detect such conditions and prevent them from leading to category B or *beyond category B events*.

B-5 Accident mitigation

Notwithstanding all preventive features to prevent radiological consequences of events, mitigative measures shall be provided to minimise the radiological consequences through the barriers.

For the *design basis* the confinement system of the building shall be designed to meet the *BLR*. The maximum allowable source terms from the confinement (including leakage rates and depressurisation) shall be defined to satisfy the *BLR* for the various *PIE*, and the means to monitor and maintain such leak rates and releases shall be provided.

The engineered safety features providing the PBMR *Safety Functions* to control the development of accidents shall be shown to meet the *BLR*.

The use of inherent characteristics and the simplification of systems are seen as important design aims. Passive safety features shall be used where appropriate and of

overall safety benefit. Adequate time scales are required for any operator actions. Simplification of systems design should facilitate elimination of adverse system interactions.

Measures shall be addressed to prevent fuel damage or to mitigate the consequences of event sequences that go beyond the *deterministic framework* of category B, using appropriate design rules. Such measures shall be implemented taking account of probabilistic safety analyses where such sequences make a significant contribution to risk.

Appendix C: Description of the General Principles of the ALARA approach

C-1 Protection Principles

In 1990 the International Commission on Radiation Protection issued a set of recommendations detailing a system of radiological protection for practices based on the following three general principles:

- **Justification**

  No practice involving exposures to radiation should be adopted unless it produces sufficient benefit to the exposed individuals or to society to offset the radiation detriment it causes.

- **Optimisation**

  In relation to any particular source within a practice, the magnitude of individual doses, the number of people exposed, and the likelihood of incurring exposures where these are not certain to be received, should all be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account.

- **Limitation**

  The exposure of individuals resulting from the combination of all the relevant practices should be subject to dose limits.

The aim of these recommendations is to avoid doses without benefit on the one hand, and not to reduce doses at all costs on the other hand. In all cases, doses are to be kept below the dose limits.

C-2 Implementation of the ALARA principle in the design stage

C-2.1 General

The first principle is a very general one that encompasses the usual engineering process (i.e. justification of the practice; here, justification of electricity generation using nuclear power *facility*). However, it has a practical application in the ALARA methodology when questioning the usefulness of a task to be performed for the operation and maintenance of a nuclear power station.

The second principle requires the implementation of an optimisation process at the design stage. The engineering process should be such that a balance between the cost of protection dispositions (reduction of sources, shielding, use of robotics...) and the corresponding savings in doses are achieved.

The third principle sets the boundary conditions for the exposure of workers. Design guidelines have to be established for *facility* layout, system and equipment engineering at the beginning of the project.

This approach makes it possible to implement necessary provisions already from the start of the design work when there is generally a lack of parameters allowing ALARA studies.

The resulting *facility* design based on guidelines allows optimisation in terms of the ALARA principle. Individual features e.g. component design alternatives or individual shielding provisions are questioned. Design modifications should also be justified by analysing the impact on the personnel exposure.

C-2.2 Occupational radiation exposure target

Even though there is no limit imposed by regulation on the collective dose, a part of the process to perform radiation protection optimisation is to set a target value. It implies the verification of the possibility to fulfil the objectives throughout the design phases.

C-2.3 Respect of the individual dose limits

In the early stage of the design, it is not possible to carry out detailed studies of individual doses, which have to be based on realistic assumptions. The problem is solved by the general dispositions that are aimed at providing low dose rates at work sites and an ergonomic working environment. These dispositions, that can be described as "passive protection", ensure that further optimisation of individual doses will be possible.

C-2.4 Approach to the design targets

The ALARA activities accompanying the project evolution reflect the actual engineering stage. They shall be conducted following a stepwise approach.
The first step corresponds to the engineering documents of a basic engineering phase, describing general principles of dose and source term optimisation and justifying the target value for the individual and collective dose based on operating experience feedback.

A more detailed investigation on doses based on an analysis of maintenance tasks must correspond to the engineering documents of a detailed design.

Means necessary to reach the collective dose target shall be implemented at design level. These means cover the various contributors to collective dose: sources, installation and maintenance program. As regards maintenance, some aspects are operational: preparation of work, training of personnel, organisation minimising number of persons and time spent in active zones; other aspects depend on design features: accessibility, separation in the layout, handling easiness, in-service inspectability, decontamination facility, use of robotics or automation. Only design aspects need to be dealt within a basic engineering phase.

C-2.4.1 Source term optimisation

Among factors determining the source term, special attention must be given in the design to the choice of materials. In order to avoid hot spots, components and piping in activity containing systems shall be designed so that deposits are limited, e.g. corners, gaps and dead zones of flow shall be avoided, a sufficient flow velocity in pumps, blowers, compressors valves, piping shall be chosen.

C-2.4.2 Layout aspects

Layout features that contribute to the collective dose through features such as e.g. accessibility, separation, shielding, handling, setdown areas, dress-out provisions, must be considered.

C-2.4.3 Maintenance and in-service inspection

In addition to source term optimisation and layout considerations, attention shall be given to various component design features, e.g.

- tanks, vessels, and heat exchangers are designed to avoid radioactive deposits or at least to remove them easily,
- adequate access and space is given to welds and parts to be inspected or maintained,
- components are made as reliable as possible.

Experience feedback shows that not all maintenance work has the same importance in terms of individual and collective dose. Work areas giving the most important contribution to the exposure doses shall be selected to be the subject of design recommendations, including possible use of remotely controlled means.

C-2.4.4 Facility management strategy

In addition to the design features of the facility, operational and maintenance aspects contribute to the reduction of the collective dose. Consideration shall be given to:

- optimisation of operating procedures to minimise transients,
- use of thermal insulation and scaffolding elements with short installation and dismantling times in order to reduce man-hours under radiation exposure,
- restrictive control of stay-time of personnel during outages and during power operation inside the controlled area.

C-2.5 ALARA results at the concept design phase

It is recognised that during the initial engineering design phase only fundamental aspects of radiation protection can be dealt with. Therefore no definitive statement about the expected total exposure is expected.

C-2.6 Provisions to maintain risk ALARA

Event management should include pre-planned and ad hoc operational practices, which, in circumstances in which the design basis specification of the facility is exceeded, would make optimum use of existing facility equipment to restore control. This applies to prevention of core damage and mitigation of *beyond cat B events*.

Event management procedures and equipment should be provided which would allow the facility to be restored to a safe shutdown state, with long-term core cooling assured and radioactive material confined.

Sufficient instrumentation has to be provided to monitor response and to allow the necessary actions to be carried out. The operability of this instrumentation shall be demonstrated under relevant conditions.

Attention shall also be given to the actions that operators may be asked to perform during and after accidental conditions. Equipment accessibility and proper evaluation of radiation dose rate where the presence of the operator is required shall be carried out.