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GOVERNMENT OF INDIA

**AERB SAFETY GUIDE**

# DETERMINISTIC SAFETY ANALYSIS FOR SODIUM COOLED FAST REACTOR BASED NUCLEAR POWER PLANTS

**ATOMIC ENERGY REGULATORY BOARD**

AERB SAFETY GUIDE

# DETERMINISTIC SAFETY ANALYSIS FOR SODIUM COOLED FAST REACTOR BASED NUCLEAR POWER PLANTS

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# FOREWORD

The Atomic Energy Regulatory Board (AERB) was constituted in 1983, to carry out certain regulatory and safety functions envisaged under Section16, 17 and 23 of the Atomic Energy Act, 1962. AERB has powers to lay down safety standards and frame rules and regulations with regard to the regulatory and safety requirements envisaged under the Act. The Atomic Energy (Radiation Protection) Rules, 2004, provides for issue of requirements by the Competent Authority for radiation installations, sealed sources, radiation generating equipment and equipment containing radioactive sources, and transport of radioactive materials.

With a view to ensuring the protection of occupational workers, members of the public and the environment from harmful effects of ionizing radiations, AERB regulatory safety documents establish the requirements and guidance for all stages during the lifetime of nuclear and radiation facilities and transport of radioactive materials. These requirements and guidance are developed such that the radiation exposure of the public and the release of radioactive materials to the environment are controlled; the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation is limited, and the consequences of such events if they were to occur are mitigated.

The Regulatory documents apply to nuclear and radiation facilities and activities giving rise to radiation risks, the use of radiation and radioactive sources, the transport of radioactive materials and the management of radioactive waste.


Fig. 1 Hierarchy of Regulatory Documents

Safety codes establish the objectives and set requirements that shall be fulfilled to provide adequate assurance for safety. Safety Standards provide models and methods, approaches to achieve those requirements specified in the safety codes. Safety guides elaborate various requirements specified in the safety codes and furnish approaches for their implementation. Safety manuals detail instructions/safety aspects relating to a particular application. The hierarchy of Regulatory Documents is depicted in Figure.1.

The AERB Safety Code on Design of Sodium Cooled Fast Reactor Based Nuclear Power Plants (AERB/NPP-SFR/SC/D) establishes the requirements for carrying out a comprehensive safety analysis by utilities to evaluate the radiation doses that could be received by plant personnel and the public, as well as the potential effects of radioactive releases on the environment. Safety analysis of the Nuclear Power Plant (NPP) design, applying deterministic methods, establishes and confirms the design basis for the items important to safety and demonstrates that the overall plant design ensures radiation doses and releases are within the prescribed limits for operational states and acceptable limits for accident conditions. This Safety Guide deals with establishing and confirming design basis for Sodium cooled fast reactor based NPPs, applying deterministic methods in safety analysis for the items important to safety and design provisions. In drafting this document, the relevant AERB Safety Codes and Guides on Design, Operation, Siting, etc., International Atomic Energy Agency (IAEA) documents and other international documents on Deterministic Safety Analysis have been used. This safety guide is effective from the date of its issue and it applies to sodium cooled reactor based NPPs built after the issue of this document. However, the guidance contained in the safety guide for performing DSA should also be applied to NPP designs currently under review by AERB, to the extent practicable, integrating the socio-economic considerations and to the existing operating NPPs during periodic safety review (PSR) to assess to which extent the existing NPPs conform to current standards.

The recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP) and the International Atomic Energy Agency (IAEA) are taken into account while developing the AERB Regulatory safety documents.

The principal users of AERB regulatory safety documents are the applicants, licensees, and other associated persons in nuclear and radiation facilities including members of the public. The AERB regulatory safety documents are applicable, as relevant, throughout the entire lifetime of the nuclear and radiation facilities and associated activities. The AERB regulatory safety documents also form the basis for AERB's core activities of regulation such as safety review and assessment, regulatory inspections and enforcement.

Safety related terms used in this safety guide are to be understood as defined in the AERB Safety Glossary (AERB/GLO, Rev.1). The special terms which are specific to this safety guide are included under section on 'Special Terms and Interpretation'. In addition, the terms already defined in AERB Safety Glossary AERB/GLO, Rev.1, and being used in this safety guide with a specific context and requires interpretation or explanation are also included in this section.

Appendix is an integral part of the safety guide, whereas annexures, references and bibliography provide information that might be helpful to the user. For aspects not covered in this safety guide, applicable and acceptable National and International codes and standards shall be followed. Industrial safety shall be assured through good engineering practices and by complying with the relevant Industrial safety requirements under prevailing statutes.

This safety guide has been drafted by an in-house working group (WCR). The draft, along with the expert review comments, was further reviewed by Standing Committee for Review of Deterministic Safety Analysis of Nuclear Facilities (SCDSA), which served as Task Force comprising specialists drawn from technical support organisations, institutions, and other consultants. The review comments obtained from stake holders have been appropriately incorporated. The safety guide has been vetted by the AERB Advisory Committee on Nuclear and Radiation Safety (ACNRS). AERB wishes to thank all individuals and organizations who have contributed to the preparation, review and finalization of the safety guide.

D. Shukla.

**(Dinesh Kumar Shukla)**
**Chairman AERB**

## SPECIAL TERMS AND INTERPRETATION

### Cumulative Damage Fraction

Cumulative Damage Fraction (CDF) is a measure used to quantify the extent of damage accumulated in a material over time due to exposure to cyclic stresses and high temperatures. It is particularly significant in the context of high-temperature design for liquid metal reactors (LMRs)

CDF is typically expressed as the sum of damage increments incurred during each loading cycle, with each increment being a fraction of the material's creep and fatigue life under the specific conditions of that cycle. Mathematically, it can be represented as:

$$CDF = \sum_{i=1}^{n} \frac{d_i}{D_i}$$

where:

- $d_i$ is the damage incurred during the $i$th cycle,
- $D_i$ is the total damage capacity for the $i$th cycle conditions,
- n is the number of cycles.

When the cumulative damage fraction reaches a value of 1, it signifies that the material has reached its creep and fatigue life and is expected to fail.

### Hotspot Factor

The hotspot factor is a parameter used to express the degree to which actual reactor performance deviates from nominal design due to uncertainties. It represents the effect of modeling, measurement and manufacturing uncertainties on the prediction of component temperatures. The hotspot factor helps determine the margin by comparing the calculated peak temperature, including uncertainties, against the design safety limits.

### Design Safety Limits

Design Safety Limits (DSL) are predefined thresholds established during the design phase of liquid metal reactors to ensure safety under operational states and accident conditions. These limits encompass various operational parameters such as temperature, pressure, stress, and radiation levels. Exceeding these limits could compromise the reactor safety.

## CONTENTS

# 1. INTRODUCTION

## 1.1 General

1.1.1 AERB safety codes on ‘Design of Sodium Cooled Fast Reactor Based Nuclear Power Plants (AERB/NPP-SFR/SC/D) [1] establish the requirements for carrying out a comprehensive safety analysis to evaluate the radiation doses that could be received by plant personnel and the public, as well as the potential effects of radioactive releases on the environment.

1.1.2 Safety analysis consists of Deterministic Safety Analysis (DSA) and Probabilistic Safety Analysis (PSA). However, in this guide, DSA and safety analysis are used interchangeably and mean the same.

1.1.3 Safety analysis of the Nuclear Power Plant (NPP) design, applying deterministic methods, establishes and confirms the design basis for the items important to safety and design provisions and demonstrates that the overall plant design ensures radiation doses and releases are within the prescribed limits for operational states and acceptable radiation dose targets for accident conditions. Based on this safety analysis, the robustness of the engineering design to withstand events, the effectiveness of items important to safety and safety related systems, design provisions and the basis for emergency preparedness can be established.

1.1.4 Safety analysis is required to be carried out for all plant states, viz. (i) normal operation (NO) (ii) anticipated operational occurrences (AOOs); (iii) design basis accidents (DBAs); and (iv) Design Extension conditions (DECs).

1.1.5 AERB Safety Code on Nuclear Power Plant Operation (AERB/NPP/SC/O, Rev. 1) [2] requires that operating limits and conditions should be related to the results of safety analysis. The safety analysis of the plant design needs to be updated in light of significant changes in NPP configuration, operational experience, improvements in technical knowledge or understanding of physical phenomena, changes in plant operating parameters and operating procedures due to changes in power rating and should be consistent with the ‘as-built’ and ‘as-operated’ state. During Periodic Safety Review (PSR), the validity of the existing safety analysis should be examined and if required it should be updated.

1.1.6 The terms used in this Safety Guide are to be understood as defined and explained in AERB ‘Glossary of Terms for Nuclear and Radiation Facilities and Associated Activities,’ (AERB/GLO), Rev.1.

## 1.2 Objective

1.2.1 The objective of this Safety Guide is to provide guidance and recommendations on performing DSA and its application for sodium cooled fast reactor based NPPs. The main aim of this safety guide is to standardise the methodology for conducting safety analysis to meet the requirements stipulated in AERB safety codes on siting [3], design [1], operation [2], and management of nuclear and radiological emergency [4].

## 1.3 Scope

1.3.1 This safety guide provides guidance on various approaches for carrying out DSA, safety analysis rules to be followed, acceptance criteria for various plant states, verification and validation of computer codes used for DSA, presentation of results, documentation and update of safety analysis. Guidance on methodology for selection of events to be analysed may be derived from AERB Safety Guide on Design Basis Events (AERB/NPP-WCR/SG/D-5, Rev. 1) [5] applicable for Water Cooled Reactor based NPPs.

1.3.2 This safety guide is applicable for DSA carried out as part of licensing requirements for Preliminary Safety Analysis Report (PSAR), Final Safety Analysis Report (FSAR), Periodic Safety Review (PSR) as well as submissions in support of modifications in operating NPPs that could affect safety. This guide is applicable for Sodium Cooled Fast breeder Reactor based NPPs, as guidance for design of the plant, analysis of events, development of Emergency Operating Procedures (EOPs), Accident Management Guidelines (AMGs), emergency preparedness and response; and for obtaining inputs for Probabilistic Safety Analysis (PSA).

1.3.3 This safety guide deals only with the events and the event sequences that originates in the reactor, the spent fuel or associated process systems. The safety guide does not consider the external and internal hazard aspects.

1.3.4 This safety guide mainly focuses on neutronics, thermal hydraulics, fuel, source term estimation and dose evaluation. Analysis of other aspects such as those involving structural mechanics or electrical transients is not in the scope of this guide.

## 2. ACCEPTANCE CRITERIA FOR PLANT STATES

### 2.1 General

2.1.1 The aim of DSA is to demonstrate that NPP design meets the relevant safety and associated regulatory requirements. The consequences arising from all events (Postulated Initiating Events and Accident Sequences) need to be addressed in the safety analysis.

2.1.2 All events that have a potential for affecting the safety of NPP are to be identified in the safety analysis. This includes all internal and external events and processes that may have harmful consequences on integrity of physical barriers intended for confining the radioactive material or that otherwise give rise to radiological risks.

2.1.3 Computational analysis of all events may not be practicable. A reasonable number of limiting cases, which are referred to as bounding or enveloping scenarios, should be selected from each functional group of events with appropriate justification. These bounding or enveloping scenarios should be chosen so that they present the greatest possible challenge to the relevant acceptance criteria and are limiting for the performance parameters of items important to safety.

2.1.4 Additional guidance on identification, categorization and grouping of events to be analysed in DSA may be derived from AERB Safety guide on design basis events (AERB/NPP-WCR/SG/D-5, Rev. 1) applicable for water cooled reactor based NPPs. The methodology for practical elimination, and relevant events/ phenomena to be practically eliminated may also be referred from this AERB Safety guide. The typical list of events applicable for SFRs is provided in Appendix-I.

### 2.2 Plant States

2.2.1 Plant states for NPPs are divided into operational states and accident conditions. Operational states include Normal Operation (NO) and Anticipated Operational Occurrences (AOOs). Accident conditions include Design Basis Accidents (DBAs), and Design Extension Conditions (DECs). The Design Extension Conditions are sub-grouped as DEC without core melt (DEC-A) and DEC with core melt (DEC-B). Any accident sequence, which may lead to 'Early' or 'Large' radioactive release, should be 'Practically Eliminated'. The list of events to be practically eliminated is also included in Appendix-II.

2.2.2 All events to be analysed should be assigned to a particular plant state to check the consequences against the defined acceptance criteria for that plant state including additional acceptance criteria for that event, if any.

### 2.3 Acceptance Criteria

2.3.1 Once the events to be analysed in DSA are identified, and categorized into particular plant states, then the safety analysis should show compliance with the set of acceptance criteria associated with the plant state to demonstrate safety case.

2.3.2 The criteria should be sufficient to meet the General Design Objective, and the Radiation Protection Objective as given in AERB Safety code on 'Sodium Cooled Fast Reactor Based Nuclear Power Plants (AERB/NPP-SFR/SC/D) [1]'. These acceptance criteria are specified primarily in terms of radiological consequences and also with respect to fundamental safety functions and condition of barriers to radioactivity release. Acceptance criteria are also specified in terms of end state (such as controlled state, safe shutdown state, safe state, severe accident safe state as applicable) mandated for each plant state.

2.3.3 As per Safety Code on design [1], "acceptance criteria shall be assigned to each plant state, such that frequently occurring plant states shall have no, or only minor, radiological consequences and plant states that could give rise to serious consequences shall have a very low frequency of occurrence". Acceptance criteria are established with an aim at preventing damage to relevant barriers in order to prevent radioactive releases (and hence consequences) above acceptable limits.

2.3.4 No event related to spent fuel pool should lead to significant fuel degradation. The acceptance criteria for events up to DEC-A should be appropriately applied for spent fuel pool related events of similar category. The sub-criticality of spent fuel pool should be permanently maintained and spent fuel cooling should be maintained for NO, AOO, DBA and DEC-A.

2.3.5 Acceptance Criteria for Normal Operation (NO)

The annual release limits for all the facilities within a particular site (taken together) shall ensure that the effective dose limit for any individual at off-site, considering all routes of exposure or exposure pathways due to normal operation (including anticipated operational occurrences) is less than the limit specified by AERB Safety Code on Site Evaluation of Nuclear Facilities (AERB/NF/SC/S, Rev. 1) [3]. In addition, plant parameters should be within their respective Limiting Conditions for Operation (LCO), as defined in Station Technical Specifications. LCO include limits on process variables such as cold and hot pool temperatures, average subassembly coolant hotspot[1] temperature, clad hotspot temperature for driver fuel subassembly, blanket subassembly, storage subassembly and fuel centerline hotspot temperature. The limiting conditions on structural temperatures should ensure that the cumulative damage fraction[2] is below specified limits during normal operation.

2.3.6 Acceptance Criteria for Anticipated Operational Occurrences (AOOs)

The annual release limits for all the facilities within a particular site (taken together) shall ensure that the effective dose limit for any individual at off-site, considering all routes of exposure or exposure pathways due to normal operation (including anticipated operational occurrences) is less than the limit specified by AERB Safety Code on Site Evaluation of Nuclear Facilities (AERB/NF/SC/S, Rev. 1) [3]. In addition to this, the following acceptance criteria should also be met for AOOs:

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[1] To express the degree to which actual reactor performance departs from the nominal design as a result of uncertainties, hotspot methods were developed. Hotspot methods are an analytic means for representing the effect that modeling and manufacturing uncertainties have on the prediction of subassembly temperatures. The rod with the greatest nominal temperature, cladding or fuel centerline depending on the application, is selected and then the effect of uncertainties on the temperature at the hotspot for that rod is calculated. The resulting temperature is then compared with design limits to determine the design margin. If sufficient margin exists then all pins in the subassembly meet or are within design margin.

2 The main damage to clad is from creep considerations and cumulative damage Fraction (CDF) approach is followed. CDF is apportioned for each category of event and accordingly temperature limits along with time durations are arrived at. For example, in PFBR [15] CDF apportionment for reactor core components is 0.25 for category 1, 2 & 3 events each and a fraction of 0.25 is reserved for spent fuel storage and handling.

(a) Control of reactivity should be established following AOOs. Reactor Shutdown System and safety systems should not be actuated to the extent possible. In case of reactor shutdown, reactor should be brought to controlled state [1].
(b) The fuel centerline hotspot temperature should not exceed the fuel melting temperature.
(c) Bulk boiling of coolant within a subassembly should not occur[3].
(d) Burnout should not occur at locations of local coolant boiling.
(e) The clad hotspot temperature for driver fuel, blanket and storage SAs should not exceed the design safety limit[4].
(f) Average temperature of cold and hot pools for pool type reactor should be within design safety limits[5].
(g) Primary, secondary and tertiary systems pressure should be maintained below acceptable limits specified in standard used for design.
(h) External pressure on main and safety vessels should be below the buckling limits.
(i) Containment pressure should remain above the designed negative pressure and below the designed positive pressure.
(j) Differential pressure on containment internal structures should not exceed design limit.

### 2.3.7 Acceptance Criteria for Design Basis Accidents (DBAs)

Permitted calculated off-site releases during accident conditions shall be linked to the radiological consequence targets as specified. For DBAs in an NPP, there shall be no need for offsite countermeasures (i.e. no need for prophylaxis, food control, shelter or evacuation) involving public, beyond Exclusion Zone. In such cases, the design target for effective dose should be less than acceptable limit specified in AERB Safety Code on Site Evaluation of Nuclear Facilities (AERB/NF/SC/S, Rev. 1) [3]. In addition to this, the following acceptance criteria should also be met for DBAs:

a) Reactor should be brought to 'controlled state' and subsequently to 'safe shutdown state' [1]. Safe shutdown state should be achieved within 24 hours
b) Reactor should not become prompt critical.
c) Bulk boiling of coolant within a subassembly should not occur.
d) Burnout should not occur at locations of local coolant boiling.

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[3] Sodium hotspot temperature at the subassembly exit should be less than the liquid saturation temperature.
[4] Design safety limits (DSLs) are derived based on the apportioned CDF for given category of events and are typically a combination of temperature and duration limits. For example, typical values of design safety limits for Driver fuel subassembly clad hotspot temperature for PFBR are as below [15]:

- Driver fuel SA: 974-1023 K for 75 minutes & 1023-1073 K for 15 minutes

[5] Typical values of cold pool and hot pool temperature PFBR are as below [15]:

- Cold pool temperature: 813 K
- Hot pool temperature: 873 K

e) The clad hotspot temperature for driver fuel, blanket and storage SAs should not exceed the design safety limit.[6]

f) Average temperature in cold and hot pools for pool type reactor should be within design safety limit.[7]

g) The fuel hotspot temperature should not exceed the fuel melting temperature for event frequency $10^{-4}$/Ry <f < $10^{-2}$/Ry however for event frequency $10^{-6}$/Ry <f ≤ $10^{-4}$/Ry, partial fuel melting at fuel hotspot is allowed to an extent that there is no clad degradation associated with this melting leading to loss of coolable geometry.

h) Primary, secondary and tertiary systems pressure should be maintained below acceptable limits specified in standard used for design.

i) External pressure on main and safety vessel should be below the buckling limits.

j) Containment pressure should remain above the designed negative pressure and below the designed positive pressure.

k) Differential Pressure on Containment internal structures should not exceed design limit.

### 2.3.8 Acceptance Criteria for DEC-A

There shall be no necessity of protective measures in terms of sheltering or evacuation for people living beyond Exclusion Zone. Required control on agriculture or food banning shall be limited to a small area and to one crop. However, the design target for effective dose, with such interventions considered, remains same as for DBA (AERB/NF/SC/S, Rev. 1). In addition to this, the following acceptance criteria should also be met:

a) The acceptance criteria for low frequency ($10^{-6}$/Ry <f ≤ $10^{-4}$/Ry) DBA should be met except for (a).

b) Following DEC-A, the plant should be brought to and maintained in safe state within the timelines as specified in AERB/SFR/SC/D. Thereafter safe shutdown state should be maintained [1].

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[6] Design safety limits (DSLs) are derived based on the apportioned CDF for given category of events and are typically a combination of temperature and duration limits. For example, typical values of design safety limits for Driver fuel subassembly clad hotspot temperature for PFBR are as below [15]:

- Driver fuel SA DSLs ($10^{-4}$/Ry <f < $10^{-2}$/Ry): 974-1023 K for 15 minutes & 1073-1123 K for 6 minutes & 1123-1173 K for 2 minutes
- Driver fuel SA DSLs ($10^{-6}$/Ry <f ≤ $10^{-4}$/Ry): 1473 K

[7] Typical values of cold pool and hot pool temperature PFBR are as below [15]:

For event frequency $10^{-4}$/Ry <f < $10^{-2}$/Ry:

- Cold pool temperature: 873 K
- Hot pool temperature: 898 K

For event frequency $10^{-6}$/Ry <f ≤ $10^{-4}$/Ry:

- Cold pool temperature: 913 K
- Hot pool temperature: 923 K

### 2.3.9 Acceptance Criteria for DEC-B

The release of radioactive materials shall cause no permanent relocation of population. The need for offsite interventions shall be limited in area and time as per the requirement specified in AERB Safety Code (AERB/NF/SC/S, Rev. 1). In addition to this, the following acceptance criteria should also be met:

a) Severe accident safe state should be reached within the timelines as specified in AERB/NPP-SFR/SC/D [1].
b) Structural integrity of primary boundary shall be ensured taking into account of mechanical energy released during a Core Disruptive Accident (CDA). The subsequent re-criticality shall be avoided simultaneously ensuring cooling of core debris.
c) Containment integrity should be ensured.
d) Containment by-pass should be prevented.

# 3. GUIDANCE ON PERFORMING DETERMINISTIC SAFETY ANALYSIS

## 3.1 General

3.1.1 The safety analysis should proceed in parallel with the design process, with exchange of feedback between the two activities. The scope and level of details of the safety analysis should be commensurate with the stage of the design process so that the final safety analysis reflects the final plant design as constructed and commissioned.

3.1.2 The DSA approaches and rules provided in this chapter are applicable to licensing analysis of SFR based NPPs.

3.1.3 The plant design models and data in the plant analytical models (which are essential foundations for the safety analysis) should be kept up to date during the design phase and throughout the lifetime of the plant.

3.1.4 For licensing calculations, if at any time throughout the lifetime of the plant credible information comes to light which brings into question the conservatism of the existing analysis, re-analysis with new information should be performed and it should be shown that the acceptance criteria is met.

3.1.5 The safety analysis process should be credible, with sufficient scope, quality, completeness and accuracy to generate confidence in safety of the plant design.

## 3.2 Objectives of DSA

3.2.1 DSA should formally assess the safety of the NPP under various plant states against acceptance criteria including radiological releases.

3.2.2 DSA should assess that all levels of defense in depth are adequate to ensure the required level of safety under various plant states.

3.2.3 The DSA is carried out to:

a) understand operational transients and plant system response,

b) develop a basis for various limits and 'Limiting conditions for Operation' (LCO) to be specified in the technical specifications for operation of the plant,

c) arrive at performance requirements for design of safety systems[8],

d) demonstrate performance of additional safety features and complimentary safety features (items important to safety and design provisions of specific safety class),

e) provide support in establishing and validating accident management strategies, procedures and guidelines, aid to formulate operating procedures, EOPs, AMGs and human factor aspects,

f) confirm that modifications to the design and operation of the NPP have no significant adverse effects on safety, and

g) predict source term and doses during accident conditions for licensing analysis and to

[8] Examples are requirements for speed of actuation, and 'reactivity worth' of reactor shutdown devices; containment design parameters; design parameters and settings of safety systems. Also, whether a particular corrective action can be manual or should be automated based on how fast the action is required to be completed.

support emergency preparedness and response.

## 3.3 Safety Analysis Procedure

3.3.1 The steps involved in DSA using appropriate computer codes are illustrated below (Figure 3.1).

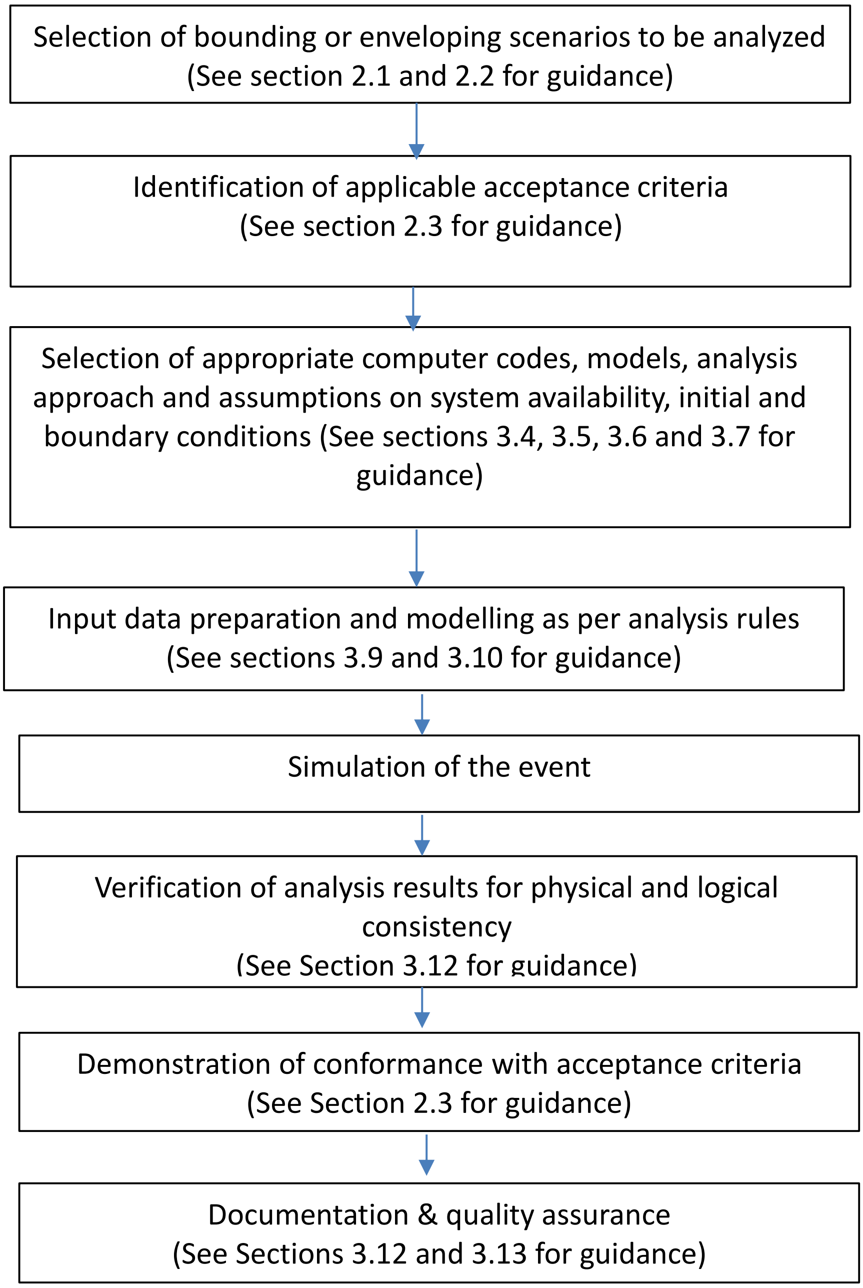


Figure 3.1: Steps Involved in Deterministic Safety Analysis

## 3.4 Computer Codes for DSA

3.4.1 Various computer codes are used for safety analysis of the NPPs. Computer codes for safety analysis broadly cover the following areas:

- Reactor physics
- Fuel behaviour
- Thermal hydraulics

- Computational fluid dynamics
- Containment analysis
- Atmospheric dispersion and Radiological Impact Assessment (RIA)
- Structural analysis
- Thermo-mechanical behavior

3.4.2 A conservative code comprises of combination of all the models necessary to provide a pessimistic estimate for a physical process relating to specified acceptance criteria.

3.4.3 A best estimate code comprises combination of the best estimate models necessary to provide a realistic estimate of the overall response of the plant during an accident. Best estimate model provides a realistic estimate of a physical process to the degree consistent with the currently available data and knowledge of the phenomena concerned. The term 'best estimate code' means that the code is free of deliberate pessimism and contains sufficiently detailed models and correlations to describe the relevant processes for the transients that the code is designed to simulate.

3.4.4 All the important phenomena identified should be represented by the models embedded in the computer code used for calculation. The suitability of models and computer code for the events to be analysed should be demonstrated. The computational/input model of the plant systems should be verified to reflect 'as-built' and 'as-operated' plant conditions. Safety analysts should ensure that the selected computer codes and their models are appropriate for their end use.

## 3.5 Assumptions on System Availability

3.5.1 Conservative assumptions on system availability: This includes applicable single failure criterion over and above the systems and component, which can be taken on maintenance as per design intent. Applicability of single failure criterion is defined in section 3.10.5.

3.5.2 Best estimate assumptions on system availability: Except system under maintenance as per design intent, application of single failure criterion may not be required.

## 3.6 Initial and Boundary Conditions

3.6.1 Conservative type of initial and boundary conditions (as used in option 1 & 2 of Table 3.1): Plant parameters, initial and boundary conditions are chosen to give a conservative result, in relation to specified acceptance criteria. This includes the error in measurement/prediction of parameter by experiment or suitable models. The complete analysis requires use of sensitivity studies to justify conservative selection of input data.

3.6.2 Best estimate type of initial and boundary conditions (as used in option 4 of Table 3.1): Plant parameters, initial and boundary conditions correspond to nominal values corresponding to operating condition.

3.6.3 Best estimate plus uncertainty for selected parameters and partly most unfavourable conditions for others (as used in option 3 of Table 3.1): Few parameters on initial and boundary conditions are selected as nominal values and for few parameters conservative values can be considered.

3.6.4 The parameters should be enveloping the entire operating range of reactor from beginning to end of operating cycle.

## 3.7 Analysis Approaches

3.7.1 Safety analyses are carried out using computer code, initial and boundary conditions, and taking credit for availability of the system in the analysis. Various approaches to carry out licensing safety analysis are given in Table 3.1 [6], [7].

**Table 3.1. Approaches for Safety Analysis**

| Option Number | Option Name | Type of computer code | Assumptions on systems availability | Type of initial and boundary conditions | Applicable category of initiating events |
|---|---|---|---|---|---|
| 1 | Conservative | Conservative | Conservative | Conservative | DBA |
| 2 | Combined-1 | Best Estimate | Conservative | Conservative | DBA |
| 3 | Combined-2 | Best Estimate | Conservative | Best estimate plus uncertainty for selected parameters and partly most unfavourable condition for others | DBA |
| 4 | Realistic | Best Estimate | Best Estimate | Best Estimate | AOO, DEC-A and DEC-B |

3.7.2 DSA performed according to options 1, 2 and 3 is considered as conservative analysis, with a decreasing level of conservatism from option 1 to 3.

3.7.3 Any option out of options 1 or 2 or 3 can be selected for licensing analysis for design basis accidents, to demonstrate that the safety systems alone in short term and with operator action in long term are capable of fulfilling fundamental safety functions and meeting the acceptance criteria of DBAs.

3.7.4 Option 3 can be termed as "Combined-2". The difference between Option 2 and 3 is that in Option 3, whenever extensive data are available, the best estimate input data is used, and whenever data are scarce, use is made of the conservative input data. The use of "best estimate" requires that the uncertainties be accounted for by a statistical combination of uncertainties. This approach is defined in section 3.11.

3.7.5 The main objective of the realistic analysis of AOOs (option 4) is to check that the control systems can prevent a wide range of AOOs from evolving into accident conditions and that the plant can return to normal operation following an AOO. Therefore, analysis of AOOs, using Option 4, should aim at providing the most possible realistic response of the plant to the initiating event. It should be demonstrated in the analysis that if the plant control and limitation systems operate as intended, they will be capable of preventing the need for actuation of the safety systems and acceptance criteria for AOOs are met. However, it is recognized that some AOOs themselves require the actuation of safety systems. For simplicity, the terms 'realistic approach' or 'realistic analysis' are used in this Safety Guide to mean best estimate analysis without quantification of uncertainties.

3.7.6 The conservative analysis for AOOs should be performed with option 1 or 2 or 3. Control systems should be credited only if their functioning would aggravate the situation. No credit should be taken for the operation of the control systems in mitigating the effects of the initiating event. In such analysis, consequences are expected to go beyond AOO acceptance criteria; however, they should be limited to DBA acceptance criteria.

3.7.7 For conservative AOO analysis, the initial and boundary conditions can be selected at their limit of Limiting Conditions of Operation (LCOs) to be specified in Station Technical Specification in such a way that they maximize the consequences of the AOO in the corresponding functional group. This would be done so that the analysis can confirm that the selection of LCO value is effective. Alternatively, the analysis results may be employed to derive a suitable value for use as an operating limit. Measurement error and accuracy of the instrumentation should be taken into account to decide conservative bound of these parameters. To ensure conservatism in boundary conditions, errors considering instrument accuracy for set point of different automated actuation logic (e.g. setback, reactor trip) should be accounted. In case of lack of clarity of conservative value of any input parameter due to counteracting effects of different phenomena, nominal value may be used.

3.7.8 For analysis carried out with option 2, when a best estimate code is used in combination with conservative inputs (initial and boundary conditions) and assumptions on system availability, it should be ensured that the uncertainties associated with the best estimate code are sufficiently compensated for by conservative inputs. The analysis should include a combination of validation of the code, use of conservatisms and use of sensitivity studies to evaluate and take into account the uncertainties relating to code models. These studies may be different depending on the type of event and therefore should be carried out for each deterministic safety analysis. Initial conditions that cannot occur at the same time in combination do not need to be considered. However, the initial conditions considered should include the most unfavorable combinations that are possible.

3.7.9 Analysis of DEC-A and DEC-B should be carried out using option 4. In addition, a systematic process involving expert engineering judgment should be used to identify potential cliff edge effects, such as fuel dry out, pressure boundary failure and inventory depletion and identify the dominant parameters by assessing their influence on the parameters of acceptance criterion. Where the likelihood is considered to be high and the potential impact is large, sensitivity analyses should be used to demonstrate to the extent practicable that, when more conservative assumptions are considered for dominant parameters, there are still margins with respect to cliff edge effect.

3.7.10 As mentioned in Table 3.1, options 1, 2 and 3 (conservative approach) is used for DSA for DBAs and Option 4 (realistic analysis – less conservative approach) is used for DECs. Following this approach, if dose estimates for DBAs are more than that resulting from DECs (resulting from the event sequence escalated from same DBA), proper explanation should be provided.

## 3.8 Computer Code Verification and Validation

3.8.1 Verification

a) The responsible organization should have mechanisms for verification of computer codes to ensure that the code correctly performs all the intended functions and does not perform any unintended function. In general, the verification of the code design should ensure that the numerical methods, the transformation of the governing equations into a numerical scheme, solution methods, model limitation and range of validity (analytical and experimental), user options and their restrictions are appropriately implemented in accordance with the design requirements. The verification of the code design should be performed by means of review, inspection and audit. Independent verification process by independent person/group of responsible organization other than the person/group involved in the development of the code should be carried out.

b) The verification of the code design should include a review of the design concept, basic logic, flow diagrams, numerical methods, algorithms and computational environment. The results of the all verification activities should be documented and preserved as a part of the system for quality management
c) If the code is run on a hardware or software platform other than that on which the verification was carried out, the continued validity of the code verification should be assessed. The code design may contain the integration or coupling of codes. In such cases, verification of the code design should ensure that the links and/or interfaces between the codes are correctly designed and implemented to meet the design requirements.
d) Comparisons with independent calculations should be carried out where practicable to verify that the mathematical operations are performed correctly. The tracking of errors and reporting of their correction status should be a continuous process and should be a part of code maintenance. The impacts of such errors on the results of analyses that have been completed and used as part of the safety assessment for a plant should be assessed. Further the impacts of corrections on the results of existing analysis and safety assessment should be brought out clearly and documented.

3.8.2 Validation

Computer code validation should be performed and documented for all computer codes that are used for the DSA of NPPs. The purpose of validation is to provide confidence in the ability of a code to realistically or conservatively, as required, predict the safety parameter(s) of interest. If the code is upgraded by changing the models of the code, appropriate validation should be carried out. Adequate documentation should be maintained to track changes in each versions of code. For validation of computer codes, combination of the following approaches as applicable are acceptable:

a) Computational checks: Checking of individual model against analytical solutions or with existing correlations derived from experimental data wherever possible.
b) Solving standard/benchmark problems.
c) Separate effect tests: These tests address specific phenomena that may occur on a nuclear power plant but do not address the other phenomena that may occur at the same time.
d) Integral effect tests: These are test cases that are directly related to a nuclear power plant. All or most of the relevant physical process are represented. However, these tests may be carried out at reduced scale, may use substitute material or may be performed with condition that are different from the plant (e.g. at low pressure, low temperature, low flow etc.).
e) Commissioning tests.
f) Operational transients: Operational transients occur either in an actual nuclear power plant or an experimental rig which represents the plant at full scale and in realistic conditions. Validation through operational transients together with NPP tests is crucial to qualify the plant model, though it is noted that data from actual operational transients are subject to measurement as available at the time of incident.
g) Inter code comparisons.

3.8.3 Computer Code Documentation: Responsible organisation should maintain documentation for each computer code used for safety analysis. Information from the code documentation may be used for facilitating review of the models and correlations employed and to ensure that the models for important phenomena are appropriate. The code documentation should also include user manual and input descriptions to ensure that user can use the software properly.

3.8.4 When performing validation against experimental data, allowance for uncertainties in the measured data should be included in the determination of the uncertainty in the computer code's predictions. In addition, the evaluation of uncertainties based on scaled experimental results should be transposed to the real power plant application, and this transposition should be evaluated and justified in assessing the overall uncertainty in the results[9].

### 3.9 Input Data Preparation for Safety Analysis

3.9.1 Authentic input data should be used for safety analysis. Appropriate reference of the source of the input data should be provided in the safety analysis report. The input data should be collected from plant design documents, technical specifications of equipment, documentation generated during the commissioning and startup of NPP, operation documents for the plant (limits and conditions, operating instructions, and record of operational regime, 'As built' plant information). It is preferred that all data necessary for the preparation of a particular computer code input deck (input file) is compiled and formalized into a single document, which can be referred to in deterministic safety analysis reports. This source of information needs to contain all necessary information, such as information on geometry, thermal hydraulic parameters, material properties, characteristics of control system and set points, and the range of uncertainties in plant instrumentation devices, including references to drawings and other permanent documents. Physical properties used in the analysis/input should be well documented and referenced and its range of applicability and dependencies on pressure, temperature, etc. should also be mentioned.

3.9.2 Nodalization schemes should be selected with sufficient details for all the important phenomena of the scenario and design characteristics of NPP under investigation. For example, adequate number of representative fuel assemblies (with adequate number of radial and axial nodes) should be modelled when such modelling is expected to influence results significantly. Important geometrical parameters, boundary conditions and initial conditions of achieved steady state should be compared with design values/nominal plant parameters and reported. It should be ensured that the effect of change of spatial size of node in final nodalization on the results of analysis is not significant. Important phenomena to be observed during different events should be verified and reported. It should also be ensured that the effect of time step on the result of analysis is negligible.

3.9.3 User Effect: The user has to make many input decisions for typical system code calculations, including the level of system nodalization; input parameters for code models and specific system characteristics and components; initial and boundary conditions; state transport properties. User effect could be reduced in the following ways:

a) by using a code which has capabilities to identify probable input errors,
b) by reducing the number of code options to be selected by code users,

[9] It is desirable to justify the scaling laws. In many cases, the reduced model experimental result cannot be scaled for prototype and in that case additional suitable analytical / experimental analysis backed with justification to use the results should be provided for its applicability at the plant level.

c) by enhancing qualification and training of users,
d) by mutual discussions among users, and
e) by providing recommendations and default values of parameters.

3.9.4 If more than one code is used for the analysis of an initiating event or event sequence then methodology used for coupling of codes should be addressed and documented in detail to facilitate users to understand information exchanged among codes and its influence on numerical convergence in each code.

### 3.10 Safety Analysis Rules

Assumptions and other considerations for the safety analysis should depend on the plant state for which analysis is being carried out. These are given below:

3.10.1 Initial and Boundary Conditions

As deliberated in section 3.6.

3.10.2 Safety Analysis Phenomena Considerations

Deterministic safety analysis for PIEs associated with reactivity changes require the solution of reactor kinetics equations (either point kinetics or space time kinetics in 1,2 or 3 dimensions) coupled with thermal hydraulics. Use of point kinetics instead of 3-D neutron kinetics should be justified on event basis for all plant states. The applicability of the solution method, its accuracy and conservatism, should be assessed and ensured.

Consideration should be given to simulation of different phenomena/aspects like, stored energy in fuel and structural components, sources of heat in fuel, heat losses from structural components, phenomena related to fuel under different conditions, thermal and flow stratification, flow reversal, thermo-syphoning, different modes of heat transfer, reactivity changes due to gas entrainment, thermal striping, simulation of different components like pump, steam generators, valves, containment thermal hydraulics etc., depending upon scenario. Consideration should also be given to DEC-B associated phenomena such as core melt, fuel melting and slumping, reactivity excursion, mechanical energy release, sodium leak to containment, sodium burning and associated aerosol behavior, behavior of non-condensables etc. Multi-physics phenomena and multidimensional aspects should be adequately simulated. Coupling between other codes like neutronics, structural, source term and codes for specific application should be appropriately accounted**.** Some of the key phenomena that are to be considered in the analysis is covered in Annexure-I.

3.10.3 Shutdown System Considerations

The analysis of AOOs, DBAs, and DECs should be conducted separately for each shutdown system, crediting the first trip parameter of respective shutdown system. Realistic analysis of AOOs carried out with the first trip parameter of the SDS-1 should meet the AOO acceptance criteria. When analysed with first trip parameter of SDS-2, utilisation of probabilistic inputs (postulated failure or non-availability of SDS-1) is acceptable, however, consequences should be limited and should remain well within those of DBAs. For DBAs, acceptance criteria of DBA should be met crediting the first trip parameter of the respective shutdown system.

3.10.4 Delay in Reactor Trip

For reactor trip to initiate, first the sensor should sense that the trip set point is reached. Thereafter there are delays in processing the information before the shutdown system actuates. For reactor trip the total instrumentation delays should be accounted, till the shutoff rods begin to fall/be inserted. This time delay in reactor trip should be considered for analysis of AOOs, DBAs and DECs.

### 3.10.5 Single Failure Criterion

The single failure criterion should be applied to each safety group or the assembly of equipment designated to perform all actions required for a particular PIE, to ensure that the limits specified in the design basis for DBAs are not exceeded [1]. For AOOs (realistic analysis) and DECs, single failure criteria may not be considered.

### 3.10.6 Control System

For analysis of DBAs, no credit[10] should be taken for the control systems provided for normal plant control in mitigating effects of the initiating event. The control system should be credited if the control action could aggravate the accident or delay the actuation of the protection features. For realistic analysis of AOOs, credit of plant controls may be taken unless the PIE itself leads to unavailability of a particular control system.

### 3.10.7 Offsite Power

For DBAs, in addition to a single failure and any consequential failures, a loss of offsite power should be assumed if it has unfavorable results. The loss of offsite power should be assumed at the time of initiation of the event or it can be assumed at a time such that the timing of initiation of shut-off rod movement due to loss of offsite power coincides with that due to the initiating event; whichever is conservative. Additional assumption of loss of offsite power along with PIEs may not be required for the analysis of AOOs (realistic analysis) and DECs.

### 3.10.8 Consideration of Systems

a) For analysis of AOOs and DBAs, credit for equipment and systems for mitigation can be taken only if such equipment and systems are designed for the environmental conditions expected to be prevailing during the event.
b) For analysis of AOOs and DBAs, credit of systems/features provided exclusively for mitigation of DECs should not be taken.
c) For analysis of DECs, items important to safety that are not affected by the failures assumed in DEC sequence and shown to survive under these conditions for the period that is necessary to perform their intended functions, may be credited..
d) For conservative assumptions on availability of the systems for analyses of all plant states, minimum allowed configuration of equipment and system as per limiting condition for operation of Technical Specifications should be considered. This consideration is over and above the single failure criterion.
e) For analyses of all plant states, any process equipment that is operating prior to the event is assumed to continue operating, if it is not affected by the initiating event itself and by its consequences.

[10] This could be modelled in different ways e.g. considering control signals frozen at pre-accident condition, not considering any control action, as if control systems do not exist etc. In any case, in analysis of DBAs, ‘not crediting control systems’ should not result in any beneficial effect on accident progression.

f) In the analysis of DEC-A and DEC-B, the credit of non-permanent equipment[11] should not be considered in demonstrating adequacy of NPP design.

3.10.9 Decay Heat

For DBAs, decay heat enveloping all fuel burn-ups should be considered; whereas for analysis of other plant states, decay heat corresponding to average maximum possible burn-up can be considered.

3.10.10 Operator Action Time

Safety analysis of the plant should take proper account of potential human errors in operational states and accident conditions. The time available for operator actions should be considered from the first clear and unambiguous indication of the necessity for operator actions. Operator actions should be as follows:

a) Credit for operator action should not be considered earlier than 20 minutes (if actions are taken from main control room) [1].
b) Credit for operator action should not be considered earlier than 30 minutes (if actions are taken from the field) [1].
c) Safety analysis should take into account that the credit for such operator intervention ('a' and 'b' above) is acceptable only if the:
    i) design can demonstrate that the operator has sufficient time to decide and to act,
    ii) necessary information and instruction on which the operator must base a decision to act is simple and unambiguously presented,
    iii) physical environment following the event is acceptable in the control room or in the supplementary control room/backup control points, and
    iv) access route to that supplementary control room/backup control points, is available.
d) Action from supplementary control room should be counted as field action.
e) In certain exceptional circumstances, which must be justified, an operator action shorter than 20 minutes for main control room action might be assumed, provided that:
    i) the operator is exclusively focused on the action in question;
    ii) the required action is unique, and does not involve choice from several options; and
    iii) the required action is simple and does not involve multiple manipulations.

### 3.11 Analysis approach for Combined 2 (Best Estimate Plus Uncertainty)

Uncertainties in deterministic safety analysis, in particular for DBAs, need to be addressed when Option 3 is adopted (best estimate computer codes are used in combination with best estimate initial and boundary conditions and availability of systems is assumed in a conservative way) (Table 3.1). To achieve conservative safety analysis, uncertainties should be identified and assessed to confirm that the actual plant parameters will be bounded by the results of calculation plus uncertainty with an adequate confidence. There are three potential sources of uncertainties [8]:

3.11.1 Plant uncertainty: Uncertainty in the parameters used in measuring or monitoring or

[11] The definition of non-permanent equipment as given in AERB/NPP-SFR/SC/D and associated guidance provided in AERB/NPP/SG/D-1 (Rev.1) (Draft) on "Safety Classification and Seismic Categorization for Structures, Systems and Components of Nuclear Power Plants" should be considered.

representing a real plant which has significant effect on the acceptance criteria should be accounted for, such as reference plant parameters, instrument error, set points, instrument response. Typical examples are the conductivity of the fuel, gap between the pellets and the cladding/gap conductance, decay heat, primary pressure, secondary pressure etc. Some of the considerations to account plant uncertainty is indicated below:

a) For the uncertainties associated with input parameters, the preferred means is the collection of NPP data of initial and boundary conditions that are relevant to the events being considered and based on these data obtain a probabilistic distribution.

b) Uncertainties associated with input parameters are obtained by performing a sufficient number of calculations varying these input uncertain parameters and monitoring the output parameters of relevance. Because there are several plant parameters, one must first identify the sensitive ones (those which affect in a major way the parameters which are used in the acceptance criteria) and the uncertain input parameters should include the most significant ones. The selected input parameters have to be independent or dependencies between uncertain input parameters should be identified and quantified and a specific processing should be applied.
The selected input parameters should be ranged and their probability distribution specified using relevant experiments, measurements of parameters, records of plant operational parameters etc. If probability distribution for certain parameters cannot be quantified, "partly most unfavorable" approach (option 3 of Table 3.1) may be followed where conservative values from the given range should be selected for such parameters and for the remaining parameters (low sensitive ones) nominal values can be chosen.

c) The selection of uncertain input parameters, their ranges and probability distributions is crucial for the reliability of results, since it strongly affects the width of the uncertainty bands of the results that is essential for engineering applications.

d) Overall quantification of uncertainties should be based on statistically combined uncertainties in plant conditions and code models to ensure, with a specified probability, that a sufficiently large number of calculated results meet the acceptance criteria.

3.11.2 Representation or simulation uncertainty: Uncertainty in representing or idealizing the real plant, such as that due to the inability to model a complex geometry accurately, three dimensional effects, scaling, control and system simplifications (e.g. modelling few assemblies instead of all the assemblies, radial and axial subdivision in the nodalization scheme etc.). Some of the considerations to account simulation uncertainty are indicated below:

a) The amount of uncertainty introduced by the necessary simplifications in modelling a real plant can be estimated by performing a sensitivity study in which the simplification introduced in the model is reduced in a stepped manner. One then obtains a measure of the uncertainty in the results introduced by the simplification.

b) Results produced by computer codes are sensitive to decisions that are made by the user, such as the number and structure of nodes that are used. Such user effects could be particularly large for a specific analysis. The procedures, code documentation and user guidelines should be carefully followed to limit the user effects.

3.11.3 Code or Model Uncertainty: This includes uncertainty associated with the models and correlations, the solution scheme, model options, un-modelled processes (processes that are not modelled) and data libraries. Some of the considerations to account model uncertainty are indicated below:

a) Validation of the code should be performed to assess the uncertainty of values predicted by the code. Outputs of the code are compared with relevant experimental data and with operational transients, if possible, for the important phenomena expected to occur. The code accuracy obtained as the result of validation work should be used as a source for uncertainties of relevant modelling parameters. It is important to focus the end point of the uncertainty analysis on parameters which, are used directly in comparison with acceptance criteria, for example, the clad hotspot temperature, the radiological dose to the public and the peak containment pressure. The purpose of uncertainty analysis is not to quantify the uncertainty in every prediction, but only in the parameters used directly in the comparison with acceptance criteria.
b) Uncertainties are deemed to be accounted for in a code intended to be conservative regarding certain acceptance criterion. In that case, it should be demonstrated that the code prediction is conservative when compared against the experimental data.

## 3.12 Presentation and Evaluation of Results

3.12.1 The results of safety analysis should be structured and presented in an appropriate format in such a way as to provide a good understanding and interpretation of the course of the accident. The presentation of the results should be sufficiently complete to allow the entire process to be displayed, starting from the initial steady state up to the long term safe stable condition. The presentation of accident analysis results should contain those parameters reflecting the key phenomena expected to occur in the course of the transient or accident. The format of the results needs to be such as to allow an inter comparison with the results obtained from the same or different codes.

3.12.2 For presenting deterministic safety analysis in preliminary/final safety analysis report, format given in AERB Safety Guide on 'Standard Format and Contents of Safety Analysis Report for Nuclear Power Plants' (AERB/NPP/SG/G-9) [9] should be followed.

3.12.3 Before submitting safety analysis to AERB, the responsible organisation should ensure that an independent verification of the safety analysis is performed by individuals or groups separate from those carrying out the original analysis.

3.12.4 Review of Deterministic Safety Analysis Results

a) Before any use of the results, their correctness needs to be carefully checked. This could be done on the basis of user experiences and logical judgment, comparison with similar calculations, sensitivity analysis and consistency with general findings. The results derived should be reviewed and evaluated in relation to the initial goal and purpose of the analysis, such as licensing, improvement of operational documentation or plant upgrading.
b) The prime objective of reviewing the results is to check by comparison of calculated values with criteria whether the acceptance criteria have or have not been satisfied. If the analysis is used for evaluation of the system safety performance, the review and discussion of the results needs to be focused on the safety functions and the status of the physical barriers.
c) A certain amount of attention should be devoted in the discussion of the results to their sensitivity to the key input parameters as well as to the expected uncertainties and the tolerance band of the parameters, if the analysis is not considered conservative and analysis results are very close to the acceptance criteria.

The review of the results may lead to a specification of the additional analysis and the resolution of the relevant safety issues (if necessary).

3.12.5 Update of Safety Analysis

a) The objective of the update of safety analysis is to check the extent of validity of existing safety analysis taking into account the actual plant status, expected degradation till the next update of safety analysis or the end of predicted life and current analytical methods, safety standards and knowledge.
b) Overall safety analysis of the NPP should be reviewed and updated as required to ensure that the NPP does not pose any undue hazard to the surrounding. During review, it should be ensured that the actual state of the plant including modifications are considered. In addition, the completeness of the list of events should be checked. Current analytical methods including computer codes should be used wherever re-analysis is required.
c) Accepted rules for analysis, operator action, common cause failures, redundancy, diversity, separation, etc. should be used. Required modification in any input data should be incorporated based on plant operation and operational feedback.
d) A revision of the safety analysis should be made on the basis of:
   i) feedback from operational experience, the findings of periodic safety reviews, regulatory requirements,
   ii) changes to the applicable rules and regulations,
   iii) advances in knowledge and improvements in technology,
   iv) modernisation of the plant,
   v) changes in the described plant configuration as implemented,
   vi) changes in operating procedures due to operational experience, and
   vii) up-rating of the reactor power, use of improved types of fuel and innovative principles for core reloads.

## 3.13 Quality Assurance in Deterministic Safety Analysis

3.13.1 Accident analysis should be subjected to comprehensive quality assurance programme, applied to all activities affecting the quality of the final results.

3.13.2 Formal quality assurance procedures and/or instruction need to be developed and reviewed for the whole accident analysis process, including:

a) collection and verification of plant data,
b) verification of the computer input file/deck, and
c) validation of plant models.

3.13.3 It is helpful to have an approved document on the method of analysis prior to performing an analysis. Such a document lists the models to be used, system assumptions, acceptance criteria and system nodalization.

3.13.4 The responsibilities of all individuals in the organisation involved in the analysis need to be clearly specified. Safety analysts need to be trained and qualified. All documents, including calculation notes and results, need to be recorded to allow them to be independently checked by qualified reviewers. Validated and accepted methods and tools need to be used, and their uses need to be referenced. All sources of data should be clearly referenced and documented.

3.13.5 The result should be checked using one or more of the following techniques depending on the importance of the analysis.

a) Peer review
b) Independent review by competent individuals
c) Independent calculation (on sample basis) by a competent individual

3.13.6 All safety analyses used for plant licensing need to be archived so that the code version, code documentation, input data and calculation results are recoverable.

# 4. SOURCE TERM ESTIMATION AND DOSE EVALUATION

## 4.1 General

4.1.1 Source term (ST) is the amount and isotopic composition of radioactive material released (or postulated to be released) from a facility to environment.

4.1.2 To evaluate the source term from a nuclear power plant, it is necessary to know the sources of radiation, the inventories of radionuclides and the mechanisms by which these get transported through different barriers and released to the environment.

4.1.3 Estimation of the behaviour of fission products, radioactive corrosion products, activation products in coolant and impurities, and actinides following possible accidents at the NPP should be carried out early in the design stage. This is required to identify all important phenomena that affect source term behaviour and to identify the possible design features that could increase their retention in the plant. The evaluation, before a plant is operated, of the source terms for operational states should include all the radionuclides that, owing to either liquid discharges or gaseous discharges, may make a significant contribution to doses.

## 4.2 Source Term Estimation

4.2.1 The radionuclides release from various barriers and its treatment can be addressed in following ways depending on the requirements i.e. (a) radionuclides release to containment (b) radionuclides in containment (c) radionuclides release to outside environment. Fuel failure criteria for the estimation of source term should be considered and detailed guideline on the source term and its modelling are given in AERB Safety Manual on “Methodology for Radiological Impact Assessment of Nuclear Power Plants under Postulated Accident Conditions” (AERB/NPP/SM/RIA-1) [10].

4.2.2 Source term should be evaluated for operational states and accident conditions for the following reasons:

a) To ensure that the design is optimised to reduce the source term (hence the dose to the public) to a level that is as low as reasonably achievable.

b) To demonstrate that the design ensures that requirements for radiation protection, and radiological consequences, are met.

c) To provide a basis for the emergency planning arrangements that are required to protect the public in the vicinity of the NPP.

d) To demonstrate that the qualification of equipment that are required to survive, including instruments and gas treatment systems, is adequate.

e) As input to the software for use in emergency planning that employs estimated source term of postulated event to provide an early indication of what emergency measures are required. This allows decisions to be made early, before measurements of the activity levels of released radioactive material outside the plant can be made.

4.2.3 The source term should be evaluated for the bounding scenario(s) in each plant state including severe accidents. Bounding scenario(s) should be selected in each plant state based on different radioactive material release pathways and functional groups. The evaluation of source term should include a comprehensive analysis of PIEs/ or event sequences in which the release of radioactive material would occur outside the containment. This exercise ensures that the design is optimised so that requirements for radiation protection, including restrictions on doses, are being met.

4.2.4 Radionuclides Release to Containment

a) Release to containment should consider the magnitude, composition, chemical kinetics during transport, physical and chemical form and timing of the release of fission products and other aerosols from core as a result of a reactor accident. The different mechanisms of releases from core are given below
   - Gap release on sheath failure
   - Diffusional release on heating
   - Grain boundary sweeping on oxidation in presence of steam
   - Melt release and
   - Vaporization release

   Activity present in the coolant at the time of the event should also be taken into account.

b) The methodology to be adopted to calculate source term release from the fuel or to containment should consider the effects of operating conditions, nuclide properties, thermo-mechanical behaviour of the fuel and distribution of the fission products within the core at equilibrium core conditions.

c) Appropriate consideration should be given to account the retention of radionuclide in primary, secondary circuit etc. as applicable before its release to primary containment, secondary containment or directly to the environment.

4.2.5 Radionuclides in Containment: The magnitude, composition, characteristic, physical and chemical form of the radionuclides and aerosols which are airborne in the primary containment environment with time should be considered. All relevant attenuation processes (plate-out, washout, decay etc.) inside containment should be modelled.

4.2.6 Radionuclides Release to Outside Containment: The magnitude, composition, characteristic, physical and chemical form of radionuclides and aerosols that are leaked out of containment to the outside environment should be considered.

4.2.7 The evaluation of source term should also include a comprehensive analysis of postulated accidents in which the release of radioactive material would occur directly outside the containment. Accidents in which the release of radioactive material could bypass the containment form a very important category, because a bypass accident with a relatively small release of radioactive material from the fuel may have the same radiological consequences as an accident with a large release into the intact containment.

## 4.3 Dose Evaluation

4.3.1 The dose should be evaluated for the bounding scenario(s) in each plant state including severe accidents. Dose evaluation can be carried out by using either conservative or realistic analysis. The applicant should summarise the assumptions (stability class, metrological data, atmospheric dispersion model, ground deposition, dose conversion factors, protection factor/shielding, annual breathing rate, water intake and default dietary intake data, treatment for complex and coastal terrain for the evaluation of concentration of radionuclides (if applicable) etc.) and calculation methods used to determine the doses that result from operational states and accident conditions should be provided. The annual release of radioactive material to the environment can be evaluated by using an average value for the activity of the primary coolant. Values for the effect of spiking on the activity of the primary coolant due to applicable operational transient should be considered based on relevant operational data. Sufficient information should be provided such that an independent analysis can be performed.

4.3.2 All radionuclides which have considerable contribution on dose should be accounted. For the dose calculation, all pathways (ingestion, inhalation, cloud immersion, skin dose, plume, ground shine etc.) should be considered. Details on the estimation of dose and its modelling are given in AERB Safety Manual on ‘Methodology for Radiological Impact Assessment of Nuclear Power Plants under Postulated Accident Conditions’ (AERB/NPP/SM/RIA-1) [10] and AERB Safety Manual on Methodology for Radiological Impact Assessment for Public Dose Computation and Dose Apportionment during Operational States of Nuclear Facility (AERB/NF/SM/RIA-2) (Draft) [11].

4.3.3 In presenting the assumptions and methodology used in determining the radiological consequences, it should be ensured that analyses are adequately supported with backup information, either by reporting the information or by citing references where appropriate.

# 5. APPLICATION OF DETERMINISTIC SAFETY ANALYSIS

## 5.1 General

5.1.1 DSA should be carried out for:

a) design of nuclear power plants,
b) licensing of nuclear power plants,
c) providing inputs for probabilistic safety analysis,
d) development of emergency operating procedures and accident management guidelines,
e) analysis of events occurred at nuclear power plants,
f) review and refinement of safety analysis as part of periodic safety review,
g) review and assessment of modifications in NPPs, and
h) predicting source term and doses for accident conditions to support emergency preparedness and response (the details are provided in Chapter 4).

## 5.2 Design of Nuclear Power Plants

5.2.1 Deterministic safety analysis should be used iteratively with the design process for meeting the requirements/acceptance criteria stipulated for design of NPPs. The design basis for items important to safety and design provisions should be established and confirmed by means of comprehensive safety assessment through both deterministic and probabilistic safety analyses. With reference to the deterministic safety analysis, applicability of the analytical assumptions, methods and degree of conservatism used should be verified. The design requirements for structures, systems and components important to safety must be met for safe operation of a nuclear power plant, and for preventing or mitigating the consequences of events that could jeopardize safety.

5.2.2 For establishing the robustness of the control systems in the design, governing AOOs in each functional group should be evaluated with the consideration of conservative initial and boundary conditions or can be justified with available margin with respect to acceptance criteria in the realistic analysis. The objective of this assessment is to confirm the robustness of the control system to overcome the uncertainty related to the selected initial and boundary conditions.

5.2.3 Trip Coverage Analysis: AOOs & DBAs should be analyzed realistically crediting second trip parameter for each shutdown system separately. For DBAs, acceptance criteria of DBA should be met. For AOOs, any deviation from acceptance criteria could be accepted with adequate justification based on probabilistic considerations. However, the acceptance criteria should not exceed that of DBAs

## 5.3 Licensing of Nuclear Power Plants

5.3.1 DSA carried out for licensing of NPP should be used for showing compliance with applicable regulations and standards and other relevant safety requirements. This should be presented to AERB through safety analysis reports as per AERB Safety Guide on 'Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants' (AERB/NPP/SG/G-9) [9].

5.3.2 The safety analysis for licensing purpose should examine

a) All planned modes of the plant in normal operation
b) Plant performance in AOOs

c) Design basis accidents
d) Event sequences that may lead to DEC-A and DEC-B

5.3.3 On the basis of this analysis, the robustness of the engineering design in performing its safety functions during Design Basis Events (DBEs) should be established. In addition, the effectiveness of the items important to safety should be demonstrated, and guidance for emergency response should be provided.

5.3.4 Analyses should be performed for transients that can occur in all planned modes of the plant in normal operation, including operations during shutdown. For the shutdown mode, the main objectives of the analysis are to evaluate the ability of plant systems to perform safety functions and to determine the time available for the operators to establish safety functions, considering the likely configuration of systems and equipment in shutdown state.

5.3.5 The range of scenarios should be evaluated to determine whether abrupt changes in the results of the analysis occur for a realistic variation of inputs (usually termed cliff edge effects).

### 5.4 Providing Inputs for Probabilistic Safety Analysis

5.4.1 The DSA for providing inputs to PSA should be performed using option 4 (Table 3.1).

5.4.2 DSA should be carried out to determine the 'success criteria' of the required systems and time available for operator actions for use in probabilistic safety assessment. The DSA should support PSA to identify, for various combinations of equipment failures and human errors, a minimum set of safety features that can prevent nuclear fuel degradation and minimize release of radioactive materials from the containment

### 5.5 Development of Emergency Operating Procedures and Accident Management Guidelines

5.5.1 DSA should be performed to confirm the strategies that have been developed to restore normal operating conditions at the plant following transients due to AOOs and accident conditions using option 4 ( Table 3.1). These strategies are reflected in the EOPs that define the actions that should be taken during such events.

5.5.2 DSA is required to provide the input that is necessary to specify the operator actions including time available for the actions to be taken in response to some accidents, and the analyses should be an important element of the review of accident management strategies. In the development of the recovery strategies, to establish the available time period for the operator to take effective actions, sensitivity calculations should be carried out on the timing of the necessary operator actions, and these calculations may be used to optimise the procedures.

5.5.3 After the EOPs have been developed, a validation analysis should be performed. This analysis is usually performed by using a simulator. The validation should confirm that a trained operator can perform the specified actions within the time period allowed and that the reactor will reach a safe end state.

5.5.4 When the predictions of a computer code that has been used to support or to verify an EOP do not agree with observed plant behaviour during an event, the code and the procedure should be reviewed. Any changes that are made to the emergency operating procedure should be consistent with the observed plant behaviour.

5.5.5 DSA should also be performed to assist the development of the strategy that an operator should follow if the EOPs or preventive guidelines fail to prevent a severe accident from occurring. The analyses should be carried out by using one or more of the specialised computer codes that are available to model relevant physical phenomena.

5.5.6 The analyses should be used to identify what challenges can be expected during the progression of accidents and which phenomena will occur. They should be used to provide the basis for developing a set of guidelines for managing accidents and mitigating their consequences.

5.5.7 The analysis should start with the selection of the accident sequences that, without intervention by the operator, would lead to core damage. A grouping of accident sequences with similar characteristics should be used to limit the number of sequences that need to be analysed. Such grouping may be based on several indicators of the state of the plant: the postulated initiating event, the shutdown status, the coolant pressure boundary, the secondary heat sink, the system for the removal of containment heat and the containment boundary.

5.5.8 The measures can be broadly divided into preventive measures and mitigatory actions. Both categories should be subjected to analysis.

5.5.9 Preventive measures are recovery strategies to prevent core damage. They should be analysed to investigate what actions are possible to inhibit or delay the onset of core damage. Conditions for the initiation of the actions (criteria for stopping the actions or changing to another action) should be specified.

5.5.10 Mitigatory measures are strategies for managing severe accidents to mitigate the consequences of significant core degradation. Possible adverse effects that may occur as a consequence of taking mitigatory measures should be taken into account, such as pressure spikes, return to criticality and thermal shock.

5.5.11 For some DEC-A and DEC-B, non-permanent equipment are typically considered to operate for long-term sequences and is assumed available in accordance with the emergency operating procedures or accident management guidelines. The time claimed for the availability of non-permanent equipment should be justified.

5.5.12 Additional guidance on analysis related to accident management may be derived from AERB Safety Guide on ‘Accident Management Programme for Water Cooled Nuclear Power Plants’ (AERB/NPP/SG/D-26) [12] applicable for water cooled NPPs.

## 5.6 Analysis for Emergency Preparedness and Response

5.6.1 Safety principle 9 mentioned in AERB Safety Code on Design of SFR [1] requires that arrangements must be made for emergency preparedness and response for nuclear or radiation incidents. It further mentions that, on the basis of the safety analysis, the inputs (prerequisites) for emergency planning shall be established.

5.6.2 Hazard Assessment is carried out as part for making adequate arrangements for the emergency preparedness and response that are commensurate with the hazards identified and the potential consequences of an emergency. Events that could affect the facility or activity, including events of very low probability (AERB/NPP-WCR/SG/D-5 Rev.1) and events not considered in the design (e.g. practically eliminated event) are also to be considered for the analysis for emergency preparedness and response. This will result in the knowledge on composition of the source, magnitude of the areas that are likely to be affected, requirements on the resources and means to reduce the consequence based on meteorological condition, topography of the area and demographic attributes.

5.6.3 Safety analysis related to emergency preparedness and response covers, analysis for identification of emergency situations identified as initiating conditions (ICs) for classification of emergency, estimation of source term to assess the potential consequences, establishment of generic criteria/operational interventional level (OIL) and identification of range of protective and other actions that might be warranted as part of protection strategy.

5.6.4 The analysis should be carried with realistic methodology (option 4 of Table 3.1). The results obtained using this methodology should be interpreted with consideration of the information and data available along with associated uncertainties. Single failure criterion, additional consideration of loss of off-site power and ignoring of first reactor trip parameter may not be taken. When large uncertainty is observed with available input data, conservative assumptions are acceptable for preparedness planning. In place of realistic methodology, conservative methods can also be applied for addressing its implications of conservatism on protection strategy and associated emergency arrangements.

5.6.5 Along with source term estimation, assessment should be carried out for the inventory of radionuclides of the release (radionuclides mix) and distribution of radioactive material (dispersion and deposition).

## 5.7 Analysis of Events Occurred at Nuclear Power Plants

5.7.1 DSA may be used as a tool for obtaining a full understanding of events that occur during the operation of NPPs and should form an integral part of the feedback from operating experience. Operational events may be analysed with the following objectives:

a) To check the adequacy of the selection of PIEs;
b) To determine whether the transients that have been analysed in the safety analysis report bound the event;
c) To provide additional information on the time dependence of the values of parameters that are not directly observable using the plant instrumentation;
d) To check whether the plant operators and plant systems performed as intended;
e) To check and review EOPs;
f) To identify any new safety issues and questions arising from the analyses;
g) To support the resolution of potential safety issues that are identified in the analysis of an event;
h) To assess the severity of possible consequences in the event of additional failures (such as severe accident precursors);
i) To validate and adjust the models in the computer codes that are used for analyses and in training simulators.

5.7.2 The analysis of operational events should be carried out using option 4 of Table 3.1. Actual plant data should be used. If there is a lack of detailed information on the plant state, sensitivity studies (with the variation of all the identified critical parameters) should be performed.

5.7.3 The evaluation of safety significant events is a very important aspect of the feedback from operating experience. State-of-the art best estimate computer codes should be used to investigate and to gain a detailed understanding of plant behaviour. Conclusions from such analyses should be incorporated into the plant procedures that address the use of feedback from operating experience.

## 5.8 Review and Refinement of Safety Analysis as part of Periodic Safety Review

5.8.1 The extent of validity of the existing safety analysis taking into account the actual plant status including aging effect, current analytical methods/computer codes, safety standards and knowledge should be checked. Based on the review, new deterministic analyses may be required to refine previous safety analyses, considering above.

### 5.9 Review and Assessment of Modifications in Nuclear Power Plants

5.9.1 NPP may be modified on the basis of feedback from operating experience (including a major event occurred at any NPP anywhere), the findings of periodic safety reviews, regulatory requirements and advances in knowledge or developments in technology. The modification of existing nuclear power plants may be undertaken to counteract the ageing of the plant, to justify the continued operation of the plant, to take advantage of developments in technology or to comply with changes to the applicable rules and regulations. A revision of the safety analysis of the plant design should be made when

a) major modifications or modernisation programmes are implemented.
b) advances in technical knowledge and understanding of physical phenomena are made.
c) changes in the described plant configuration are implemented.
d) changes in operating procedures are made owing to operating experience.

5.9.2 Other important applications of DSA are aimed at the optimum utilisation of the reactor and the nuclear fuel. Such applications encompass uprating of the reactor power, the use of improved types of fuel and the use of innovative/improved methods for core reloads for optimum utilization of fuel. DSA for such applications should be used for checking safety margins to operating limits, and it should be ensured that the limits are not exceeded.

5.9.3 Changes that require significant plant modifications, such as power uprating and achieving higher burnup, longer fuel cycles and life extensions, should be addressed by comprehensive deterministic safety analysis to demonstrate compliance with acceptance criteria.

## APPENDIX-I: Typical List of Events

| Sl. No. | Event | Category |
|---|---|---|
| 1.0 | **Increase in heat removal from system** | |
| 1.1 | Acceleration of one or both PSPs-On low power | 2 |
| 1.2 | Acceleration of one or both SSPs-On low power | 2 |
| 1.3 | Loss of heating in one of the feed-water heaters | 2 |
| 1.4 | Loss of heating in more than one feed-water heaters | 2 |
| 1.5 | Sudden increase in feed-water flow in one or both loops-On low power | 2 |
| 1.6 | Steam flow increase to Turbine (malfunction of the steam pressure controller) | 2 |
| 1.7 | False actuation of SGDHR | 2 |
| 1.8 | Compensable leak in main-steam line | 2 |
| 1.9 | Spurious opening of the SG depressurisation valve | 2 |
| 1.10 | Medium/Large Size break in steam line outside the containment[#] | 3 |
| 2.0 | **Change in primary or secondary coolant inventory** | |
| 2.1 | Minor sodium leaks | 2 |
| 2.2 | System malfunction resulting in increase of the Secondary coolant inventory | 2 |
| 2.3 | Failure of one tube in a IHX | 2 |
| 2.4 | Main vessel leak[#] | 3 |
| 2.5 | Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant system. [#] (Applicable for loop type reactors) | 3 |
| 3.0 | **Anomalies in reactivity and power distribution in the reactor core** | |
| 3.1 | Continuous withdrawal of one CSR-Pre-critical | 2 |
| 3.2 | Continuous withdrawal of one CSR-Low power | 2 |
| 3.3 | Continuous withdrawal of one CSR-High power | 2 |
| 3.4 | Drop/Insertion / Drift in of one or group of reactivity devices | 2 |
| 3.5 | Errors during loading or unloading of core SA-Cold Shutdown | 2 |
| 4.0 | **Decrease in heat removal from system** | |
| 4.1 | One SSP trip | 2 |
| 4.2 | Spurious dumping of one secondary sodium loop- On power | 2 |
| 4.3 | Spurious closure of a sodium side isolation valve in one SG module | 2 |
| 4.4 | Spurious closure of the feed-water isolation valve in one SG module | 2 |
| 4.5 | Spurious closure of the steam side isolation valve in one SG module | 2 |

| | | |
|---|---|---|
| 4.6 | Loss of external electric load | 2 |
| 4.7 | Turbine trip | 2 |
| 4.8 | Loss of one condenser CCWP | 2 |
| 4.9 | One CEP trip with the standby not starting | 2 |
| 4.10 | One BFP trip with the standby not starting | 2 |
| 4.11 | Both BFPs and the standby BFP trip | 2 |
| 4.12 | Loss of all the condenser CCWP | 2 |
| 4.13 | Loss of all the CEP | 2 |
| 4.14 | Loss of feed-water flow in one or both loops | 2 |
| 4.15 | Loss of condenser vacuum | 2 |
| 4.16 | Small leak in feedwater line | 2 |
| 4.17 | Complete closure of one Intermediate Heat Exchanger (IHX) sleeve valve | 3 |
| 4.18 | One Secondary Sodium Pump (SSP) seizure | 3 |
| 4.19 | Feed water pipeline break[#] | 3 |
| 4.20 | Inadvertent closure of the main steam isolation valve | 2 |
| 4.21 | Decrease in grid frequency ( high rate or low rate of frequency drop) | 2 |
| 4.22 | Secondary Sodium Pump shaft break | 3 |
| 5.0 | **Decrease in Primary Coolant Flow rate** | |
| 5.1 | One PSP trip | 2 |
| 5.2 | Partial blockage in a fuel SA | 2 |
| 5.3 | Class IV power failure during normal operation | 2 |
| 5.4 | One Primary Sodium Pump (PSP) Seizure | 3 |
| 5.5 | Design basis blockage in a fuel SA | 3 |
| 5.6 | Design basis blockage in a blanket fuel SA | 3 |
| 5.7 | Design basis blockage in a control SA | 3 |
| 5.8 | Design basis blockage in a stored fuel SA | 3 |
| 5.9 | One primary pipe rupture[#] (applicable for Pool type reactors) | 3 |
| 5.10 | Primary Sodium Pump shaft break | 3 |
| 6.0 | **Radioactive release from a sub-system or a component** | |
| 6.1 | Failure of cover gas purification system. | 2 |
| 6.2 | Large leak of argon cover gas | 2 |
| 6.3 | Leak or failure of the system containing radioactive liquid | 2 |
| 6.4 | Fuel handling accidents during transfer to spent fuel storage bay | 2 |
| 6.5 | Compensable leak of spent fuel pool lining | 2 |
| 6.6 | Spent fuel subassembly failure during handling | 3 |
| 6.7 | Drop of a core SA-Cold Shutdown[#] | 3 |
| 7.0 | **Sodium water reaction related events** | |
| 7.1 | Small sodium water reaction (SWR) in a SG module | 2 |
| 7.2 | Small sodium water reaction (SWR) + Failure of SSP trip or speed reduction | 2 |

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| 7.3 | Small sodium water reaction (SWR) + Failure of steam side isolation | 2 |
| 7.4 | Small sodium water reaction (SWR) + Failure of water side isolation | 2 |
| 7.5 | Large sodium water reaction (SWR) in a SG module (design basis leak) - On power | 3 |
| 8.0 | **Events to be analysed for Containment Analysis** | |
| 8.1 | Disturbance of heat removal from the containment | 2 |
| 8.2 | Malfunction or inadvertent operation of the system resulting into containment pressure decrease / increase | 3 |
| 9.0 | **Fuel Pool related Accidents** | |
| 9.1 | Failures of cooling in Spent Fuel Storage Bay (SFSB) | 2 |
| 9.2 | Compensable leak of spent fuel pool lining | 2 |
| 10.0 | **Malfunction of support/auxiliary systems** | |
| 10.1 | Failure of main vessel - safety vessel differential pressure control system | 2 |
| 10.2 | Fuel failure leading to Delayed Neutron Detector (DND) SCRAM | 2 |
| 10.3 | Failure of first or second stage turbine bypass valve to open following turbine trip | 2 |
| 10.4 | I&C related spurious shutdowns | 2 |
| 10.5 | Failure of computer based systems important to safety | 2 |
| 10.6 | Failure of one division of any of the various plant services | 2 |
| 10.7 | Loss of one SGDHR circuit during reactor on power | 2 |
| 10.8 | Failure of primary sodium purification system-Any | 2 |
| 10.9 | Failure of secondary sodium purification system-Any | 2 |
| 10.10 | Loss of instrument air system. | 2 |
| 10.11 | Loss of argon cover gas pressure in the secondary sodium system | 2 |
| 10.12 | Oil entry into primary sodium | 3 |
| 10.13 | Total loss of cooling system of roof slab | 3 |
| 10.14 | Total loss of cooling system of rotatable plugs | 3 |
| 10.15 | Failure of core component handling machine (stuck fuel SA) – Cold Shutdown | 3 |
| 10.16 | Loss of fuel SA decay heat removal system during handling – Cold Shutdown | 3 |
| 10.17 | Loss of on-site electrical power supply buses (class-III,II or I: one at a time) | 3 |
| 11.0 | **Design Extension Condition without core melt (DEC)** | |

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|---|---|---|
| 11.1 | Station Black Out (including effects on fuel pool cooling) | 4A |
| 11.2 | Loss of both secondary sodium loops-On power | 4A |
| 11.3 | Unconfined sodium-air-water reaction in SG building of one secondary sodium loop initiated by leak in outermost row of SG tubes, which impinges and penetrates SG shell. | 4A |
| 11.4 | Fuel handling failure in transit coupled with containment impairment characterised by<br>i. Failure of one set of containment isolation dampers or<br>ii. Failure of containment isolation logic or<br>iii. One door of main airlock stuck open and seals on second door deflated | 4A |
| 11.5 | Steam line break of largest size outside containment with single SG tube rupture. | 4A |
| 11.6 | Inadvertent closing of steam isolation valve with stuck open relief valves | 4A |
| 12.0 | **Design Extension Condition with core melt (Severe Accidents)** | |
| 12.1 | Loss of steam water system without SCRAM (ULOHS) | 4B |
| 12.2 | Loss of Class IV power without SCRAM (ULOF) | 4B |
| 12.3 | Continuous withdrawal of one control rod without SCRAM (UTOP) | 4B |
| 12.4 | Total and instantaneous blockage of a fuel SA | 4B |

#Events with low probability of occurrence ($10^{-6}$/Ry <f ≤ $10^{-4}$/Ry)
Categories indicated in above table are as below:

- Category 2: Anticipated Operational Occurrences
- Category 3: Design Basis Accidents
- Category 4A: Design Extension conditions without core melt
- Category 4B: Design Extension conditions with core melt

## APPENDIX-II: Events to be Practically Eliminated

For practical purpose, the cases to be addressed for 'practical elimination' could be grouped within the following five categories:

1) Events/Phenomena that could lead to prompt reactor core damage and consequent early containment failure due to :
    a) Failure of a large component in the reactor coolant system (RCS);
    b) Uncontrolled Reactivity Accidents;
    c) Prompt Criticality.
2) Severe accident phenomena which could lead to early containment failure due to:
    a) Direct Containment Heating;
    b) Molten Fuel Coolant Interaction (MFCI);
    c) Large Sodium Fire.
3) Severe accident phenomena which could lead to late containment failure due to:
    a) Molten Core Concrete Interaction (MCCI);
    b) Loss of containment heat removal.
4) Accident Conditions with Containment Bypass/ Failure of Isolation;
5) Significant Fuel Degradation in a Storage Pool.

ANNEXURE-I Phenomena Matrix [13], [14]

| No | Phenomena | Description | Safety significance | Indicative applicable events |
|---|---|---|---|---|
| 1. | Pressure loss in core region | Pressure loss along the coolant flow path of a subassembly in the core is caused by acceleration, friction, and form losses. They include losses from flow contraction and expansion depending on the inlet and outlet geometry, and losses from the wire wrap spacer. | Affects the Power to flow (P/F) ratio. The impact on P/F can be large in the case of enrichment error (over-enriched subassembly is loaded at lower flow region in the core) and subassembly flow blocked condition. | All PIEs except PIEs in functional group 6, 7, 8 & 9 |
| 2. | Natural Convection | Coolant natural convection in the core is driven by buoyancy force developed due to temperature dependence of density of coolant in the primary sodium circuit. Strength of natural convection in the core is dependent on the thermal centre difference between the hot and cold locations in the primary sodium circuit. | This phenomena is not applicable during normal power operation and decay heat removal condition since the coolant flow is forced through the core under these conditions. This phenomenon has an effect on decay heat removal condition with all the primary coolant pumps tripped, which is possible under some of the design extension conditions. | Long term SBO extending beyond the capacity of battery power source. |
| 3. | Heat transfer between fuel, cladding and coolant | • Heat transfer between the cladding and coolant is dependent on coolant flow velocity near the cladding surface, the shape and diameter of spacer, and P (fuel pin pitch)/D (fuel pin diameter).<br>• If the fuel fails, fission gas is released, which results in deterioration of the heat transfer rate between cladding | It affects the fuel and clad temperature predictions. Reduction in fuel heat transfer may lead to the fuel heat up. | All PIEs except PIEs in functional group 6, 7 & 8 |

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| | | and coolant | | |
| 4. | Intra-subassembly flow distribution | • Flow distribution among subassemblies in the core is achieved through orifices fitted at the foot of fuel assemblies.<br>• Flow distribution inside a fuel assembly is governed by bundle geometry. During the natural convection decay heat removal phase, it is also governed by the temperature distribution inside the bundle.<br>• The temperature distribution within the fuel subassembly is governed by the flow and power distribution within the subassembly and by inter subassembly heat transfer.<br>• Flow distribution inside the fuel subassembly may deviate from the initial prediction due to various uncertainties, viz., tolerances, manufacturing errors, and fuel geometry changes due to burnup. | Results in fuel, cladding, and coolant temperature variation with respect to the nominal predicted values. | All PIEs except PIEs in functional group 6, 7, 8 & 9 |
| 5. | Coolant boiling | Coolant boiling could occur due to flow reduction caused due to pump failure or local blockage that results in decrease of coolant flow through a particular subassembly. These result in temperature rise of subassemblies. | When coolant boiling occurs, fuel pin integrity is not likely to be maintained, and fission gas is released from the heated pins. Also, reactivity changes are caused resulting in core power changes, which may | DEC-B events. |

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| | | | result in further disintegration of the fuel subassembly. | |
| 6. | Flow induced vibration in a subassembly | Fuel pins may vibrate due to fluid-structure interaction when the coolant flow velocity around the fuel pin in the fuel subassembly is high. | There is a possibility of fuel pin failure due to fretting. | Events involving primary pump acceleration. |
| 7. | Coolant flow between wrapper tubes | Inter-subassembly heat transfer is enhanced by heat conduction and convection of sodium between the wrapper tubes during natural circulation decay heat removal. | Coolant flow between wrapper tubes influences inter and intra-subassembly heat transfer phenomena during natural circulation decay heat removal. Core temperature during this condition gets benefitted due to this phenomenon. | Long term SBO extending beyond the capacity of battery power source. |
| 8. | In pin fuel melting | During over power events with core flow unaffected, partial fuel melting may occur prior to clad failure. | Molten fuel can move inside the cladding before the cladding fails. Negative reactivity is inserted when melted fuel is relocated in the gas plenum region. (Since central region of the fuel pins alone melt and coolant flow is not affected, the clad remains intact.) | Unprotected Transient Over Power Accident (UTOPA). |
| 9. | Eutectic formation | Molten fuel reacts with inside wall of the cladding, resulting in the formulation of the liquid phase rapidly. The cladding fails due to erosion and creep.<br><br>[Eutectic reaction of uranium, plutonium, and FPs in the fuel alloy and cladding materials such as steel can occur in | Affects the cladding integrity. | DEC-B events. |

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| | | metallic fuel.<br><br>• Eutectic reaction occurs at temperature higher than 650 $^{o}C$ and becomes more severe as temperature rises.<br>• Rapid eutectic reaction occurs around 1100 $^{o}C$ (for metallic fuel), which is higher than fuel melting point.<br>• Under an enrichment error condition during which fuel temperature reaches 650 $^{o}C$, fuel failure may occur at the end of life.<br>• Under a local blockage event, fuel failure may occur because temperature increases locally, which may initiate an eutectic reaction.<br><br>Same phenomena occurs in oxide fuels also but only at very high temperatures.] | | |
| 10. | Relocation of the molten material | After failure of the cladding, the liquidus fuel flows into the coolant path. | The molten material is dispersed and frozen in the short term after it flows into the coolant path. Negative reactivity is inserted when the solidified fuel flows out of the core. In the case of solidified fuel trapped in the gap of the wire spacer between the fuel pins, however, local blockage occurs and reactivity effects due to coolant boiling may be caused leading to fuel pin failure | DEC-B. |

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| | | | propagation. | |
| 11. | Bowing of fuel pin | Differential temperature and swelling of the fuel pin results in its bowing, which narrows the coolant passage, and in the worst case, point/line contact between adjacent pins occurs. | Effects the flow within the subassembly. May lead to fuel pin failure | All PIEs except PIEs in functional group 6, 7, 8 & 9. |
| 12. | Nuclear thermal hydraulic feedback including spatial effects | Thermo-hydraulic – nuclear feedback due to Doppler, fuel and clad axial expansion, volumetric sodium expansion, control rod drive line expansion and grid plate expansion causes reactivity effects in the core. | This phenomenon is very essential for the fission power and associated heat production in the core. | All PIEs except PIEs in functional group 6, 7, 8 & 9. |
| 13. | Sodium stratification | The temperature difference of coolant in the heat transport circuits of sodium cooled fast reactors (SFRs) is normally large. Typically, the temperature difference between the core inlet and the core outlet is around 150 K. The difference of temperature of coolant between the steam generator outlet and the core outlet is as high as 200 K. During normal steady-state operating conditions at full flow, buoyancy forces do not significantly influence the coolant flow pattern in the heat transport systems except in some specific regions where the velocity is low. Under certain off-normal conditions, the coolant flow rate is reduced and large temperature variation still remains. In such transient situations, buoyancy influence can be very important and it can modify the global thermal hydraulic behaviour. Within the reactor vessel, thermal stratification may be caused in the hot pool | From the viewpoint of the reactor system analysis, the influence of buoyancy forces and especially thermal stratification in reactor vessel may change the global system behaviour. So, a particular attention must be paid to take into account the possible influence of thermal stratification on the system behaviour. The transition from normal operation to decay heat removal condition at reduced flow rate in the primary sodium circuit would have serious effects due to thermal stratification. This is important to accurately estimate the thermal conditions at which primary sodium enters the decay heat exchangers. The stratification effects are more pronounced | All PIEs except PIEs in functional group 6, 7, 8 & 9. |

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| 15. | | and the cold pool resulting in temperature oscillations in the interfaces. | in case of natural convection conditions in the primary sodium circuit.<br><br>(From the viewpoint of thermal stress analysis on the piping, the occurrence of thermal stratification must be predicted to prevent any crack or damage leading to a sodium leak. Effectively, due to thermal stratification, the temperature difference between the upper and lower regions of a horizontal pipe produces bending moments and local stresses. These stresses must be added to existing mechanical and thermal stresses. So, one needs to predict not only the occurrence of thermal stratification but also the amplitude of the temperature difference between the upper and lower regions in horizontal pipes) | |
| 14. | Fuel-coolant interaction | Molten fuel comes in contact with sodium coolant. | Fuel-coolant interaction may affect the rate of voiding and may lead to additional mechanical energy release and pressurization of main vessel (Rapid sodium expulsion and reactivity changes may also occur). | DEC-B. |
| 15. | Recriticality | SFRs are not designed at | Huge amount of | ULOFA. |

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| | | optimum reactive configuration. After melting, core can get compacted and add reactivity greater than delayed neutron fraction. This state is known as super prompt critical state. | energy is released in this state in a very short time. The resulting effect is the disruption of the core. | |
| 16. | Core disassembly | Once super prompt critical state is reached in the core, the core starts disassembling due to rapid heating and vaporization of the fuel. | Rapid heating and vaporization of the fuel produce high pressures that may lead to damage of the reactor vessel and other internal structures. The expansion of the core produces negative reactivity effects and terminates the nuclear chain reaction. | ULOFA & Total and instantaneous blockage of a fuel SA. |
| 17. | Coolant voiding | Coolant voiding in the core/sub-assembly may occur due to loss of flow or over-power transients. | Sodium voiding in the central region of the core introduces positive reactivity feedback that leads to increase of reactor power resulting in fuel heat-up and melting. | DEC-B. |
| 18. | Slug expulsion | Once coolant starts boiling in the core, slug flow conditions prevail in the sub-assembly. | Flow evolution in the boiling channels is governed by the slug expulsion phenomenon that occurs. Transient thermal effects in the core depend on the modelling of this phenomenon. (Re-criticality may lead to the high power density) | DEC-B. |
| 19. | Sodium-water reaction | Sodium reacts chemically with water in the case of an unexpected tube failure of a steam generator (SG). | The boundary wall between the primary and secondary circuit in the intermediate heat exchanger should withstand the pressure propagated from the | Events in functional group 7. |

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| | | | steam generator due to a credible sodium water reaction, and also the integrity of the steam generator shell and the secondary circuit components should be maintained. | |
| 20. | Thermal stripping | ‘Thermal stripping’ is a phenomenon, which leads to random temperature fluctuations at the interface between non-isothermal streams, arising out of jet instability. Due to the large heat transfer coefficient associated with liquid sodium, these temperature fluctuations are transmitted to the adjoining structures with minimal attenuation, which eventually lead to high cycle fatigue and crack initiation in structures. Detailed thermal hydraulic investigation calls for identification of zones prone to thermal stripping and prediction of temperature fluctuations by special modelling techniques. | Leads to high cycle fatigue and crack initiation in the structures | All PIEs except PIEs in functional group 6, 7, 8 & 9. |
| 21. | Gas entrainment | Due to the presence of sodium to argon interface in several systems in the plant, gas entrainment into sodium flow is possible. Free level of hot pool, over- flow system of main vessel cooling system and vortex formed due to shaft rotation in pump tank are some of the potential locations of gas entrainment. | Gas entry into reactor core leads to reactivity effects leading to power variation. | All PIEs except PIEs in functional group 6, 7, 8 & 9. |
| 22. | Cellular Convection | Natural convection flow pattern developed in the narrow component penetrations of reactor assembly are generally cellular in nature. | Cellular convection leads to circumferential temperature gradient in components resulting in their tilting. | All PIEs except PIEs in functional group 6, 7, 8 & 9. |

## REFERENCES


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## LIST OF PARTICIPANTS

## WORKING GROUP

Dates of meeting:

October 10, 2019
November 07, 2019
November 29, 2019
January 9, 2020
January 30, 2020
February 5, 2020
February 21, 2020
March 9, 2020

Members of the Working Group:

| | | |
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| Dr. R. S. Rao, NSAD, AERB | : | Convenor |
| Shri Utkarsh S. C, NPSD, AERB | : | Member |
| Dr. R. B. Solanki, RDD, AERB | : | Member |
| Dr. Mansoor Ali, NSAD, AERB | : | Member |
| Dr. Santosh K. Pradhan, NSAD, AERB | : | Member |
| Dr. Subrata Bera, SRI, AERB | : | Member |
| Shri Anjit Kumar, NPSD, AERB | : | Member |
| Shri Sumit Awasthi, OPSD, AERB | : | Member |
| Shri Abhinav Kumar, NPSD, AERB | : | Member |
| Shri Prashant Sharma, NSAD, AERB | : | Member-Secretary |
| Dr. S.K. Dubey, DRPE, AERB | : | Invitee |
| Shri K. Natesan, IGCAR | : | Invitee |

**TASK-FORCE (SFR)**

**STANDING COMMITTEE FOR REVIEW OF DETERMINISTIC SAFETY ANALYSIS OF NUCLEAR FACILITIES (SC-DSA)**

**Dates of meeting**: October 07, 2022, November 03, 2022, November 15, 2022, December 13, 2022, December 21, 2022, February 8, 2023, October 26, 2023, August 21, 2024

Members and Invitees of SC-DSA:

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| Dr. R. K. Singh (Chairman) | : BARC (Former) |
| Shri Avinash J Gaikwad (Vice-Chairman) | : AERB (Former) |
| Shri H. G. Lele | : BARC (Former) |
| Shri S. Hajela | : NPCIL |
| Shri Joe Mohan | : BARC |
| Dr. Deb Mukhopadhyay | : BARC |
| Dr. K. Natesan | : IGCAR |
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| Dr. R. S. Rao | : AERB |
| Dr. Seik Mansoor Ali | : AERB |
| Dr. S. K. Dubey | : AERB |
| Shri D. Bhattacharya | : AERB |
| Shri Jagannath Mishra | : AERB |
| Shri Nalini Mohan | : NPCIL (Former) |
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| Shri P. N. Prasad | : NPCIL (Former) |
| Shri Manoj Kansal | : NPCIL |
| Dr. Santosh Kumar Pradhan | : AERB |
| Shri Anuj Kumar Deo (Member Secretary) | : AERB |
| Shri Prashant Sharma (Responsible Officer) | : AERB |
| Shri Abhay Kumar, (co-opted Member) | : BARC |
| Shri D. Naga Sivayya, Invitee | : IGCAR |

## ADVISORY COMMITTEE FOR NUCLEAR AND RADIATION SAFETY (ACNRS)

**Dates of meeting:** November 9, 2019, February 14, 2024; July 11, 2024; September 09, 2024, October 24, 2024, January 14, 2025

**Members and Invitees of ACNRS:**

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| Shri S.S. Bajaj (Chairman) | : | AERB (Former) |
| Shri S. B. Chafle | : | AERB (Former) |
| Shri A.K. Balasubrahmanian | : | NPCIL (Former) |
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| Dr. (Smt.) Sadhana Mohan | : | BARC (Former) |
| Dr. L.R. Bishnoi | : | AERB (Former) |
| Shri S.T. Swamy, Member Secretary (Till Jan 2020) | : | AERB (Former) |
| Shri Shri H. Ansari, Member-Secretary (Till Feb 2024) | : | AERB (Former) |
| Dr. R. B. Solanki, Member-Secretary | : | AERB |

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