PBMR

Pebble Bed Modular Reactor (Pty) Ltd.
Reg. No: 1999/17946/07

3rd Floor, Lake Buena Vista Building 1267 Gordon Hood Avenue Centurion Republic of South Africa
PO Box 9396 Centurion 0046 Tel (+27 12) 677-9400 Fax (+27 12) 663-3052/8797/ 677-9446

Date:	Your Ref.:	Our Ref.:	Enquiries:
June 13, 2006		USDC20060613-1	E.G. Wallace TEL:US 423-344-6774

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

NRC Project No. 732

Attention: Mr. N. Prasad Kadambi

Subject: PBMR White Paper: PRA Approach

Ref: PBMR (Pty) Ltd. Letter, Subject: Submittal of PBMR Preapplication White Papers, May 1, 2006

Enclosed is the PBMR white paper entitled "Probabilistic Risk Assessment Approach for the Pebble Bed Modular Reactor", Revision 1. As described in the reference, this paper describes our approach to the development of a full scope PRA for the PBMR design and is necessary to understand the discussions provided in other papers on risk-informed Licensing Basis Event Selection, SSC Classification and Defense in Depth that will be submitted separately as part of our preapplication work.

If you have any questions about this submittal, please feel free to contact me.

Yours sincerely,

Edward G. Wallace
Senior General Manager- US Programs
PBMR (Pty) Ltd

Enclosure
cc: Mr. Farouk Eltawila, RES
Ms. Christiana Lui, RES
Mr. James Danna, RES
Ms. Margaret T Bennett, RES
Mr. Stuart J. Rubin, RES
Mr. Ujagar Bhachu, NRR

US DESIGN CERTIFICATION

PROBABILISTIC RISK ASSESSMENT APPROACH FOR THE PEBBLE BED MODULAR REACTOR

Document Number	**:**	**039144**
Revision	**:**	**1**
Status	**:**	**Approved**

Signatures for approved documents are held on file in the Document Control Centre of PBMR (Pty) Ltd

ABSTRACT

This paper identifies the regulatory issues related to the Probabilistic Risk Assessment (PRA) approach for which US Nuclear Regulatory Commission feedback is desired during the preapplication review of the Pebble Bed Modular Reactor (PBMR). The regulatory foundation for review of the approach to PRA is summarized, compliance with the regulatory criteria is described, and specific issues for which feedback is requested are described.

CONFIGURATION CONTROL

Document History

Rev.	Date	Preparer	ECPs	Changes
A	2006/06/05	K Fleming	n/a	New document.
B	2006/05/22	K Fleming	n/a	Updated after Comments Review.
1	2006/056/13	K Fleming	n/a	Edited and sent for approval.

Document Approval

Action	Function	Designate	Signature
Prepared	Author	K Fleming	See signatures on file
Reviewed	Independent Reviewer	E Burns	See signatures on file
Approved	Project Manager	E Wallace	See signatures on file

Document Retention Time

This document is a Quality Record and shall be retained in accordance with PRC0012.

CONTENTS

FIGURES

TABLES

ABBREVIATIONS

This list contains the abbreviations used in this document.

Abbreviation or Acronym	Definition
ACRS	Advisory Committee on Reactor Safeguards
ACS	Active Cooling System
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BDBE	Beyond Design Basis Event
C&I	Control and Instrumentation
CBCS	Core Barrel Conditioning System
CCS	Core Conditioning System
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
COL	Combined Operating Licence
DBA	Design Basis Accident
DBE	Design Basis Event
DCA	Design Certification Application
DOE	Department of Energy
DPP	Demonstration Power Plant
DWS	Demineralized Water System
EMDAP	Evaluation Model Development and Assessment Process
EP	Emergency Planning
EPBE	Emergency Planning Basis Event
EPCC	Equipment Protection Cooling Circuit
EPRI	Electric Power Research Institute
EPS	Equipment Protection System
ESD	Event Sequence Diagram
ESKOM	ESKOM Holdings Limited – RSA (the utility company)
FHSS	Fuel Handling and Storage System
FIVE	Fire Induced Vulnerability Evaluation
FPS	Fire Protection System
GCR	Gas-cooled Reactor
HAZOP	Hazard and Operability (study)
HPB	Helium Pressure Boundary
HPS	Helium Purification System
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation and Air-conditioning
HX	Heat Exchanger

Abbreviation or Acronym	Definition
LBE	Licensing Basis Event
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MHSS	Main Heat Sink System
MHTGR	Modular High Temperature Gas-cooled Reactor
MPS	Main Power System
NEI	Nuclear Energy Institute
NNR	National Nuclear Regulator (RSA)
NRC	Nuclear Regulatory Commission (USA)
OCS	Operational Control System
PBMR	Pebble Bed Modular Reactor
PCU	Power Conversion Unit
PDS	Plant Damage State
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PRS	Pressure Relief System
PSID	Preliminary Safety Information Document
QHO	Quantitative Health Objective
RAI	Request for Additional Information
RCCS	Reactor Cavity Cooling System
RCS	Reactivity Control System
RPS	Reactor Protection System
RSS	Reserve Shutdown System
SAS	Small Absorber Spheres
SC	Safety Case
SFT	Spent Fuel Tank
SRM	Staff Requirements Memorandum
SRP	Standard Review Plan
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
TG	Turbine Generator
UFT	Used Fuel Tank

1. INTRODUCTION

1.1 SCOPE AND PURPOSE

The PBMR Design Certification Application (DCA) will include a design-specific Probabilistic Risk Assessment (PRA) as required by 10 CFR Part 52. This paper outlines the relevant regulatory policy and guidance for this PRA, describes the approach being followed for the development of the PRA, and sets forth certain issues for review and discussion in order to facilitate preparation of the PBMR design certification application under 10 CFR Part 52. Key elements discussed include the scope and objectives for the PRA, regulatory guidance used in the formulation of these objectives, and how the objectives have been factored into the PRA framework. The focus of the paper is to identify potential issues related to the performance of the PRA and its use in the selection of Licensing Basis Events (LBEs) for the PBMR DCA.

The PRA for the PBMR serves two primary purposes, both in keeping with the regulatory requirements for the design certification submittal. The first involves developing the PRA to demonstrate adequacy of the design in meeting the objectives of a deterministic safety design philosophy, safety requirements embodied in the available regulations and guidance, and risk management objectives for protecting the public and the plant investment. The second involves using information from the PRA in a manner that complements traditional deterministic approaches such as defense-in-depth, safety classification, special treatment, and the formulation of criteria and guidance that will be needed to certify the PBMR design.

A risk-informed approach that uses both probabilistic and deterministic elements is appropriate, because the current set of deterministic licensing requirements was developed for reactors with a fundamentally different safety design philosophy. Risk insights from the PBMR PRA are viewed as essential to developing a design that is optimized in meeting safety objectives and is expected to be useful in interpreting the applicability of the existing requirements to the safety design approach of the PBMR. Results and insights from the PRA will be needed to assure the safety of the design against the Nuclear Regulatory Commission’s (NRC’s) Safety Goals, and to determine whether any additional requirements may be necessary that are specific to the PBMR.

The PRA that will be submitted to support the PBMR DCA will correspond to a design for a multi-module PBMR that is to be certified for a range of possible US Sites. This design stage and site-independent PRA will be developed with the benefit of having already performed a site-specific PRA for the Demonstration Power Plant (DPP) that is being built near Cape Town in the Republic of South Africa. The DPP site-specific PRA is expected to add credibility to the assumptions that will need to be made in order to perform a PRA for a design certification applicable to a range of sites.

1.2 STATEMENT OF THE ISSUES

The issues addressed in this paper are framed in terms of the following questions about the PRA that will be performed to support the PBMR DCA:

1. Are the intended uses of the PBMR PRA, including the selection of LBEs, input to the safety classification of Structures, Systems and Components (SSC) and the development of special treatment requirements for SSC, acceptable?
2. Given the intended uses described in Issue 1, what is the appropriate scope of the PRA?

3. How will the PBMR safety design approach, such as the inherent and passive PBMR safety characteristics, be taken into account?
4. How will the PRA be used to understand the adequacy of the defense-in-depth approach in the design, construction and operation of a PBMR plant?
5. Are risk metrics such as Core Damage Frequency (CDF) appropriate for the PBMR? What risk metrics will be applied in the PRA?
6. How will the PRA use and interface with deterministic safety analyses?
7. How will the limited PBMR service data be addressed in the development of PRA data?
8. How will source terms for event sequence be developed? How are uncertainties addressed?
9. How will modular reactor aspects of the design be accounted for in the PRA?
10. What is the approach to ensure the adequacy of the PRA quality for the intended applications?
11. What is the approach in the PRA for treating uncertainties associated with as-procured, as-built, site-specific and as-operated information that is not available when the DCA is submitted? How will these uncertainties be accounted for in the selection of a stable set of LBEs for operating plants?
12. What preapplication activities can be defined to set the stage for a successful PRA review following the DCA submittal?

The regulatory foundation for deriving this list of issues is developed in Chapter 2 of this paper. The PBMR approach to solving these issues is outlined in Chapter 3 and will be discussed at future NRC workshops.

1.3 SUMMARY OF PREAPPLICATION OUTCOME OBJECTIVES

The objective of this paper and the follow-up workshops and paper revision that are planned is to get NRC agreement on the list of issues for the use of PRA to support PBMR certification as well as agreement on the approach to solving these issues as outlined in this paper. Specifically, we would like the NRC to agree with the following statements, or provide an alternative set of statements with which they agree.

1. The scope of the PBMR PRA outlined in this paper is appropriate for the intended uses of the PRA in the design certification of the PBMR. These uses include input to the identification of any risk vulnerabilities and risk-significant systems; the selection of LBEs; safety classification of SSC; and derivation of regulatory design criteria and special treatment requirements for SSC.
2. The PRA framework outlined in this paper is a reasonable approach to capture the unique and specific elements of the PBMR safety design approach and to delineate the elements of the PBMR PRA that are common to PRAs for Light Water Reactors (LWRs).
3. The PBMR approaches to initiating event selection, event sequence development, end state definition, and risk metrics are appropriate.
4. The approach to the treatment of inherent characteristics and passive SSC outlined in this paper is reasonable and consistent with the current state-of-the-art of PRA.

5. The approach to the use of deterministic engineering analyses to provide the technical basis for predicting the plant response to initiating events and event sequences, success criteria, and mechanistic source terms yields an appropriate blend of deterministic and probabilistic approaches to support the PBMR design certification.
6. The approach to the development of a PRA database outlined in this paper, including the use of applicable data from LWRs, use of expert opinion and treatment of uncertainty, is a reasonable approach for the PBMR PRA.
7. The process for the development of a mechanistic source term outlined in this paper is a reasonable approach for this PRA application.
8. The approach for the PRA treatment of single and multiple reactor accidents in a multi-module design is sufficient to support certification of the basic single module of the PBMR for multi-module configurations.
9. The approach to using current guides and standards for LWR PRA quality and independent peer review taking into account the differences due to the PBMR's safety design approach that is outlined in this paper is an acceptable approach to determining the adequacy of the PBMR PRA for its intended uses outlined above.
10. The DCA PRA will account for uncertainties associated with as-procured, as-built, site-specific, and as-operated information in a conservative and bounding manner to provide assurance that the LBEs derived from the DCA PRA will be appropriate for as-built and as-operated plants. A design review at the Combined Operating Licence (COL) stage will be performed to confirm the validity of the LBEs.
11. Preapplication activities to set the stage for a successful PRA review suggested in this paper and agreed upon in subsequent workshops are sufficiently well defined.

1.4 RELATIONSHIP TO OTHER PREAPPLICATION FOCUS TOPICS/PAPERS

This paper addresses the PRA approach for the PBMR as required by 10 CFR Part 52. As an integrated part of the submittal for design certification, the PRA supports and is supported by several other key activities covered by related papers. These activities include elements of a technically sound deterministic engineering evaluation. Elements of this evaluation include deterministic safety design principles, codes and standards; the development of reactor-specific success criteria; and the use of verified computer models in the prediction of the plant response to initiating events and event sequences, and in the development of mechanistic source terms

The PRA is developed on a foundation of technically sound deterministic engineering assessments to establish the plant response to initiating events, success criteria and mechanistic source terms. The safety design approach itself is based on deterministic engineering principles. Key design parameters such as core size and shape, power density, and reactor citadel configuration are, in fact, derived from deterministic engineering evaluations. Key elements of the deterministic engineering bases for the PRA and other elements of the approach to design certification are covered in other papers

Licensing Basis Event (LBE) selection, a topic of a companion paper, includes the description of the methodology applied in the determination of Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs) and Beyond Design Basis Events (BDBEs), which for the PBMR are derived from a risk-informed and integrated decision-making process. Once the LBEs are selected with input from the PRA, the selection and classification of safety related SSC is performed deterministically. Information from the PRA is used to support the derivation of special treatment requirements for SSC, which is the subject of the SSC Classification Paper. The LBEs and SSC safety classification are then subjected to rigorous, conservative, deterministic safety analyses to demonstrate that deterministic safety requirements are met with sufficient safety margins.

The principles of risk-informed regulation in Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis* [1] are carefully adhered to. Regulatory Guide 1.174 requires consideration of defense-in-depth, which will be discussed in detail in a paper that will specifically look at the interrelated elements of defense-in-depth for the PBMR design. Taken together, these topics cover the essence of the systematic, disciplined method for defining nuclear safety requirements for the PBMR, and defining and delineating the functional nuclear safety requirements for the plant. The approach as outlined in the papers is responsive to both the regulatory requirements of the NRC and the design and technical requirements of industry codes and standards. It is based on a balanced and integrated blend of deterministic design principles and safety assessments as well as a state-of-the-art probabilistic evaluation. The use of the PRA complements the role of deterministic approaches, and helps to focus the deterministic elements on the areas most important to risk. The risk significant event sequences and the LBEs will be considered as part of an integrated assessment of the safety characteristics of the PBMR.

The PRA itself incorporates key inputs and analyses from deterministic codes and analytical methods covered by other papers. These include the analytical code verification and validation as performed in the Evaluation Model Development and Assessment Process (EMDAP), Phenomena Identification and Ranking Table (PIRT) comparisons and application to the PBMR design, fuel response modelling, systems response modelling, mechanistic source terms and release modelling, and the PBMR test programme. The PRA will be developed in a manner that supports the integration of probabilistic and deterministic methods in a consolidated framework to assure safety. This includes a use of the PRA to evaluate the adequacy of defense-in-depth in the design, construction, and operation of the PBMR as discussed more fully in the white paper on defense-in-depth. The PRA development and its use to support design certification are consistent with a risk-informed approach.

2. REGULATORY FOUNDATION

2.1 NRC REGULATIONS

The PBMR application will contain and rely upon a state-of-the-art PRA. The use of PRA to address nuclear reactor safety issues has increased over the 30 years since WASH 1400 was published. In keeping with the findings of the Kemeny [2] and Rogovin [3] Commissions findings on the accident at Three Mile Island, and in agreement with numerous staff recommendations and Commission Policy statements since, the reliance on and use of PRA in the regulatory process has greatly increased. The PBMR application for design certification will follow this regulatory history, make use of the available policy statements and supporting guidance, and be consistent with the increased use of risk-informed and performance-based regulation that has been advanced by the nuclear industry. The PBMR DCA will utilize a design-specific PRA as part of a balanced and integrated set of deterministic and probabilistic approaches to guide the design and to develop the licensing basis.

At the heart of the requirements for the PBMR is 10 CFR Part 52, Subpart B, ‘Standard Design Certifications’ [4], which sets out the requirements and procedures applicable to the issuance of design certifications for nuclear power facilities. It states:

> *§ 52.47 Contents of applications.*
>
> *(a) The requirements of this paragraph apply to all applications for design certification.*
>
> *(1) An application for design certification must contain:*
>
> *(v) A design-specific probabilistic risk assessment;*

The DCA for the PBMR, which will be made under 10 CFR Part 52, will comply with this requirement.

The PRA will address a full range of events including three categories of LBEs: AOOs, DBEs and BDBEs. The approach to utilize the PRA results to define these LBEs is described in a companion paper on LBE selection [5]. This will serve as an integral part of the application in conformance with the intended standard for review of applications as stated in 10 CFR § 52.48, which states:

> *§ 52.48 Standards for review of applications.*
>
> *Applications filed under this subpart will be reviewed for compliance with the standards set out in 10 CFR part 20, part 50 and its appendices, and parts 73 and 100 as they apply to applications for construction permits and operating licenses for nuclear power plants, and as those standards are technically relevant to the design proposed for the facility.*

It is noted that 10 CFR Part 52 does not contain specific guidance on the form, scope, content or criteria for establishing the adequacy of the PRA in meeting this requirement. Other Commission statements and staff requirements provide this information. They are summarized in the following paragraphs.

In summary, the PBMR application will comply with 10 CFR Part 52 and contain a PRA which will be demonstrated to have the requisite quality, scope and content to be relied upon for the application and its approval. The PRA will be shown to be fully integrated into the design and intended operation of the facility so that all applicable requirements are met.

2.2 NRC POLICY STATEMENTS

There are many NRC policy statements which state that a PRA should be prepared and submitted as part of the application. Since this requirement is adopted explicitly in 10 CFR Part 52, these references are not repeated here. Specific PRA policy statements are discussed in paragraph 2.2.1.

2.2.1 PRA Policy Statement

For many years the NRC has had a policy that PRA should be relied upon to help assure safety. The programmes to implement these statements and the supporting regulatory guidance have provided an extensive definition of the requirements for PRAs and their use. These PRA requirements are discussed in paragraph 2.3.

On August 16, 1995, the Commission adopted the following Policy Statement regarding the expanded use of PRA [6]:

- '*The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.*
- *PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices.'*

The approach to performing the PBMR PRA in support of the DCA and the expected uses of the information provided by the PRA to support the licensing basis are consistent with the expectations raised in this policy statement. The PBMR DCA will address both the stated NRC intent to rely more on PRA methods, and the need to acknowledge and meet existing regulations.

The traditional licensing requirements referred to in this policy statement are embodied in a primary manner in 10 CFR Part 50, Appendix A [7]. Risk insights from the PRA will be used to guide the application of the traditional deterministic licensing requirements to the PBMR and its safety design philosophy. The risk informed design certification approach includes a systematic review of the regulations to assure that all are met to the extent they are applicable and that the associated licensing principles are applied in a manner appropriate for the PBMR. Risk insights from the PRA are also expected to be useful to identify any safety issues specific to the PBMR for appropriate regulatory treatment. Through the process of integrating the risk significant event sequences with the LBEs, an enhanced level of coherence between the deterministic and probabilistic perspectives is expected.

Thus the approach complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

2.2.2 Policy Issues Related to Certification of Non-LWRs

The PBMR is subject to policy issues involving non-LWR applications. It is the intent of the PBMR application to comply with the NRC's guidance on these issues.

SECY 2003-0047, 'Policy Issues Related to Licensing Non-Light Water Reactor Designs' [8] offers staff recommendations on seven relevant policy issues that had been originally defined in SECY 2002-0139. Of these seven issues there are two, Issue 4: 'Use of PRA to Support Licensing Basis' and Issue 5: 'Use of Mechanistic Source Terms', which specifically relate to the PBMR PRA and are discussed herein. On these two issues the Staff Requirements Memorandum for SECY 2003-0047 [9] stated the Commissioner's approval of the staff recommendations on both of these issues.

Also included, but left unresolved from the seven issues of SECY 2003-0047, were policy issues associated with the treatment of integrated risk on multi-reactor sites and for modular reactor designs, which is part of Issue (a). These are addressed below.

With respect to Issue 4, the staff recommended that the Commission take the following actions:

> *'Modify the Commission's guidance, as described in the SRM of July 30, 1993, to put greater emphasis on the use of risk information by allowing the use of a probabilistic approach in the identification of events to be considered in the design, provided there is sufficient understanding of plant and fuel performance and deterministic engineering judgment is used to bound uncertainties.*
>
> - *Allow a probabilistic approach for the safety classification of structures, systems, and components.'*
> - *Replace the single failure criterion with a probabilistic (reliability) criterion.*
>
> *This recommendation is consistent with a risk-informed approach. It should be noted that this recommendation expands the use of probabilistic risk assessment (PRA) into forming part of the basis for licensing and thus puts greater emphasis on PRA quality, completeness, and documentation.'*

The PBMR application will include a design-specific PRA and will demonstrate compliance with this staff recommendation. Risk information is being used and will be presented to support the 'probabilistic approach in the identification of events to be considered in the design'. The need for 'sufficient understanding of plant and fuel performance' is acknowledged and will be addressed by other papers on the verification and validation of evaluation models and code suites; fuel design and qualification and in the development of the mechanistic source terms that will be used in the PRA and in the deterministic safety analysis of design basis events. The integration of the PRA with and reliance upon deterministic analyses and engineering judgment will be demonstrated. This includes a use of the PRA to evaluate the application of prevention and mitigation strategies as discussed in the paper on defense-in-depth. The classification of SSC will follow a deterministic approach based on the LBEs derived from the PRA results as described in another paper on the safety classification of SSC.

With respect to Issue 5 of SECY 03-0047, the staff recommended that the Commission take the following action:

- *'Retain the Commission's guidance contained in the July 30, 1993, SRM that allows the use of scenario-specific source terms, provided there is sufficient understanding*

> *and assurance of plant and fuel performance and deterministic engineering judgment is used to bound uncertainties.*
>
> *This recommendation will allow credit to be given for the unique aspects of plant design (i.e., performance-based) and builds upon the recommendation under Issue 4. Furthermore, this approach is consistent with prior Commission and ACRS views. However, this approach is also dependent upon understanding fuel and fission product behaviour under a wide range of scenarios and on ensuring fuel and plant performance is maintained over the life of the plant.'*

SECY 2003-0047 states that in NUREG-1338, *Draft Preapplication Safety Evaluation Report for the Modular High Temperature Gas-Cooled Reactor* [10], the staff stated that final acceptance of the mechanistic source term was 'contingent on the satisfactory resolution of technical and policy considerations and noted that extensive research and testing was needed to address the technical issues.'

The ACRS, in a letter dated February 19, 1993 [11], stated that 'the staff proposal to base the source term on mechanistic analyses appears reasonable, although it is clear that the present data base will need to be expanded.' and 'It will be appropriate for the staff to consider using newer approaches when it develops source terms, and to take specific account of the unique features of …the reactor type'.

The PRA that will be performed to support the PBMR DCA, which is discussed more fully in Chapter 3 of this paper, will be of sufficient scope and detail to calculate the frequencies and radiological consequences of PBMR design-specific event sequences, and will address the uncertainties in both frequencies and consequences. The information to be provided by the PRA is similar to that provided in an LWR Level 3 PRA. Hence, the PRA will utilize mechanistic source terms as well as site boundary radiological dose calculations to assess the consequences of the modelled event sequences. Mechanistic source terms will also be required to perform the deterministic safety analysis for Design Basis Accidents (DBAs) that will be included in the DCA. Among the issues to be resolved in support of both the probabilistic and deterministic safety analysis is the issue of establishing the adequacy of the mechanistic source terms. This includes demonstrating sufficient understanding of fuel and plant performance and all significant radionuclide transport phenomena for a sufficiently wide range of scenarios. Resolution of these issues is expected to place requirements on both the probabilistic and deterministic safety analyses that will be included in the DCA.

The PRA that is performed to derive the LBEs for the PBMR DCA will be updated as necessary to reflect changes in the plant design, construction and operational stages to the extent needed to ensure that conclusions derived from the PRA in support of the licensing basis and design certification remain valid. The methods for selecting LBEs and for making safety classification of SSC as described in other papers include deterministic elements to address uncertainties in the PRA results so that LBE selection is not expected to be sensitive to expected numerical changes in the PRA results during subsequent PRA updates.

The PRA will address the integrated risk of the multi-module design of the PBMR to be supported by the DCA. This approach is expected to accommodate a full range of outcomes of the ongoing policy discussions among the staff, Commissioners and ACRS regarding the issue of integrated risk.

The PBMR PRA performed to support the DCA will include a full-scope PRA treatment of internal and external hazards. The treatment of relevant issues in the PBMR PRA is consistent with the staff expectations for future designs.

In 1993, the NRC Staff issued SECY-93-087, *Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs* [12]. Section II.N of SECY-93-087 presented the staff's position on 'Site-Specific Probabilistic Risk Assessments and Analysis of External Events.' This position considered the technical challenges posed in addressing site-specific hazards in a site-independent design certification process. These challenges arise from the requirement to perform a PRA to meet the requirements for design certification as set forth in 10 CFR Part 52, and the fact that PRA results are dependent on site characteristics that define these hazards as well as detailed and potentially site-specific design characteristics to protect against these hazards that are not known during design certification.

These challenges are noted in the following statements in this SECY paper:

> *'Details of the specific site characteristics will likely not be available until the COL review stage. The staff intends to require a COL applicant to perform a site-specific PRA that addresses all applicable site-specific hazards (such as river flooding, storm surge, tsunami, volcanism, or hurricanes). The staff will review the site-specific PRA to ensure that no vulnerabilities are introduced by siting the standardized plant at a location where external hazards could pose an unacceptable or unanticipated risk.*
>
> *In its letter of May 13, 1992, and in discussions with the staff, ACRS noted that the staff, an ALWR vendor, and a COL applicant may experience significant obstacles if design vulnerabilities from site-specific external events are discovered at the COL review stage. ACRS requested more information on how the staff proposes to deal with unacceptable findings, resulting from the site-specific PRA, identified during the COL application review process.*
>
> *In SECY-92-287, 'Form and Content for a Design Certification Rule,' the staff proposed the appropriate processes to modify Tier 1 and Tier 2 design certification information. For example, if a site-specific PRA identifies a serious generic design flaw that meets the 'adequate protection' threshold, the NRC can initiate rulemaking to amend the design certification rule. If the site-specific PRA identifies a site-specific design weakness, the COL applicant will have the option to request an exemption to the design certification rule to correct the deficiency.*
>
> *However, the staff will require that ALWR vendors perform bounding analyses of site-specific external events likely to be a challenge to a plant. When a site is chosen, its particular siting characteristics can then be compared to those of the bounding analyses in order to minimize the potential for the site-specific PRA to identify significant site-specific weaknesses for the standard design. Before certifying the design, the staff will evaluate fires, internal floods, and other external events that are not site dependent and will evaluate submitted bounding analyses for site—specific external events.'*

In their recommendations for the treatment of external hazards in PRAs performed for design certifications of LWRs, the staff concluded the following:

> *'Lessons from past risk-based studies indicate that fire, internal floods, and seismic events can be important potential contributors to core damage. However, the estimates of core damage frequencies for fire and seismic events continue to include considerable uncertainty. Consequently, the staff concludes that fire and*

seismic events can best be evaluated using simplified probabilistic methods and margins methods similar to those developed for existing plants, supported by insights from internal event PRAs (including ALWR design-specific PRAs). The designer should use traditional probabilistic techniques to study internal floods.'

'Fire events can be evaluated using simplified methods such as EPRI's Fire Induced Vulnerability Evaluation (FIVE) methodology rather than full scale PRAs. Ascribing to these methods, the designer should focus on the capacity of the design to withstand the effect of fire, using qualitative and quantitative methods rather than a strictly quantitative PRA fire analysis.'

On July 21, 1993, the Commission approved the following staff recommendations regarding 'Site-Specific Probabilistic Risk Assessments and Analysis of External Events' [13]:

'The Commission approves, in part, and disapproves, in part, the staff's position on site-specific probabilistic risk assessment and analysis of external events, as listed below.

The Commission approves the position that the analyses submitted in accordance with 10 CFR 52.47 should include an assessment of internal and external events.
The Commission disapproves the staff's recommendation to use two times the Design Basis SSE for margins-type assessment of seismic events.

The Commission approves the use of 1.67 times the Design Basis SSE for a margin—type assessment of seismic events.
The Commission approves the following staff recommendation, as modified:

> *PRA insights will be used to support a margins-type assessment of seismic events. A PRA-based seismic margins analysis will consider sequence-level High Confidence, Low Probability of Failures (HCLPF5) and fragilities for all sequences leading to core damage or containment failures up to approximately one and two-thirds the ground motion acceleration of the Design Basis SSE.*

The Commission approves the staff's position that the simplified probabilistic methods, such as but not limited to EPRI's FIVE methodology, will be used to evaluate fires.
The Commission approves the staff's position that traditional probabilistic techniques should be used to evaluate internal floods.

The Commission approves the staff's position that the ALWR vendors should perform bounding analyses of site-specific external events likely to be a challenge to the plant (such as river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes).
The Commission approves the staff's position that when a site is chosen, its characteristics should be compared to those assumed in the bounding analyses to ensure that the site is enveloped.

The Commission approves the staff's position that if the site is enveloped, the COL applicant need not perform further PRA evaluations for these external events. The COL applicant should perform site-specific PRA evaluations to address any site-specific hazards for which a bounding analysis was not performed or which are not enveloped by the bounding analyses to ensure that no vulnerabilities due to siting exist.'

Upon review of this guidance, it is noted that the focus of the PRAs for design certification of evolutionary and passive ALWRs was to meet the PRA requirements of 10 CFR Part 52, whose PRAs are performed to evaluate the risks of beyond design basis core damage events. As discussed more fully in Chapter 3, the uses and scope of the PBMR PRA go beyond the requirements of Part 52. The purposes of the PBMR PRA are much broader and include input to the selection of LBEs. This calls for a full-scope PRA treatment of all events and hazards so that the resulting event sequences can be addressed in a realistic and consistent manner. This is important for the intended use of the PRA to provide input to the definition the LBEs. Simplified treatment of one type of hazard relative to another is viewed as not appropriate for this PRA application. In addition, the risk metrics that will be used will not include CDF, but rather risk metrics that reflect the PBMR safety design approach.

After these differences are accounted for, the PBMR DCA PRA will face similar challenges as addressed in SECY 98-087; namely that the PRA will be based on a site-independent design and may exhibit external hazard contributions to the PRA results that may be different to those for a specific site. Hence, some assumptions and simplifications will be necessary to address certain events and hazards, including internal fires, seismic events, and other external hazards in order to complete the PBMR DCA PRA. It is recognized that such simplified approaches will not be able to address certain requirements of the available PRA standards that require knowledge of the as-built and as-operated details for a specific plant. This limitation is addressed by using deterministic criteria and conservative assumptions in using the PRA results in the selection of the LBEs as explained more fully in the LBE Selection paper. As noted in the Staff Requirements Memorandum (SRM), a site-specific PRA of external hazards need not be performed at the COL stage if the events are bounded in their treatment in certification design stage PRA. The approach to the treatment of internal and external events and hazards for the PBMR DCA is described in Chapter 3.

A review of the site-specific hazards and design details against the assumptions made in lieu of this information in the DCA PRA during the COL application stage is expected to be required by NRC in order to confirm the validity of the selection of LBEs at the COL stage. The focus of such a review should be to confirm that the PRA results as they impact the selection of LBEs are not adversely affected by the site-specific information and design details. That appears to be the logic that was used by the staff and the Commissioners in addressing this issue for the ALWRs and this logic applies to the PBMR for similar reasons.

The Advanced Reactor Policy Statement [14] states that for advanced reactors the Commission expects, as a minimum, at least the same degree of protection of the public and the environment that is required for current generation LWRs. This policy was reaffirmed in the SRM to SECY 2003-0047. Thus, the Commission expects that advanced reactor designs will comply with the Commission’s Safety Goals Policy Statement [15]. Furthermore, the Commission expects that advanced reactors will provide enhanced margins to safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety function. According to the Advanced Reactor Policy Statement, advanced reactor designers are encouraged as part of their design submittals to propose specific review criteria or novel regulatory approaches which NRC might apply to their designs. The design certification approach and the PRA proposed for the PBMR are specifically designed to meet the objectives set forth in these cornerstones of the NRC policies guiding the licensing of advanced reactors. Risk insights from the PRA are expected to be useful in demonstrating conformance to the safety goals and in demonstrating that increased reliance on inherent and passive means to accomplish safety functions in the PBMR is acceptable.

2.3 NRC GUIDANCE

There are a number of NRC guidance documents applicable to PRAs and the use of PRAs in risk-informed decision making. The PBMR PRA will conform to applicable sections of these documents and will identify those sections that do not apply, with justifications. This section identifies the key references which are considered to be applicable or partially applicable. The following paragraphs separately discuss each of the more important guidance documents.

2.3.1 Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-specific Changes to the Licensing Basis

Regulatory Guide 1.174 establishes an integrated process for the determination of acceptability of changes to design of existing plants using Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) considerations. This Regulatory Guide also provides a basis for using PRA in the design certification of PBMR with the exception of using CDF and LERF which are not appropriate as risk metrics for the PBMR. PBMR-specific risk metrics that relate to PBMR-specific event sequence end states as described in Chapter 3 of this paper will be used to provide risk management functions similar to those specified for these LWR-specific risk metrics in this guide.

Regulatory Guide 1.174 also addresses the need for consideration of defense-in-depth. Regulatory Guide 1.174 offers several considerations for ensuring that defense-in-depth is maintained in risk-informed decision making, including those to ensure:

- '*A balance between accident prevention and mitigation,*
- *No over-reliance on programmatic activities to compensate for weaknesses in plant design,*
- *System redundancy, independence, and diversity are employed,*
- *Potential common cause failures are minimize through the use of passive, and diverse active systems to support key safety functions,*
- *Barriers to radionuclide release are independent, and*
- *The potential for human errors is minimized'.*

The PBMR PRA and its use to support design certification decisions will address these issues by evaluating the roles of SSC in the prevention and mitigation of accidents, and by thoroughly evaluating the impacts of uncertainties on the PRA results and the design certification decisions that are derived from these results. Deterministic elements of the design certification approach, including a deterministic safety classification of SSC and the performance of conservative deterministic safety analysis of DBEs will be applied with a view toward addressing these defense-in-depth criteria.

Section 2.2.3 of this Regulatory Guide provides general information on the quality of PRAs. It states:

> *'The quality of a PRA analysis used to support an application is measured in terms of its appropriateness with respect to scope, level of detail, and technical acceptability. The scope, level of detail, and technical acceptability of the PRA are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process. The more emphasis that is put on the risk insights and on PRA results in the decision making process, the more requirements that have to be placed on the PRA, in terms of both scope and how well the risk and the change in risk is assessed.'*

In addition, Section 2 of Regulatory Guide 1.174 includes the following general principles related to PRAs:

'The plant-specific PRA supporting the licensee's proposals has been subjected to quality assurance methods and quality control methods.

Appropriate consideration of uncertainty is given in analyses and interpretation of findings, including using a program of monitoring, feedback, and corrective action to address significant uncertainties.

The use of core damage frequency (CDF) and large early-release frequency (LERF) as bases for PRA acceptance guidelines is an acceptable approach. Use of the Commission's Safety Goal quantitative health objectives (QHOs) in lieu of LERF is acceptable in principle, and licensees may propose their use. However, in practice, implementing such an approach would require an extension to a Level 3 PRA, in which case the methods and assumptions used in the Level 3 analysis, and associated uncertainties, would require additional attention.'

Section 2.2.5 of Regulatory Guide 1.174 states that the impact of three classes of uncertainty on the results of PRAs should be addressed: parameter uncertainty, model uncertainty and completeness uncertainty. Finally, Section 2.5 of the Regulatory Guide states that the following quality assurance requirements from Appendix B to 10 CFR Part 50 should be met to ensure that the PRA is sufficient to be used for regulatory decisions:

- *Use personnel qualified for the analysis.*
- *Use procedures that ensure control of documentation, including revisions, and provide for independent review, verification, or checking of calculations and information used in the analyses (an independent peer review or certification program can be used as an important element in this process).*
- *Provide documentation and maintain records.*
- *Use procedures that ensure appropriate attention and corrective actions are taken if assumptions, analyses, or information used in previous decision making are changed or determined to be in error.*

Consistent with Regulatory Guide 1.174, Standard Review Plan (SRP) 19.1 [16] provides guidance to the NRC on how to review a licensee's PRA findings and risk insights to support changes in an individual plant's licensing basis. The guidance in SRP 19.1 parallels the guidance in Regulatory Guide 1.174. PBMR will utilize these criteria to establish the technical adequacy of the PRA.

2.3.2 Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities

Regulatory Guide 1.200 [17] describes one acceptable approach for determining that the quality of a PRA is sufficient to provide confidence in the results such that the PRA results can be used in regulatory decision making. When used in support of a licence application, this Regulatory Guide is intended to obviate the need for an in-depth review of the PRA by NRC reviews.

Regulatory Guide 1.200 provides guidance in the following four areas:

a. A minimal set of functional requirements of a technically acceptable PRA

   The level of detail required of the PRA model is determined ultimately by the application. However, a minimum level of detail is necessary to ensure that the impact of designed-in dependencies (e.g. support system dependencies, functional dependencies, and dependencies on operator action) are correctly captured and the PRA represents the as-built, asperated plant. A complete PRA would include a Level 1 PRA to evaluate CDF and Level 2 PRA to evaluate LERF. Analyses should also include internal fires, internal floods, and external hazards.

b. PRA consensus standards and industry PRA programme documents

One acceptable approach to demonstrate conformance with Regulatory Position 1 (above) is to use an industry consensus PRA standard or standards that address the scope of the PRA used in decision making or an industry developed peer review program. These standards include the American Society of Mechanical Engineers (ASME) Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME RA-S-2002; the American National Standards Institute (ANSI)/American Nuclear Society (ANS) External-Events PRA Methodology, ANSI/ANS-58.21-2003; and the Nuclear Energy Institute (NEI) Probabilistic Risk Assessment Peer Review Process Guidance, NEI-00-02.

c. Demonstration that the PRA used in regulatory application is of sufficient technical adequacy

This demonstration is application–specific but generally includes adequate assurance that the PRA was performed in a technically correct manner and assurance that the assumptions and approximations used in developing the PRA are appropriate.

d. Documentation to support the regulatory submittal

The licensee must provide archival documentation that includes enough information to demonstrate that the scope of the PRA is sufficient to support the application and submittal documentation to demonstrate the technical adequacy including identification of key assumptions and approximations relevant to the results.

This Regulatory Guide provides useful guidance for establishing the adequacy of the PBMR PRA for risk-informed decision making. However, this guide and several of its key references such as the ASME and ANS PRA standards and the NEI PRA Peer Review Process were specifically developed for PRAs on currently licensed LWR plants. PBMR-specific risk metrics that are described in Chapter 3 of this paper will be used in lieu of the LWR-specific metrics CDF and LERF that are used in this Regulatory Guide. Although most of this guide and the associated standards and peer review approaches are applicable to the PBMR PRA, there are several areas where this guidance does not address certain quality issues that are important to the PBMR PRA. As discussed more fully in Chapter 3, there are significant differences in the way in which event sequences are modelled in PRAs for the PBMR and LWRs stemming from differences in the safety design approach. Secondly, this guide and its key references do not address the development of mechanistic source terms or the estimation of radiological consequences, especially for the PBMR.

Finally, there are a significant number of requirements suggested in this guidance and the supporting standards that were developed for PRAs on plants already built and do not apply to design certification PRAs which lack site-specific, design, and operational details. These requirements include those that address PRA model to plant fidelity issues. As explained more fully in Chapter 3, the assumptions that are made in lieu of this specific plant and site knowledge will be clearly documented and will be taken into account in the treatment of uncertainties and the application of conservative assumptions. There may be some PRA requirements such as those that require application of industry- and plant-specific service experience that will not be fully satisfied for certain parts of the PRA until a significant number of reactor years of PBMR operating experience has accumulated. As discussed more fully in Chapter 3, the PBMR PRA for the DCA will take this into account in the development of the PRA database, and in the treatment of uncertainties in quantification of event sequence frequencies.

2.3.3 Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

Regulatory Guide 1.183 [18] provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analysed risk; and content of submittals. While this regulatory guide refers to source terms that are specific to LWRs, it provides useful guidance in defining what is generally meant by the term ‘mechanistic source term’ and the resulting NRC expectations for the analysis. However, the PBMR source terms must deal with fundamental differences in fuel design, reactor materials and intrinsic design features, as well as different radionuclide transport phenomena. The technical basis for the PBMR mechanistic source terms is the topic of another paper. Such source terms will be needed to establish the radiological consequences of event sequences modelled in the PRA as well as to support the deterministic safety analysis of the DBAs in Chapter 15 of the DCA.

2.4 NRC PRECEDENTS INVOLVING GAS-COOLED REACTORS

2.4.1 Exelon PBMR Preapplication Review

In 2001 to 2002, the NRC staff conducted a preapplication review of the PBMR at the request of Exelon, who proposed to use a PBMR PRA to support licensing decisions [19]. In a letter to Exelon dated March 26, 2002 [20], the NRC Staff provided feedback on various technical, safety and policy issues raised by Exelon during preapplication reviews for the PBMR. With respect to the PRA, the staff stated as follows:

> *‘The staff supports Exelon’s plan to develop a full-scope, detailed PRA including internal events and external events (e.g., fires, earthquakes, floods, high winds) and to follow the fundamental applicable aspects of industry PRA standards (i.e., ASME, ANS). While such a PRA may not fit into the mold of the Level 1-2-3 framework, it can provide equivalent information regarding radiological consequences. However, the staff believes that further development of standards is necessary because the current ASME standard focuses on LERF analysis for Level 2 PRAs and does not address Level 3 PRAs. Although the ASME standard provides requirements for treating uncertainties, the lack of operating experience (e.g., initiating event frequencies, component reliability, phenomenology, fuel performance) to factor into the PBMR PRA will lead to relatively large uncertainties in the PRA results.’*

The staff further stated that:

> *‘…the PRA should include accidents involving spent fuel stored on site (analogous to spent fuel pool accidents in LWRs).’*

The PBMR PRA that will be used to support the DCA will address these staff comments from the Exelon PBMR preapplication review. The treatment of uncertainties will address the available PBMR service experience. The PRA will also account for all the sources of radioactive material that are included in the scope of the design certification, including the spent fuel stored on site and associated fuel handling and storage systems.

2.4.2 NUREG-1338, NRC Preapplication Review of MHTGR

The PRA approach proposed for the PBMR design certification builds upon the approach that was originally proposed for and applied to the Modular High Temperature Gas-cooled Reactor (MHTGR) as part of a design certification effort that was initiated by the US Department of Energy (DOE) [21]. The PBMR approach takes into account insights from NRC's review of the MHTGR Preliminary Safety Information Document (PSID) in NUREG-1338 [10]. The scope of the MHTGR effort and associated NRC review included:

- A PRA [28] that included:
  - MHTGR-specific initiating events, event sequences, and end states.
  - Fault tree models and data to estimate event sequence frequencies.
  - Plant transient response analysis for each event sequence.
  - Off-site dose consequences for each MHTGR-specific release category.
- A risk-informed licensing approach based on:
  - Current LWR requirements and the NRC safety goals.
  - LBEs derived from a PRA based on probabilistic and deterministic criteria including AOOs, DBEs and Emergency Planning Basis Events (EPBEs).
  - A method for selecting safety-related SSC based on probabilistic and deterministic criteria and application of the method to the MHTGR.
  - Regulatory design criteria for safety-related SSC during MHTGR-specific DBEs in the performance of MHTGR-specific safety functions.
- Deterministic safety analyses for all AOOs, DBEs and selected EPBEs.

Insights from the NRC review of the MHTGR preapplication submittal provide an important input to the development of the PBMR PRA. Although the MHTGR has significant design differences to the PBMR, this is another example of a modular HTGR that shares many of the inherent and passive features of the PBMR safety design approach. Insights from the MHTGR PRA were used to develop the PBMR PRA, and issues raised in the NRC review were specifically taken into account in developing the PBMR PRA models. The risk-informed approach was exercised with a conceptual design and NRC design review, a design-specific PRA and its NRC review, a specific set of LBEs, and independent analyses by NRC and NRC contractors. This PRA and its review are expected to provide useful background to the NRC in the review of the PBMR DCA. Although the DOE submittal and the NRC review had to be terminated before firm regulatory decisions could be finalized, these references provide concrete examples of how PRA was used to support the selection of LBE for a specific modular HTGR design using an approach that is similar to that which is proposed for the PBMR DCA. The NRC review report provides useful guidance on potential issues that will likely need to be addressed in the preparation and review of the PBMR DCA.

3. PBMR APPROACH

The technical approach to performing the PBMR PRA is described in this chapter to the extent necessary to identify potential issues that need to be addressed and resolved prior to the DCA. This chapter describes how standard state-of-the-art PRA methods will be applied to the PBMR, how the PBMR safety design approach will be reflected in the definition and modelling of event sequences, how certain technical issues for performing a PRA on a new reactor with this safety design approach will be addressed, and how the adequacy of the PRA quality for use in support of licensing decisions is established. When comparing the PBMR PRA to PRAs on currently licensed and operating LWRs, there are many points in common and some significant points of departure. The purpose of this paper is to identify both the aspects in common as well as the significant points of departure, with a view towards a successful PBMR DCA review. The purpose of this chapter is not to provide a detailed description of the PRA methodology and its modelling details, but rather to provide a high-level summary of the key issues for the PRA in preparation for future workshops where some of the details can be more fully discussed.

3.1 OVERVIEW OF PBMR PRA

PRA provides a logical and structured method to evaluate the overall safety characteristics of the PBMR plant. This is accomplished by systematically enumerating a sufficiently complete set of accident scenarios, and by assessing the frequencies and consequences of the scenarios individually and in the aggregate to predict the overall risk profile. The PRA is selected as a tool to help identify the LBEs in part due to its capabilities to account for the dependencies and interactions among SSC, human operators and the internal and external plant hazards that may perturb the operation of the plant and lead to an accidental release of radioactive material. Rather than limit the quantification to point estimates of selected risk metrics, the PBMR PRA will be structured to give emphasis to the treatment of uncertainties. The quantification of both frequencies and consequences of event sequences and sequence families address uncertainties through the performance of quantitative uncertainty analysis where information is available to perform this function and sensitivity analyses to address other sources of uncertainty that are more difficult to quantify. The treatment of uncertainties for the PBMR will address the available applicable reactor service experience. The quantification of frequencies and consequences of event sequences and the associated quantification of uncertainties provide an objective means of comparing the likelihood and consequence of different scenarios and of comparing the assessed level of safety against the applicable requirements. The sources of uncertainty that are identified in the uncertainty analysis are given visibility for deterministic treatment in the selection of LBEs and in the development of regulatory design criteria. The PRA is structured to be able to examine the risk significance of design features and SSC in the performance of safety functions as called for in the NRC Advanced Reactor Policy Statement [14].

3.2 RATIONALE FOR USE OF PRA

PRA is selected as an analysis tool in order to:

- Comply with NRC regulations and guidance associated with the performance of PRA for design certification.

- Provide a systematic identification of an adequately complete set of initiating events and event sequences that will provide a basis for the quantification of risk to public health and safety, and serve as an appropriate and acceptable input to the selection of LBEs.
- Provide a systematic examination of dependencies and interactions, and the role that SSC and operator actions play in the development of each accident scenario; this examination will have the capability to display the cause and effect relationships between the plant characteristics and the resulting risk levels that are sufficient to support the identification of LBEs and the safety classification of SSC.
- Provide quantitative estimates of accident frequencies and consequences under a realistic set of assumptions that can be supported by available data, expert opinion and other scientific evidence.
- Define an appropriate set of PBMR-specific risk metrics from the information provided in the PRA that can be used to demonstrate that the principles of defense-in-depth have been applied and that there is a reasonable consideration of the prevention and mitigation of potential accidents for this type of reactor.
- Address uncertainties through quantification of the impact of identifiable sources of uncertainty on the results, and by appropriately structured sensitivity studies to examine the risk significance of key issues. Provide input to the development of deterministic requirements that address uncertainties and defense-in-depth considerations.
- Support appropriate and, where required, conservative decision making through the examination of uncertainty distributions.
- Provide a reasonable and acceptable degree of completeness in the enumeration of event sequences and the treatment of appropriate combinations of failure modes, including consideration of the potential for multiple failures necessary to determine risk levels, identify LBEs and perform safety classification of SSC.
- Determine the cause and effect relationships between elements of the safety design approach and the risk profile. This includes the risk significance of SSC and design features in order to support the selection of LBEs and perform safety classification of SSC.
- Provide insights into the provision of special treatments of SSC commensurate with their safety significance in any given event sequence.

All key assumptions that are used to develop success criteria, to develop and apply probability and consequence models, and to select elements for incorporation into the models will be clearly documented and possible to scrutinize. In addition, assumptions that are made in lieu of as-built and as-procured characteristics for the PBMR will be clearly identified and documented.

3.3 OBJECTIVES OF PBMR PRA

The objectives of the PBMR PRA are:

- Provide risk insights into the design of the PBMR, including the design of SSC that perform safety functions sufficient for the DCA.
- Provide an acceptably complete set of event sequences from which to select the LBEs for the DCA.
- Confirm that the applicable requirements, including the safety goal Quantitative Health Objectives (QHOs) for individual and societal risks, are met at a representative US site or sites.

- Provide input for the development of PBMR-specific regulatory design criteria for the plant.
- Support the determination of safety classification and special treatment requirements of SSC.
- Support the identification of emergency planning specifications including the location of the site boundary.
- Support the development of technical specifications.
- Provide insight on the role of PBMR SSC in the prevention and mitigation of event sequences and risk insights to evaluate the design defense-in-depth.
- Determine the risk significance of design features and SSC to the extent needed to support LBE selection and safety classification of SSC.

NRC agreement on these objectives is identified as an important outcome of this paper.

3.4 SCOPE OF PBMR PRA

The PRA will provide a primary source of candidate event sequences for the selection of LBEs and will be a key input to the safety classification of SSC. In view of this application, completeness and accuracy in the enumeration of event sequences are viewed as especially important attributes. In addition, the emphasis that is placed on the roles of inherent and passive capabilities to affect the safety design approach of the PBMR is viewed as requiring that a comprehensive scope of challenges to the PBMR passive SSC be included. Such a comprehensive scope includes a full spectrum of internal events and external hazards that pose challenges to the inherent and passive capabilities of the plant. As such, the scope of the PRA will be adequately complete in the treatment of events and hazards and consistent with the state-of-the-art of PRA technology to support the intended uses in selecting LBEs for the PBMR.

The PRA includes the following aspects of a full-scope PRA:

- The potential sources of release of radioactive material, including the sources in the reactor core and Main Power System (MPS), process systems, and Fuel Handling and Storage System (FHSS).
- All planned operating and shutdown modes, including plant configurations expected for planned maintenance, tests and inspections.
- A full range of potential causes of initiating events, including internal plant hardware failures, human operator and staff errors, internal plant hazards such as internal fires and floods, and external plant hazards such as seismic events, transportation accidents and any nearby industrial facility accidents.
- Event sequences that cover a reasonably complete set of combinations of failures and successes of SSC and operator actions in the performance of PBMR-specific safety functions. These event sequences will be defined in sufficient detail to characterize mechanistic source terms and off-site radiological consequences comparable to an LWR Level 3 PRA as defined by NUREG/CR-2300 [22].
- Quantification of the frequencies and radiological consequences of each of the significant event sequences modelled in the PRA. This quantification includes mean point estimates and an appropriate quantification of uncertainty in the form of uncertainty probability distributions that account for quantifiable sources of uncertainty in the accident frequencies, mechanistic source terms, and off-site radiological consequences. Additionally, an

appropriate set of sensitivity analyses will be performed to envelope sources of uncertainty that are not quantifiable.

- For PBMR plants covered under the DCA that are comprised of multiple reactor modules, the PRA will define event sequences that impact reactor modules independently, as well as those that impact two or more reactor modules concurrently. The frequencies will be calculated on a per plant year basis, and the consequences will consider the number of reactor modules and sources that are involved in the definition of the mechanistic source terms.
- In order to support the development of regulatory design criteria, the PRA will be capable of evaluating the cause and effect relationships between design characteristics and risk, and of supporting a structured evaluation of sensitivities to examine the risk impact of adding and removing selected design capabilities, and setting and adjusting SSC reliability requirements.

The scope of the PBMR PRA to support this risk-informed approach will be comprehensive, complete, and comparable to a full-scope, all modes, Level 3 PRA for an LWR covering a full set of PBMR internal and external events. However, due to the inherent features of the PBMR, the approach to modelling initiating events and event sequences will be simplified in terms of size and complexity in comparison to that in an LWR PRA model. These simplifications will not diminish the quality of the PRA, and will facilitate the capability to perform effective independent peer reviews as needed to meet ASME and ANS PRA standards' peer review requirements.

The PBMR PRA model is structured somewhat differently than the traditional Level 1-2-3 model for an LWR PRA as defined in NUREG/CR-2300, for several reasons. There is nothing comparable to a Level 1 PRA for the PBMR, because there is no plant state comparable to 'core damage'. The available definitions of 'core damage' are specific to LWRs, are not applicable to the PBMR, and hence there is no CDF risk metric. Such core damage states are precluded by the PBMR safety design approach as to be verified by deterministic calculations in the DCA. While the PBMR does not calculate any CDF metrics, it does calculate PBMR-specific plant state frequencies that correspond with the LBEs, as explained more fully elsewhere in this paper.

If one were to define a Level 2 PRA as a PRA that includes the frequencies of reactor-specific release categories and mechanistic source terms for these release categories, a Level 2 PRA can be defined for the PBMR. However, such a PRA would not include the calculation of a CDF risk metric.

Another simplification in the PBMR PRA model structure stems from the relative simplicity of the PBMR in terms of the number of SSC and events that need to be modelled. This factor lends itself to the capability to define a continuous event sequence development spanning from initiating events to release categories for which mechanistic source terms and radiological consequences can be calculated. For organizing the computer model, this continuous event sequence model may be broken up into different stages of event trees to represent the different responses of the plant systems and structures, and in this respect the PBMR PRA model may exhibit some similarities with the classic Level 1-2-3 LWR PRA structure. However, in the end, the elements of the PBMR PRA are combined into a single, event sequence model framework that starts with initiating events occurring in different plant operating states, and ends in PBMR-specific event families and release categories. A given release category will contain one or more event families for which frequencies, mechanistic source terms and off-site consequences are calculated. The integral PBMR PRA encompasses the functions of a full-scope Level 1-2-3 LWR

PRA. However, there are some simplifications for the DCA PRA that are expected to justify modification of some of the elements of a full Level 3 PRA. For example, it is expected that radiological consequence analysis will be limited to the performance of site boundary dose calculations with very simple conservative models to treat the risks of off-site health effects and property damage.

A comparison of the PBMR PRA structure for the DCA with that of the Level 1-2-3 framework for an LWR is provided in Table 1. As seen in this table, there are both similarities and differences in the modelling structures.

The treatment of all operating and shutdown modes for the PBMR is also simplified relative to that of an LWR, due to fewer plant configurations that give rise to unique success criteria and event sequence modelling end states. This is in part due to the use of a single phase gas for the reactor coolant and the capability of SSC for active and passive heat removal and reactivity control to perform their functions during both pressurized and depressurized conditions. For the PBMR, the more limited set of SSC that need to function in order to fulfil the key safety functions for the events involving the potential release of radionuclides in the reactor core and the MPS helps to reduce the size of the PRA event tree/fault tree models, so the need to develop separate models for different configurations and success criteria is minimized. The use of an online refuelling scheme precludes the need to create separate plant configurations to routinely refuel the reactor. However, there is a need for the PBMR to address sources of radioactive material outside the reactor vessel, such as those within in the fuel handling and storage systems and plant configurations in which the reactor is defuelled to perform certain unscheduled maintenance actions. Finally, the knowledge that all operating and shutdown modes need to be considered is known upfront, and so the low power and shutdown mode treatment is taken into account in the baseline PRA models rather than after the fact.

In the PBMR case, the PRA will include event sequence models for other sources of radioactive material such as the spent fuel storage system, and the used fuel storage that is used in the unlikely event that the reactor vessel is defuelled, the event sequence models of which are expected to be unique to those conditions.

The modular aspect of the PBMR creates the potential for multiple reactor modules to be located at the same site, with some systems shared between the modules. The PRA accounts for the risk of multiple modules. The existence of multiple modules increases the site-wide likelihood of scenarios that impact a single module independently, and creates the potential for scenarios that involve multiple modules as well as the potential for a mechanistic source term involving two or more reactors. These modular reactor considerations will impact the scope and level of detail of the PRA.

NRC agreement on the necessary scope and structure of the PRA is identified as an important outcome of this paper.

Table 1: Comparison of PBMR PRA with LWR PRA Model Structure

<table>
<tr><th rowspan="2">PRA Model Attributes</th><th rowspan="2">PRA Outputs</th><th colspan="3">LWR</th><th rowspan="2">PBMR</th></tr>
<tr><th>Level 1</th><th>Level 2</th><th>Level 3</th></tr>
<tr><td rowspan="4">Assessment of Accident Frequencies</td><td>Core Damage Frequency (CDF)</td><td>Yes</td><td>Yes</td><td>Yes</td><td>No</td></tr>
<tr><td>Plant Damage State (PDS) Frequencies</td><td>Not necessary but are sometimes included</td><td>Yes, all PDSs are variations of LWR core damage state</td><td>Yes, all PDSs are variations of LWR core damage state</td><td>Yes, PBMR plant states defined as event sequence families for LBEs</td></tr>
<tr><td>Release Category Frequencies</td><td>No</td><td>Yes, all involve core damage and are LWR specific</td><td>Yes, all involve core damage and are LWR specific</td><td>Yes, PBMR-specific release categories</td></tr>
<tr><td>Frequencies of site meteorological conditions and Emergency Planning (EP) responses</td><td>No</td><td>No</td><td>Yes</td><td>Yes, but conservative deterministic bounding treatment may be sufficient to meet DCA requirements</td></tr>
<tr><td rowspan="2">Assessment of Accident Consequences</td><td>Source Terms</td><td rowspan="2">No</td><td>Yes, Mechanistic Source Terms for each LWR release category quantified</td><td>Yes, Mechanistic Source Terms for each LWR release category quantified</td><td>Yes, Mechanistic Source Terms for each PBMR release category quantified</td></tr>
<tr><td>Consequences to the Public</td><td>No</td><td>Yes, early and latent health effects, property damage impacts quantified</td><td>Yes, like LWR Level 3, but conservative bounding treatment may be sufficient to meet DCA requirements</td></tr>
</table>

3.5 PBMR PRA ELEMENTS

The PBMR PRA has been organized into elements that are consistent with the way in which PRA elements have been defined in the ASME [23] and ANS PRA Standards [24], [25], and [26] and Regulatory Guide 1.200 [17]. The PRA elements, which may be considered building blocks of the PRA models include:

- Definition of Plant Operating States
- Initiating Events Analysis
- Event Sequence Development
- Success Criteria Development
- Thermal and Fluid Flow Analysis
- Systems Analysis
- Data Analysis
- Human Reliability Analysis
- Internal Flooding Analysis
- Internal Fire Analysis
- Seismic Risk Analysis
- Other External Events Analysis
- Event Sequence Frequency Quantification
- Mechanistic Source Term Analysis
- Radiological Consequence Analysis
- Risk Integration and Interpretation of Results
- Peer Review

The role these elements play in the development and quantification of the PBMR event sequence model is illustrated in Figure 1. These elements are similar to those associated with a full-scope Level 3 PRA for an existing LWR. Some of the key differences identified are as follows:

- The following PRA elements are design-specific and are developed specifically for the PBMR:
  - Safety functions
  - SSC to support each function
  - Success criteria
  - Functional initiating event categories
  - Plant response to initiating events
  - Human actions prior to, in the initiation of, and in response to events modelled in the PRA, including the time frames available for these actions
  - Event sequence end states
  - Mechanistic source terms
  - Radiological consequences
- The event sequences cover frequent events that are classified as AOOs, infrequent events that are classified as DBEs, and rare events that are classified as BDBEs.

- There is no calculation of a CDF, or LERF, but rather calculations of the frequencies and consequences of accident families referred to as Licensing Basis Events (LBEs). For some calculations such as those to demonstrate the plant performance against the NRC safety goal QHOs, the results are organized into PBMR-specific release category frequencies.
- Event sequence frequencies are calculated on a per plant-year, where a plant may consist of up to eight reactor modules to support an integrated treatment of risk. The consequences of event sequences may involve source terms from one, multiple, or all eight reactor modules. This will facilitate the definition of LBEs for the multi-module design and provide the capability to address the integrated risk of the multi-module plant.

The frequency-dose criteria described in the LBE Selection paper are expressed in terms of site boundary radiological doses, which do not require complex heath effects and evacuation models in their calculations. Because the mechanistic source terms are very small, the treatment of off-site radiological consequences may be simplified in relation to that in a typical LWR Level 3 PRA. As has been demonstrated in prior PRAs on HTGRs ([27] and [28]), mechanistic source terms for HTGRs are far below the dose thresholds necessary to produce early health effects. Evidence will be presented in the DCA PRA to show that this is indeed the case with the PBMR. In addition, as shown in [28], bounding estimates can be made to demonstrate that the NRC safety goal QHOs have been met without the need for complex health effects and evacuation models. Hence, complex evacuation and population dose models are generally not required for the PRA, and site boundary doses that do not credit evacuation are expected to be sufficient for this PRA application. Some of the key features of the PBMR PRA modelling process are described in paragraph 3.6.

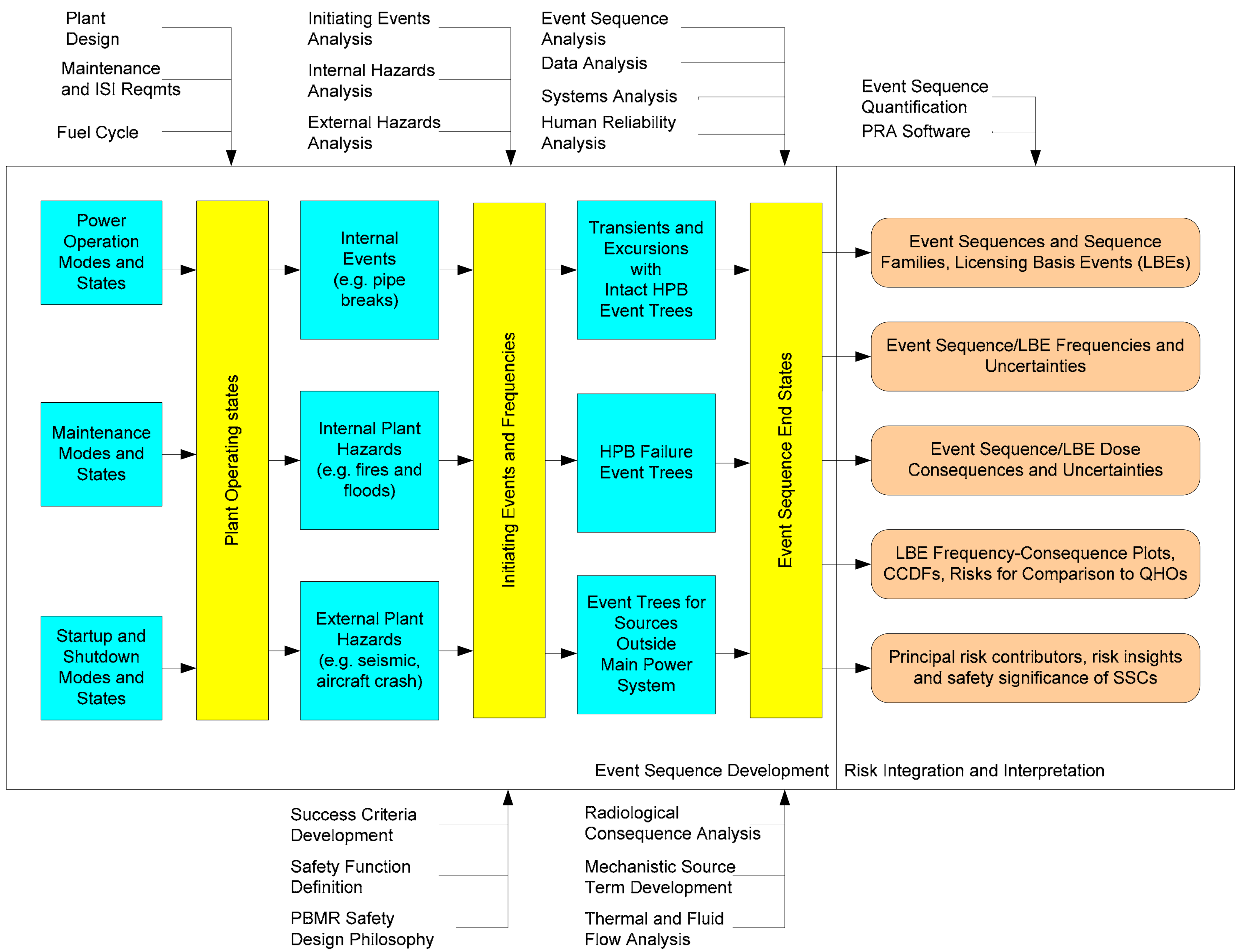


Figure 1: Overview of PBMR PRA Model Elements

3.6 TECHNICAL APPROACH TO MODELLING PBMR EVENT SEQUENCES

3.6.1 Systematic Search for Initiating Events

An important element of the PBMR PRA is the systematic approach to the search for initiating events, which begins the process of event sequence modelling. The approach to the performance of this task is derived from previous HTGR PRAs such as the MHTGR PRA [28], and is consistent with the approaches used in contemporary LWR PRAs. The initial conditions for the selection of initiating events for the PBMR cover all operating and shutdown modes expected during the PBMR operating life, including the expected shutdown configurations for conducting maintenance and refuelling configurations. A structured process known as the Master Logic Diagram method is used to ensure that a reasonably complete enumeration of initiating events appropriate for the PBMR is accomplished.

As shown in Figure 2, the process starts with the identification of the sources of radioactive material, barriers, safety functions, and initial plant operating states. There is no *a priori* assumption to limit the coverage of radionuclide sources to that inside the reactor core as in LWR PRAs. All sources of radioactivity are considered in the PBMR PRA, including the circulating coolant activity, activity plated out on MPS internal surfaces, and activity in intact, as well as damaged and defective, fuel particles contained in the reactor core and spent fuel storage systems.

For the PBMR, the following sources of radioactive material are considered:

a. Sources within the MPS HPB
   - Fuel spheres in core/FHSS
     - Intact coated particles
     - Failed or defective coated particles
     - Uranium contamination outside coated particles
     - Imbedded/attached to graphite components
   - Plateout on HPB surfaces and dust
   - Circulating coolant activity

b. Sources outside the MPS HPB
   - Fuel spheres in storage systems
   - HICS and HPS gas-borne activity
   - Solid and liquid radwaste systems

The principal barriers to each of these sources are summarized in Table 2.

Table 2: PBMR Sources and Barriers

Radioactive Material Source	Barriers to Radionuclide Transport
Fuel spheres in the core	Coated particles, graphite matrix, Helium Pressure Boundary, Citadel, reactor building
Fuel spheres outside the core	Coated particles, graphite matrix, Fuel Handling and Storage System (FHSS) piping, Spent Fuel Tanks (SFTs), Used Fuel Tanks (UFTs), or new fuel tanks, reactor building
Non-core sources within the MPS	Helium Pressure Boundary, reactor building
Other sources	Various tanks, piping systems and containers, reactor building or ancillary buildings housing waste management equipment

Once the sources, barriers, and safety functions are defined, the Master Logic Diagram follows a step-by-step process of defining the failure modes of each SSC and the impacts of these modes in challenging the barriers and safety functions, and of identifying direct initiating events, as well as challenges posed by internal and external hazards. Two separate paths are followed through these steps on the diagram (Figure 2), one from the point of view of each barrier and its set of challenges, and the other from the point of view of the SSC providing safety functions in support of these barriers. The former may be viewed as direct challenges to the integrity of the barriers and the latter as indirect challenges to the barriers.

An initial screening is performed for all SSC in the plant, including the radionuclide transport barriers. SSC that play no direct or indirect role in supporting a safety function and whose failure does not impact the safety functions of other SSC or cause an initiating event are screened out. Failure modes and effects analyses are performed for all unscreened SSC and transport barriers to identify potential internal initiating events. An analysis of internal and external plant hazards is performed to encompass the remaining challenges to the plant safety functions. These processes ensure that events specific to the PBMR design are considered. Insights from reviews of nuclear plant operating experience and previous safety and risk analyses are used to ensure completeness of the exhaustive list of events. In the design and certification of the PBMR, the systematic selection of initiating events is viewed as common to both the probabilistic and deterministic elements of the safety analysis approach. This fact is important to understand the way in which deterministic and probabilistic elements have been integrated into the PBMR design, which is the key advantage of applying PRA technology in the beginning.

3.6.2 PBMR SSC Providing Safety Functions

The PBMR PRA will include a set of reactor-specific safety functions and will define the PBMR SSC that are available or potentially available to perform these safety functions. The purpose of this paragraph is to describe the basis for defining the safety functions to be modelled in the PBMR PRA, and for selecting the SSC to be modelled in the performance of these safety functions.

The exhaustive set of initiating events determined in Step 6 is grouped according to the nature of the challenges to the PBMR safety functions. PBMR safety functions have been defined in the context of a top-down logical structure, starting with the high-level function of controlling the transport of radionuclides. Such transport is fundamentally controlled in the safety design approach by preserving the integrity of the radionuclide transport barriers as illustrated in Figure 2.

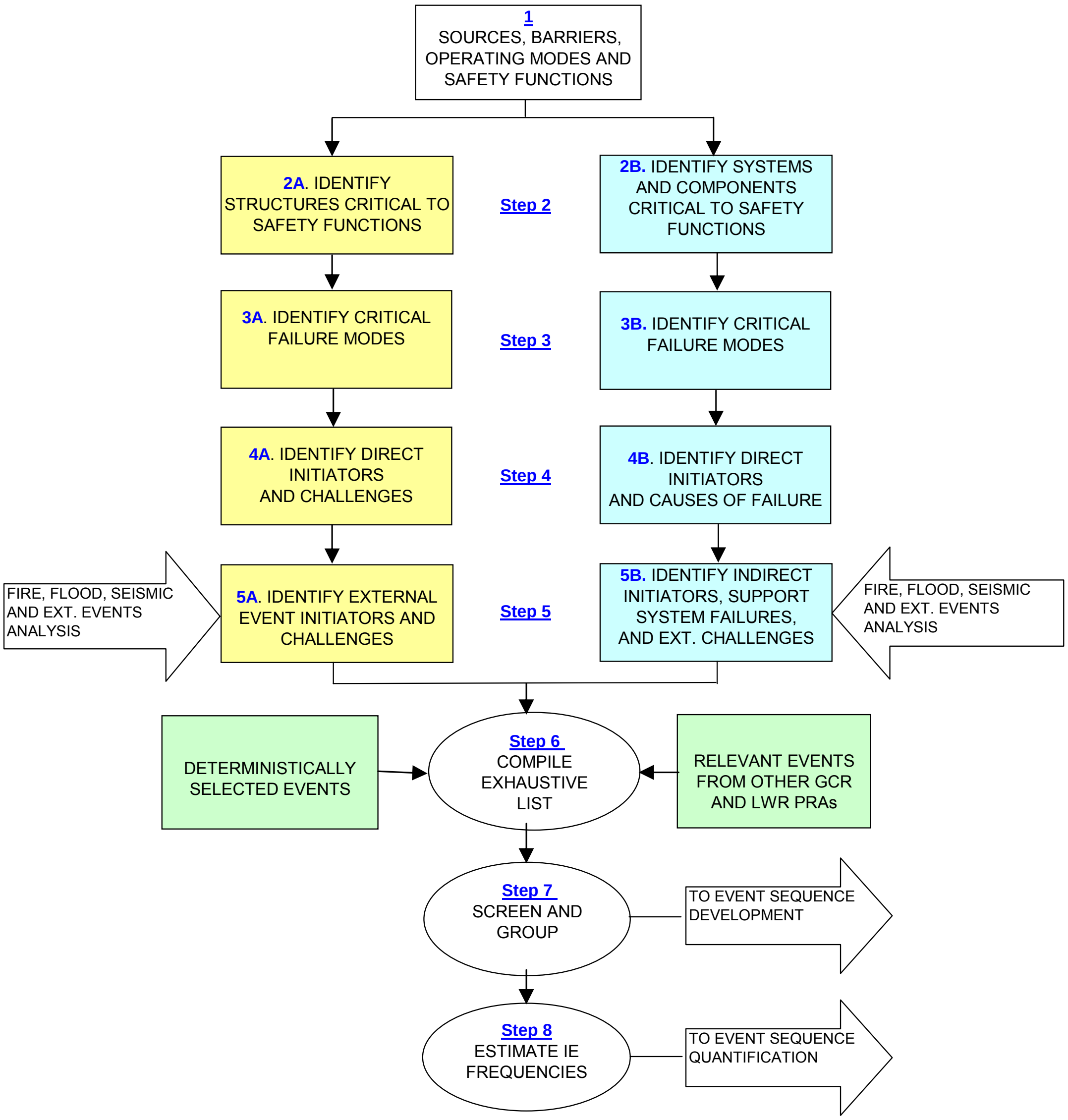


Figure 2: Master Logic Diagram for PBMR Initiating Events Analysis

Both inherent and engineered (i.e. other than inherent) safety features and SSC are included in the design to perform the safety functions. Engineered safety features include both passive and active SSC. Consistent with good PRA practice, the PBMR safety functions modelled in the PRA include those that are required to meet the minimum safety requirements, i.e. the ‘required safety functions’, as well as ‘supportive safety functions’. Supportive safety functions are performed by SSC that are included to meet availability and investment protection needs and serve defense-in-depth roles by preventing and mitigating challenges to barriers and SSC performing the required safety functions. The PBMR safety design philosophy utilizes inherent safety features and passive SSC to perform the required safety functions. Active SSC are also provided for supportive safety functions as well as to meet plant investment protection and availability performance criteria. SSC that serve both required and supportive safety functions are included in the PRA in order to capture a sufficiently complete set of safety function challenges and associated event sequences, and to apply the principle of realistic PRA success criteria. The process of using safety functions to develop the event sequences is fundamentally the same process as used in LWR PRA. The need to model both safety and non-safety classified SSC is also no different; only the functions and the SSC are differerent. Once the differences in safety functions and the SSC that provide these functions are understood, the capability to review the PRA event sequence model is available.

The safety functions for the PBMR include:

- Maintain control of radionuclides.
- Control heat generation (reactivity).
- Control heat removal.
- Control chemical attack.
- Maintain core and reactor vessel geometry.

A summary of the inherent features and passive SSC as well as the active SSC that support or provide defense-in-depth for the safety functions for the PBMR is provided in Table 3. This table defines the scope of SSC modelled in the PBMR PRA.

The following examples of functional initiating event categories have been defined for the PBMR PRA for the sources of radioactive material inside the reactor vessel and the MPS pressure boundary. Functional initiating event categories are defined by the nature of the challenge to the safety functions and these categories are used to decide which different event sequence models need to be developed.

- Power Conversion Unit (PCU) Transients with intact Helium Pressure Boundary (HPB)
    - PCU and Core Conditioning System (CCS) still capable of forced cooling operation
    - PCU failed, CCS still capable of operation
    - CCS failed, PCU still capable of operation
    - PCU and CCS not capable of operation
- PCU Transients with intact HPB and reactivity addition
    - Control rod or group withdrawal
    - Removal of Reserve Shutdown System (RSS) Small Absorber Spheres (SAS)
    - Overcooling transients
- HPB Leaks and Breaks (excluding HPB Heat Exchanger (HX) failures)
    - Small HPB failures resulting in slow depressurization < 10 mm break size
    - Moderate HPB failures resulting in rapid depressurization with break size > 10 mm and < 230 mm
    - Large HPB failures resulting in rapid depressurization with break size > 230 mm
- HPB HX Failures
    - Pre-cooler or Intercooler failures
    - CCS heat exchanger failure
    - Core Barrel Conditioning System (CBCS) heat exchanger failure

Each of the above categories represents a unique challenge to the PBMR required and supportive safety functions. These categories are used as a starting point for the development of event sequence models as described in paragraph 3.6.3.

For each of the above categories, specific initiating events or causes of initiating events can be defined having the same functional challenge to the safety functions. For example, one cause of a loss of PCU transient is loss of off-site power, and another is Power Turbine Generator Trip. Seismic events that do not cause a breach of the HPB are included as PCU transients, while those that do are included in the HPB leaks and breaks category. As part of the PRA that is submitted to support the DCA, the comprehensive treatment of initiating events and how they are dispositioned by screening and grouping will be documented according to applicable PRA guides and standards.

Table 3: PBMR Major Systems, Structures, and Components Modelled in the PRA

Safety Function	Inherent Features and Passive SSC	Active SSC[1]
Control of Radionuclides	• Fuel barrier - Coated particle barrier - Graphite matrix - Graphite reflectors and other core surfaces • Helium Pressure Boundary (HPB) barrier • Reactor building barrier - Confinement functions of reactor building - Reactor building Pressure Relief System (PRS) blowout panels	• PRS dampers • Reactor building Heating, Ventilation and Air-conditioning (HVAC) filtration system
Control of Heat Generation	• Strong negative temperature coefficient of reactivity • Reduced excess reactivity due to online refuelling • Gravity fall of control rods and Small Absorber Spheres (SAS)	• Control and protection systems - Operational Control System (OCS) - Equipment Protection System (EPS) - Reactor Protection System (RPS) • Reactivity control systems - Reactivity Control System (RCS) trip release of control rod drives - Reserve Shutdown System (RSS) release of SAS

[1] Not shown in this table are support systems such as electric power systems, instrument and service air systems, and some of the man-machine interface systems.

Safety Function	Inherent Features and Passive SSC	Active SSC[1]
Control of Heat Removal	• Large thermal heat capacity • Passive core heat removal • Core size, power density, geometry • Core, un-insulated reactor vessel, and reactor cavity configuration • Passive Reactor Cavity Cooling System (RCCS) • RCCS Tank inventory • RCCS Tank inventory + Demineralized Water System (DWS) or Fire Protection System (FPS) makeup (two places) • RCCS Tank inventory + External tank truck makeup (two places) • RCCS dry • PRS blow-out panels	• Active Reactor Cavity Cooling System (RCCS) - Equipment Protection Cooling Circuit (EPCC) → Main Heat Sink System (MHSS) - EPCC → Cooling Tower • Power Conversion Unit (PCU) - Brayton Cycle → Active Cooling System (ACS) → MHSS - Motored Turbine Generator (TG) → ACS → MHSS • Core Conditioning System (CCS) - EPCC → MHSS - EPCC → Cooling Tower • Core Barrel Conditioning System (CBCS) - EPCC → MHSS - EPCC → Cooling Tower
Control Chemical Attack	• HPB high reliability piping and pressure vessels • HPB design minimize penetrations in top of reactor vessel • High purity specifications for inert helium coolant • All interfacing systems at lower pressure than Main Power System (MPS) • Lack of HPB pressurization mechanisms to open PRS valves • ACS rupture discs protect against MPS Heat Exchanger (HX) leaks • PRS relief blowout panels	• PRS exhaust duct dampers limit air ingress • Isolation valves in MPS interfacing systems • Helium Purification System (HPS) maintains high purity levels of Helium coolant
Maintain Core and Reactor Vessel Geometry	• Reactor core and structures • Reactor pressure vessel and structures • Reactor cavity citadel • Reactor building structure	• Active RCCS maintains acceptable reactor vessel support temperatures

3.6.3 Development of Event Sequence Models

Once functional categories of the initiating events are established, event sequence diagrams and event trees are developed to define event sequences resulting from each initiating event and initial condition to be modelled. For each functional initiating event category, the event trees will be quantified for each specific initiating event in that category in order to account for significant dependencies between the causes of the initiating event and the modelled SSC failure probabilities. The event tree top events will be derived in consideration of the SSC that are provided to support each of the above safety functions. The event sequences define the possible successes and failures of each SSC to implement each safety function to a sufficient extent to determine the event sequence end states. This use of event trees is similar to that in an LWR PRA, the difference being that the end states and success criteria for terminating event sequences are developed specifically for HTGRs and for the specific design characteristics of the PBMR.

The treatment of operator actions in the modelling and quantification of event sequences follows the same process as for LWR PRAs. The following are the major differences in Human Reliability Analysis (HRA) treatment in the DCA PRA:

- Due to the safety design approach of the PBMR, there are few operator actions that must be fulfilled in order to achieve a safe stable end state to an event sequence.
- In general, the time windows available to implement the operator actions in the PRA model are very long. The application of existing HRA techniques that recognize the dependence of the human error rate on the time window may result in human error rates that are too small to be verifiable or appear credible. This is expected to result in a conservative treatment of human error rates in relation to that which would be considered realistic. This is not viewed as a problem for the PBMR PRA as the PRA results are not that sensitive to the assumed human error rates, because most of the important safety functions are fulfilled without need for time critical operator actions. Hence use of conservative human error rates is not expected to skew risk insights.
- Since there is less PRA experience in performing HRA in PRAs for reactors such as the PBMR, it is expected that the uncertainties in the human error rates will be larger than found in typical LWR PRAs. For the same reasons cited above regarding the use of conservative human error rates, the assignment of large uncertainties is not viewed to adversely impact the PRA results or their use in selecting LBEs.
- Due to the use of the PRA in selecting LBEs and the greater reliance on inherent and passive means to fulfil safety functions in the PBMR, there will be increased emphasis on the treatment of human errors of commission in the PBMR PRA.
- At the DCA design stage, some of the details of the emergency operating procedures and man-machine interface will be unknown. This will be taken into account in the human error rate uncertainty analysis and will tend to increase the uncertainties. As noted above, this is not expected to cause a problem in terms of masking risk insights or adversely impacting the capability of the PRA to support LBE selection.

Figure 3 depicts the event sequence modelling framework for the PBMR. This framework includes the following elements:

- Initiating event.
- Plant response to initiating event.

- Response of the reactor building and associated SSC.
- Factors influencing the end state, including achievement of success criteria and mechanistic source terms.

The causes of the initiating events depicted in Figure 3 include internal plant hardware failures, human errors, internal plant hazards such as fires and floods, and external hazards such as seismic events and transportation accidents. The responses of the plant and reactor building functions include the responses of SSC and the human operators that are involved in the performance of or failure to perform each function. Human responses include favourable or unfavourable acts and errors of omission and commission.

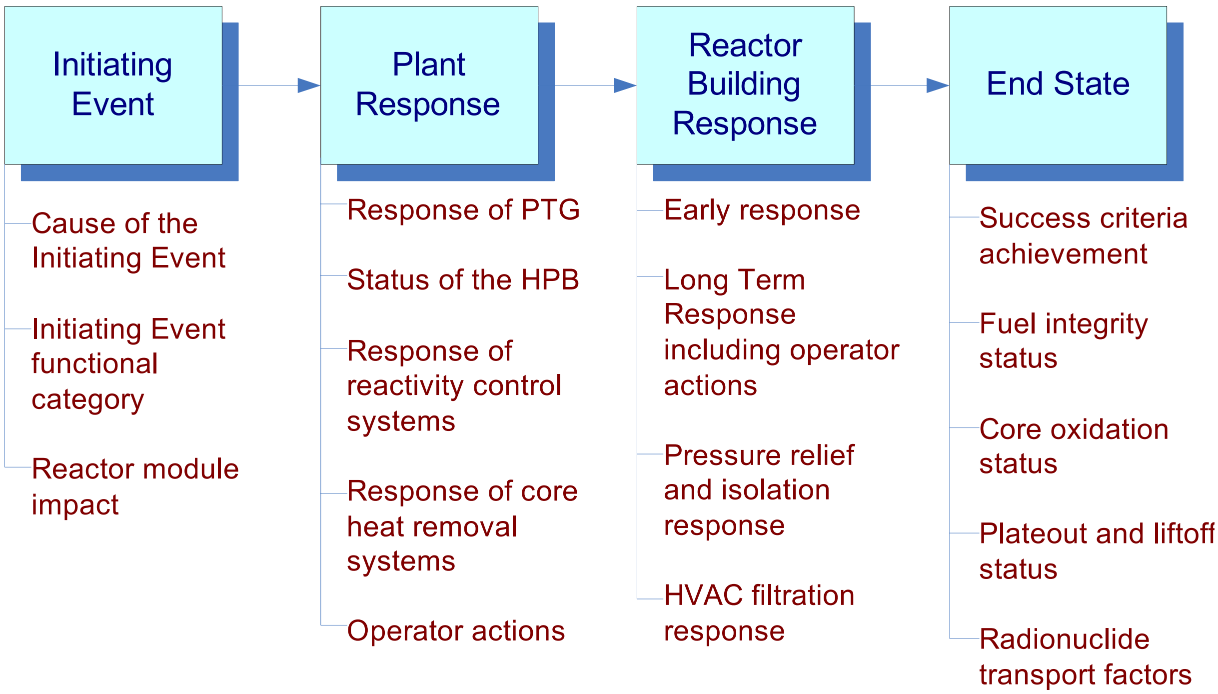


Figure 3: Event Sequence Modelling Framework for PBMR Plant

A further grouping of event sequences will be made in terms of the characteristics of any radionuclide release. PBMR release categories are listed in Table 4. The PBMR DCA PRA will include variations of these to account for design changes that may be made prior to the DCA, to account for the integrated risk of the multi-module design and other factors that may influence the mechanistic source term. The principle that will be used to make the final definition of release categories will be to define the necessary and sufficient set to capture the risk profile.

Table 4: Example PBMR PRA Release Categories

Code	Definition
RC-I	No release with an intact HPB
RC-II-F	Filtered release of all or part of circulating activity only
RC-II-U	Unfiltered release of all or part of circulating activity only
RC-III-F	Delayed filtered release from failed fuel with MPS pump-down
RC-III-U	Delayed unfiltered release from failed fuel with MPS pump-down
RC-IV-F	Delayed filtered release from failed fuel without MPS pump-down
RC-IV-U	Delayed unfiltered release from failed fuel without MPS pump-down
RC-V-F	Delayed filtered fuel release with oxidation from air ingress and lift-off of plated out radionuclides
RC-V-U	Delayed unfiltered fuel release with oxidation from air ingress and lift-off of plated out radionuclides
RC-VI	Loss of core, reactor vessel, or HPB structural integrity with unfiltered release

3.6.4 Event Sequence Families

In selecting LBEs, event sequence families are used to group together two or more event sequences when the sequences have a common initiating event, safety function response and end state. The process of defining event sequence families applies the following considerations:

- The guiding principle is to aggregate event sequences to the maximum extent possible while preserving the functional impacts of the initiating event, safety function responses, and end state. Note that for a multi-module plant the end state includes the number of reactor modules involved in the event sequence.
- The safety function responses are delineated to a necessary and sufficient degree to identify unique challenges to each SSC that performs a given safety function along the event sequence. Event sequences with similar but not identical safety function responses are not combined when such combination would mask the definition of unique challenges to the SSC that perform safety functions.
- In many cases for a single module plant, there may be only one event sequence in the family.
- For a multi-module plant, event sequence families are used to combine event sequences that involve individual reactor modules independently into a single family of single reactor module event sequences. In this case the individual event sequences are associated with a specific reactor module, and the family groups them together for the entire multi-module plant.
- Each event tree initiating event and safety function response has a corresponding fault tree that delineates the event causes and SSC failure modes that contribute to the frequencies and probabilities of these events. Hence each event sequence is already a family of event sequences when the information in the fault trees is taken into account.

- The frequency of the LBE defined by the accident family is the linear sum of the individual event sequence frequencies. The frequency units are events per plant-year. This provides a common frequency basis to compare and combine different types of sequences involving different numbers of reactor modules, and different plant operating states.

A common situation that yields event sequence families is when two or more initiating events that belong to the same functional category are quantified through the event trees separately, but follow the same event tree model and end states. For example, for MPS heat exchanger tube breaks, separate initiating events could be defined for pre-cooler and intercooler tube breaks, but since the event sequences follow the same event tree logic and result in the same end states, they are aggregated into a family. Alternatively, one MPS Heat Exchanger tube break initiating event could be defined, in which case the event sequence families already contain the individual event sequences for both pre-cooler and intercooler tube breaks. Another common situation includes the case when event sequence families are used to combine event sequences in a multi-module PRA.

Without the use of event sequence families, the level of detail in the definition of the initiating event categories and decisions to balance the level of detail between the event trees and fault trees may inadvertently impact the classification of an individual event sequence as an AOO, DBE, or BDBE. By aggregating the sequences into the event sequence families, the decisions made in structuring of the event sequence model do not impact the LBE classification. A fuller discussion of how event sequence families are used to define LBEs is provided in the LBE Selection paper.

3.7 EXAMPLE PBMR EVENT SEQUENCE MODEL

The purpose of this paragraph is to describe the following:

- The approach that is used to develop the event sequence models for the PBMR PRA through the use of an example for one selected functional initiating event category.
- The PBMR design features relevant to a selected event.
- The safety functions and SSC available to mitigate the consequence of the event.
- The elements of the modelling of event sequences specific to the PBMR.

The PBMR design assumptions and PRA models used to develop these examples are based on an early design of the PBMR and a corresponding PRA model that is the same as that used to support the Exelon PBMR Pre-licensing activities that were documented in [19]. This is done to provide examples with public domain references and to provide consistency with the LBE Selection paper which uses examples from the same design and PRA. There will be significant differences in both the design and the PRA models that are associated with the DCA. However, these examples serve the intended purpose of describing how the PBMR safety design philosophy is reflected in the PRA and help to bring out potential issues that can be addressed during the pre-licensing phase. The DCA PRA model is not currently available to provide such examples.

3.7.1 MPS Heat Exchanger Tube Break Event Sequence Diagram

To illustrate the approach that is used to model and document the event sequence development for the PBMR PRA, consider the example of the MPS heat exchanger tube break. The PBMR MPS has two physically identical gas to water heat exchangers, referred to as the Pre-cooler and the Intercooler. During normal plant operation at full power, helium flows from the outlet of the Recuperator to the inlet of the Pre-cooler at 142 °C at 2.9 MPa and exits the Pre-cooler at 24 °C at about the same pressure. The Pre-cooler outlet flow enters the low-pressure compressor and then the Intercooler at 111 °C at 5.1 MPa, and leaves the Intercooler at 23 °C at about the same pressure. The water sides of these heat exchangers are cooled by two independent closed water circuits within the Active Cooling System (ACS). These water circuits are low temperature and low pressure systems and provide cooling water to each heat exchanger at 18 °C at < 1 MPa. The water leaves the Pre-cooler and Intercooler at 71 °C and 56 °C respectively. There is a minimum of 2.0 MPa pressure drop across the gas to water heat exchanger surfaces during all modes of power operation. Hence, if there is a heat exchanger tube break, helium gas will flow into the water system. The capability of the Power Conversion Unit (PCU) as a core heat removal system is lost when this occurs, as the ACS is the only heat removal pathway for the PCU.

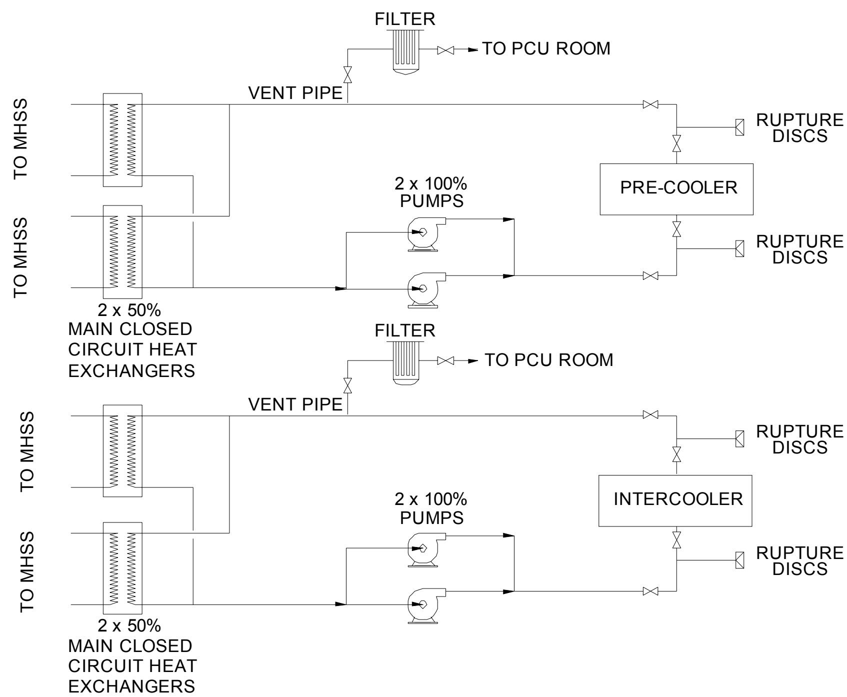


Figure 4: Schematic Diagram of Active Cooling System

As illustrated in Figure 4, the ACS is equipped with control valves and rupture discs to prevent water side overpressure and minimize subsequent water ingress in the event of a heat exchanger break. The rupture discs are set to open at about slightly above 1 MPa. The safety design philosophy to protect the plant investment, maintain plant availability goals, and assure safe plant response is to shut down the plant, provide forced circulation cooling using the Core Conditioning System (CCS), isolate the MPS from the reactor vessel using the MPS maintenance valves, and continue in this mode until the MPS heat exchanger can be repaired, ACS rupture discs replaced, and the ACS refilled, and the plant can be returned to full power operation. The fuel handling system would be shut down and the fuel would remain inside the reactor vessel while these repairs are made. The event sequence diagram for this initiating event is shown in Figure 5. This diagram shows the major event sequences for this initiating event and describes some key plant conditions that are important to determine the ultimate end state.

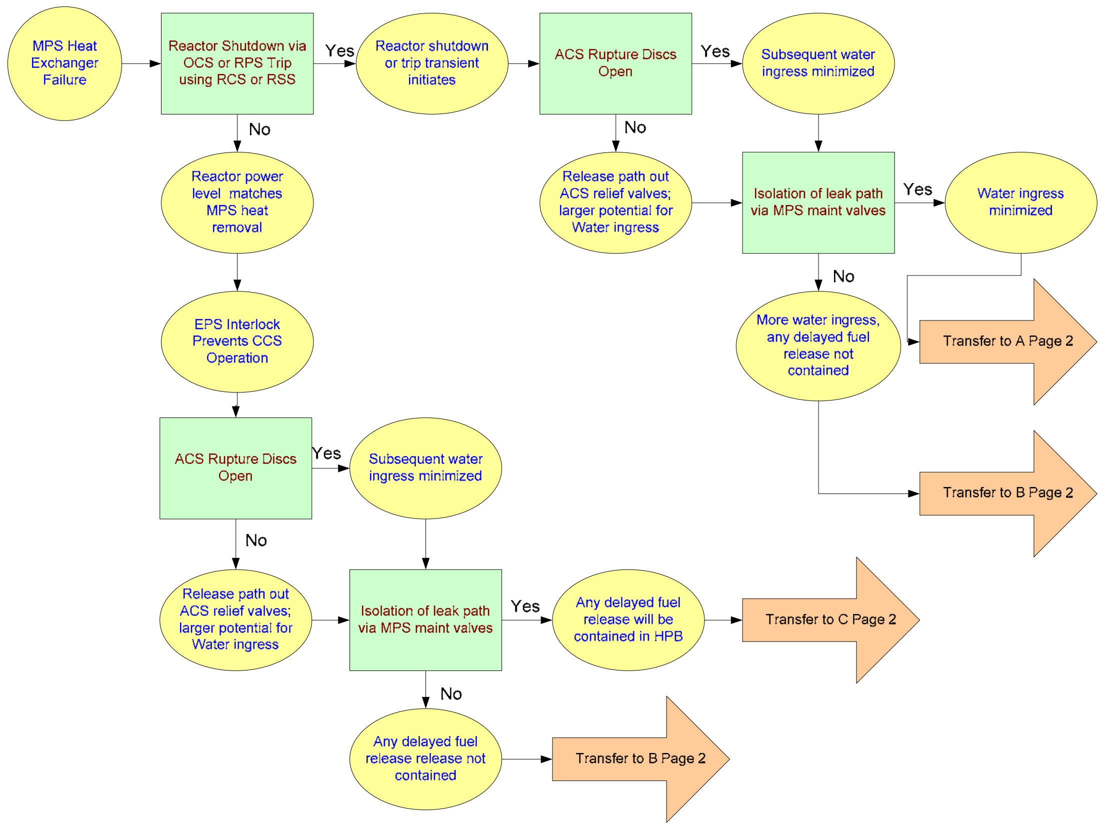


Figure 5: Event Sequence Diagram for MPS Heat Exchanger Break (Part 1 of 2)

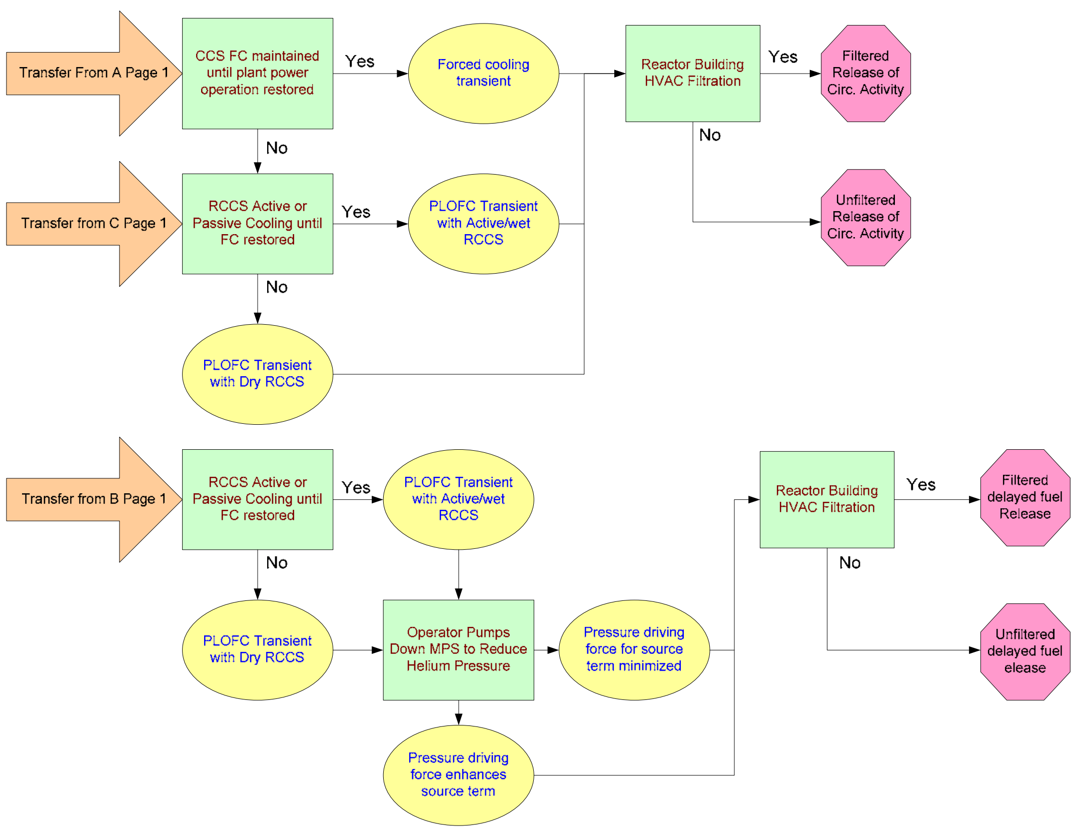


Figure 5: Event Sequence Diagram for MPS Heat Exchanger Break (Part 2 of 2)

The first key event in the Event Sequence Diagram (ESD) involves the expected plant shutdown via the Operational Plant Control System (OCS) which would shut down the reactor by a controlled insertion of the control rods. This is backed up by a Reactor Protection System (RPS) reactor trip of the control rods and operator actions to insert the control rods or the SAS of the RSS. There is physically very little difference in plant response – whether the reactor is shut down via the OCS, it automatically trips via the RPS, or if none of the reactivity control systems or operators respond – because the loss of the ACS as a consequence of the initiating event will lead to prompt negative reactivity feedback and the reactor will shut down via inherent and passive means. The ESD tracks the response of the ACS rupture discs. The disks are unlikely to fail but, should they fail, there is somewhat greater potential for water ingress to the MPS and a challenge to the chemical attack safety function.

The next key event is the action to isolate the MPS leak path which is normally done using a non-return valve that closes from the action of the CCS circulator to prevent bypass flows around the MPS circuit and eventually the closure of the maintenance valves via operator action which could occur when the MPS is sufficiently depressurized. If the isolation is successful, the CCS system can continue to be used to provide forced cooling and even if it fails to perform this function, any delayed fuel release will be contained within the reactor vessel with the environmental release path limited to leakage past these valves.

The development of mechanistic source terms for each sequence considers the following components of radioactive material inventory that could contribute to a potential source term:

- Circulating helium coolant radioactivity including elemental and dust-borne activity.
- Elemental and dust-borne radioactivity plated out on HPB surfaces.
- Radioactivity from uranium contamination outside fuel particles.
- Radioactivity in failed and defective fuel particles.
- Radioactivity in intact fuel particles.

An order of magnitude comparison of the inventories of one key radionuclide, ^{131}I in the PBMR Main Power System is shown in Table 5. ^{131}I has been determined to be an important radionuclide in many previous PRAs on HTGRs [28].

Table 5: Comparison of ^{131}I Inventories

Component of Inventory	^{131}I Curies
Circulating activity	<< 1
Plate-out on internal HPB surfaces	< 1
Uranium contamination outside coated particles	~100
Failed and defected coated particles	~500
Intact coated particles	$\sim 1 \times 10^7$

In developing the PBMR event sequence logic, the following rules are applied:

a. A breach of the HPB will permit the release of the circulating activity.
b. Lift-off of plated-out radionuclides requires large openings in the HPB that produce large shear force ratios during MPS blowdown; lift-off is not important for MPS heat exchanger tube breaks.

c. A delayed fuel release is possible only for sequences in which there is a loss of forced circulation cooling of the core; sequences with continued forced cooling via the MPS or CCS do not have a delayed fuel release. Some fraction of the 600 Ci of ^{131}I associated with failed or defective fuel particles, or uranium contamination outside the fuel particles, may be associated with a delayed fuel release.

d. The ^{131}I inventory associated with intact coated particles will remain inside the particles for all the event sequences considered in the design.

The delayed fuel release is associated with the slow release of part of the inventory in any failed or uranium contaminated fuel particles in regions of the core that experience an increasing temperature transient several days after the initiating event. This condition is satisfied only for small regions of the core, and only when there is a sustained loss of forced core cooling. As shown in Figure 6, peak core temperatures decrease with time for any pressurized or depressurized condition with continued forced circulation cooling via the CCS.

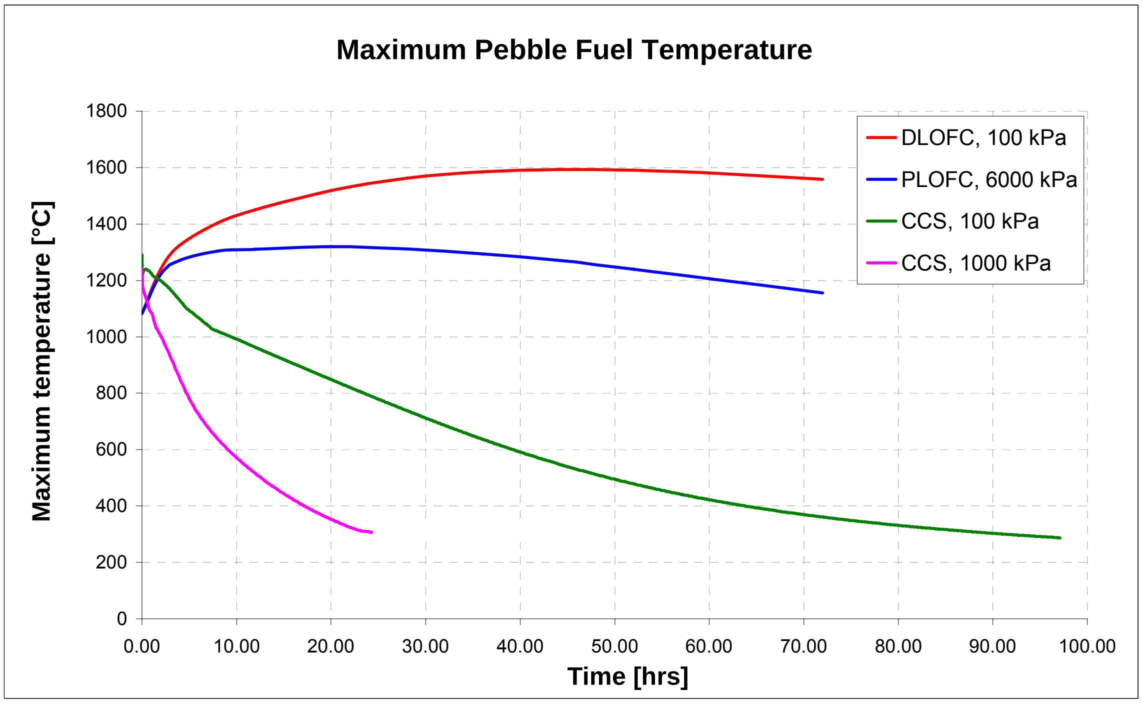


Figure 6: Peak Core Temperatures for Selected Pressurized and Depressurized Forced and Loss of Forced Cooling Transients

On sequences where there is no isolation of the MPS leak path, the mitigation strategy is to pump down the MPS helium inventory to reduce the MPS pressure and the pressure drop across the release pathway at the location of the HPB breach. If the pressure drop is reduced, the driving force for a source term involving a delayed fuel release is minimized.

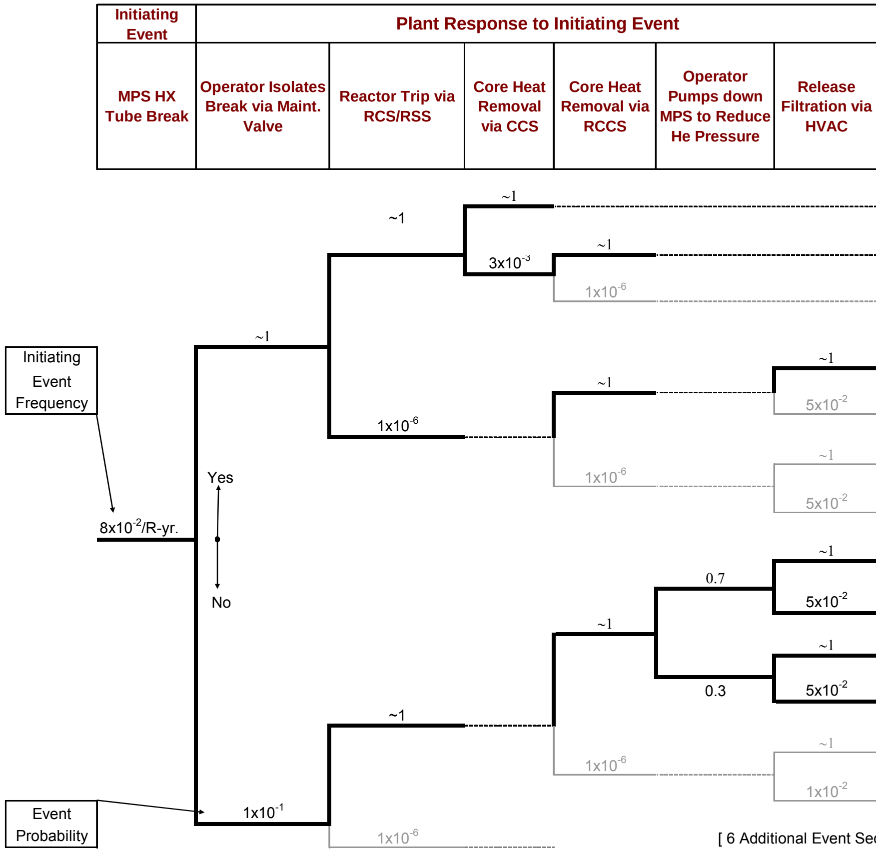
The final issue addressed in the scenario development is the response of the reactor building HVAC system which is designed to maintain a negative pressure in the PCU citadel where the MPS release path is located and pass any source term through an HVAC filtration system which would significantly reduce the mechanistic source term for filterable radionuclides such as ^{131}I. The scenarios in the event sequence diagram are organized in an abbreviated format with the note that any source term is dependent on the path followed through the entire diagram so that the condition of the core, the release path and source term mitigation factors can be properly combined.

The PBMR PRA to be submitted with the DCA will include ESDs such as these, deterministic plant transient analysis to describe the physical plant response for all key sequences, and mechanistic source terms for all risk significant sequences. The ESDs will be developed in somewhat greater detail than those presented here for illustration purposes only.

3.7.2 MPS Heat Exchanger Failure Event Tree Diagram

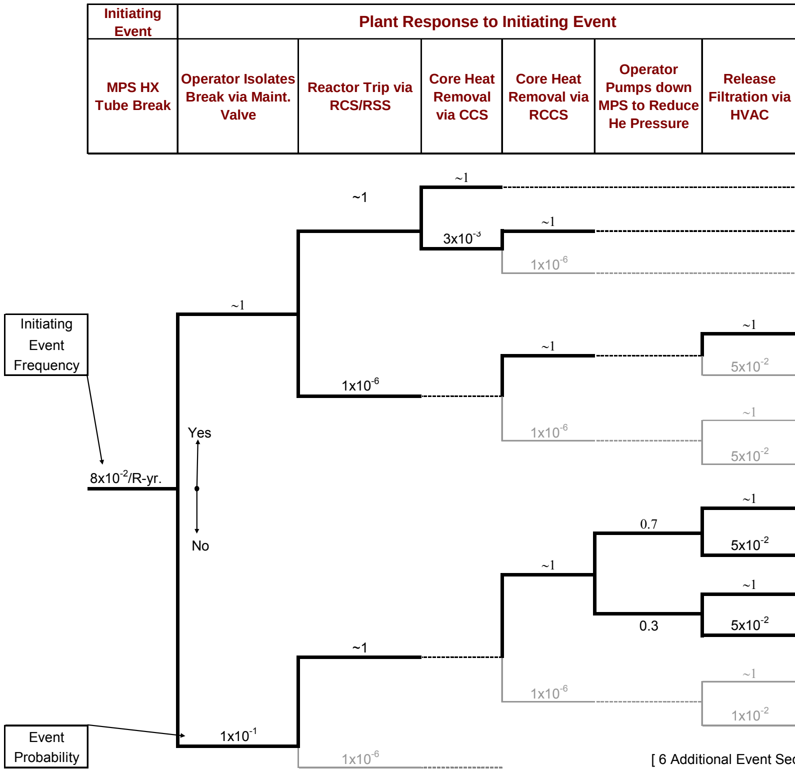
A simplified event tree diagram for this initiating event is illustrated in Figure 7. This diagram shows example initiating event frequencies and event probabilities that illustrate several aspects of how the PRA will be used to provide input to the selection of Licensing Basis Events (LBEs). Event sequences with frequencies less than 1 x 10^{-8} per plant-year are not developed in terms of a quantitative consequence analysis consistent with standard PRA practice and Regulatory Guide 1.200 [17]. In this diagram, a sequence-specific assessment of end state conditions and radiological consequence is performed. In this case, this is done qualitatively by indicating key factors that will determine the magnitude of the source term. This event tree is developed for the case of a single reactor module, and the event sequence frequencies per plant-year are the same as the frequencies per reactor-year. Those event sequences with frequencies above 1 x 10^{-2} per plant-year are classified as Anticipated Operational Occurrences (AOOs), those with frequencies between 1 x 10^{-4} per plant-year and 1 x 10^{-2} per plant-year are classified as Design Basis Events (DBEs), and those with frequencies less than 1 x 10^{-4} per plant-year as Beyond Design Basis Events (BDBEs). The significance of these event sequence classifications, the basis for the frequency ranges, and how they are used to define Licensing Basis Events (LBEs) is explained in the LBE Selection paper. This information is provided here to provide traceability between the PRA and the LBE selection process.

In the case of an eight-reactor module plant, the event tree development would be performed differently, as shown in Figure 8. Initiating events would occur on a single reactor module, because each module has its own MPS heat exchangers and own closed cooling water circuits in the ACS. The total frequencies of each sequence for eight modules would be a factor of eight higher than the former case, as these are reactor module independent events. Hence, some event sequences that are classified as DBEs or BDBEs for a single reactor plant might be classified as AOOs or DBEs, respectively when this factor of eight is taken into account. As this initiating event is reactor module independent, the magnitude of any mechanistic source term would be based on the inventories of a single reactor module. There are other initiating events such as loss of off-site power, seismic events, and other external events in which the mechanistic source term could involve events impacting two or more reactor modules. Hence both the frequencies and consequences of the event sequences could be influenced by the number of modules present. By expressing the event sequence frequencies on a per plant year basis, an integrated assessment of risk for the multi-module plant will be developed. This example ESD and event tree will of course be repeated for all the events within the scope of the PRA. The rest of the PRA addresses the models and data developed to quantify the event sequence frequencies, the mechanistic source terms, and radiological consequences and uncertainties.



Sequence No.	Event Sequence Frequency per R-yr.	Number of Reactor Modules Impacted	LBE Classification	End State: Release of Circulating Activity	Delayed Release from Fuel?	Delayed Fuel Release from MPS?	Pressure Drop Across Release Path	Release is filtered?
1	$8x10^{-2}$	1	AOO 5a	Yes	No	No	N/A	No
2	$2x10^{-4}$	1	DBE 5b	Yes	Yes	No	N/A	No
3	$<10^{-8}$	1	-					
4	$8x10^{-8}$	1	-	Yes	Yes	No	N/A	Yes
5	$<10^{-8}$	1						
6	$<10^{-8}$	1						
7	$<10^{-8}$	1						
8	$6x10^{-3}$	1	DBE 6a	Yes	Yes	Yes	No	Yes
9	$3x10^{-4}$	1	DBE 6b	Yes	Yes	Yes	No	No
10	$2x10^{-3}$	1	DBE 6c	Yes	Yes	Yes	Yes	Yes
11	$1x10^{-4}$	1	DBE 6d	Yes	Yes	Yes	Yes	No
12	$<10^{-8}$							
13	$<10^{-8}$							

Figure 7: Event Tree for MPS Heat Exchanger Break for Single Module



Sequence No.	Event Sequence Frequency per Reactor-year	Number of Reactor Modules Impacted	Event Sequence Frequency per 8-Module Plant-Year	LBE Classification	End State: Release of Circulating Activity	End State: Delayed Release from Fuel?	End State: Delayed Fuel Release from MPS?	End State: Pressure Drop Across Release Path	End State: Release is filtered?
1	$8x10^{-2}$	1	$6x10^{-1}$	AOO 5a	Yes	No	No	N/A	No
2	$2x10^{-4}$	1	$2x10^{-3}$	DBE 5b	Yes	Yes	No	N/A	No
3	$<10^{-8}$	1	$<10^{-8}$						
4	$8x10^{-8}$	1	$6x10^{-7}$	BDBE 5c	Yes	Yes	No	N/A	Yes
5	$<10^{-8}$	1	$<10^{-8}$						
6	$<10^{-8}$	1	$<10^{-8}$						
7	$<10^{-8}$	1	$<10^{-8}$						
8	$6x10^{-3}$	1	$5x10^{-2}$	~~DBE~~ AOO 6a	Yes	Yes	Yes	No	Yes
9	$3x10^{-4}$	1	$2x10^{-3}$	DBE 6b	Yes	Yes	Yes	No	No
10	$2x10^{-3}$	1	$2x10^{-2}$	~~DBE~~ AOO 6c	Yes	Yes	Yes	Yes	Yes
11	$1x10^{-4}$	1	$8x10^{-4}$	DBE 6d	Yes	Yes	Yes	Yes	No
12	$<10^{-8}$	1	$<10^{-8}$						
13	$<10^{-8}$	1	$<10^{-8}$						

19.2

LBE Classification Changed for Multi-module Design

[6 Additional Event Sequences with Frequencies < 10^{-8} per reactor year]

Figure 8: Event Tree for MPS Heat Exchanger Break for Eight-module Plant

3.8 PRA TREATMENT OF INHERENT AND PASSIVE SAFETY FEATURES

The PRA will be structured and performed in a manner that reflects the safety design philosophy of the PBMR. This is accomplished in the definition of the PBMR-specific safety functions and SSC to support those functions as described in the previous paragraphs and the development of success criteria that are derived from the properties of the inherent features as well as those of the SSC that are involved in the prevention and mitigation of accidents. The PBMR safety design approach places considerable emphasis on the use of inherent characteristics and passive design features to perform safety functions. An outline of these inherent and passive design features and the associated SSC correlated to the safety functions modelled in the PRA is provided in Table 3. This approach is reflected in the definition of the scope of the PRA, as well as the way in which the PRA models are defined and analysed. Some of the key elements of the inherent and passive safety features of the PBMR and how they are treated in the PRA are described in Table 6.

PRAs on currently licensed LWRs also address both inherent characteristics and passive design features. Examples of passive safety features in LWRs include the reactor coolant pressure boundary and containment building, natural circulation capability in the reactor coolant system which eliminates dependence on certain pumps under certain conditions, capability of removing heat by boiling off inventories of primary and secondary coolant without pumping fluid, negative temperature coefficients of reactivity, reactivity feedback from voiding in the coolant, gravity feed capabilities to make up lost coolant inventories under certain conditions, and many other examples of safety functions that rely at least in part on the performance of passive engineered safety systems.

The approach that is employed in PRAs for both types of reactors is fundamentally the same. The increased reliance on inherent characteristics and passive design features for the PBMR has the following types of impacts on the PRA:

- As is the case with LWR PRA, the PBMR PRA models and supporting assumptions are built on a technically sound foundation of deterministic and mechanistic models to predict the plant response to initiating events and event sequences and to develop the mechanistic source terms.
- The PBMR utilizes a full-scope PRA treatment of internal and external hazards, such as internal fires and seismic events, consistent with the current state-of-the-art in order to capture a comprehensive set of challenges to the inherent and passive safety features. Given the reduced reliance on active SSC to perform safety functions, it is reasonable to expect that safety function failures will be dominated by events and conditions that exceed the design basis envelope for passive SSC, and extreme external hazards represent one way that this can occur.

It is generally recognized that passive SSC tend to exhibit lower failure probabilities than active SSC. Lower failure probabilities also exhibit generally greater uncertainty. This means that while passive SSC are expected to have significantly lower failure probabilities, there are greater uncertainties in predicting the frequencies of passive SSC failures. Uncertainties in the estimation of both the event sequence frequencies and consequences will be addressed within the capabilities of the state-of-the-art and as defined in currently available PRA standards using standard PRA methods. Structured sensitivity analyses will also be applied where appropriate. The results of the uncertainty and sensitivity analysis will be taken into account in the selection of LBEs. The approach to selection of LBEs includes deterministic and conservative elements to make the selection robust in light of the uncertainties as discussed more fully in the LBE Selection paper.

Table 6: PRA Treatment of PBMR Inherent and Passive Features

PBMR Inherent and Passive Features	PBMR PRA Treatment
Fuel particle capabilities during normal and accident conditions	Failed fuel fraction from manufacture treated probabilistically based on manufacturing, operating and heat-up test data; failed fuel during burn-up and accident modelled probabilistically as part of fuel failure model in source term analysis; source term uncertainties quantified including those associated with fuel performance and other transport mechanisms.
Negative temperature coefficient of reactivity	Deterministic accident simulation models will treat this realistically; uncertainties in core reactivity and thermal response addressed as part of mechanistic source term and associated uncertainty analysis.
High thermal heat capacity (low-power density) of core and reflector	Deterministic accident simulation models will treat this realistically; uncertainties in core thermal response addressed as part of mechanistic source term and associated uncertainty analysis.
Passive core cooling capability	Event trees will define both active and passive modes of the core heat removal systems including the RCCS; seismic events and other external events will be defined that challenge and exceed the RCCS capability; fragilities assessed; potential for leakage from RCCS standpipes due to common cause failure mechanisms to be addressed; uncertainties in passive heat transfer to be assessed as part of source term uncertainty analysis.
Core, vessel and associated support structures	Full seismic and external event analysis will be performed that consider events that challenge or exceed design basis capabilities of all SSC modelled in the PRA and fragilities will be assessed for these hazards.
Coolant pressure boundary integrity and capability to limit air ingress	LWR piping experience and pipe reliability models are applied for expected PBMR applicable pipe damage mechanisms to quantify HPB failure initiating event frequencies; leak before break approaches being factored into the design will be accounted for in these estimates. Event trees will cover a range of HPB failure sizes and failure modes; consequence analysis will include a quantification of the impacts of any air ingress and oxidation reactions as part of the core thermal transient analysis, and will be addressed as part of the mechanistic source term and associated uncertainty analysis.
Reactor building structure including pressure relief features	Event trees will develop a spectrum of sequences that define a range of challenges and responses of blow-out panels. The uncertainty analysis will treat the response of the reactor building pressure relief features probabilistically if needed.

3.9 DEVELOPMENT OF A PRA DATA BASE FOR THE PBMR

The purpose of this paragraph is to discuss how the PRA database for the PBMR is being developed and to address how limitations in the available PBMR service experience will be taken into account. The PRA database being referred to is that for establishing the initiating event frequencies, component failure rates, unavailability terms, common cause parameters, and other parameters within the domain of ‘PRA data’. This adequacy of the technical basis for the data is addressed first by analysing the PRA data requirements in terms of the different types of data parameters and the evidence that is available to quantify the data parameters and has been used to develop the PBMR PRA database. The second approach to address the adequacy of the data is to review the role that service experience has played in the development of LWR PRA technology over the past three decades to gain some insights into how the service experience impacts the estimation of rare event frequencies. This review develops some insights into some limitations in the use of service experience in the quantification of PRA data parameters. The review shows that with the exception of relatively high frequency events, even large amounts of service experience do not eliminate the large uncertainties in the prediction of rare events.

3.9.1 Types of PRA Data Required for the PBMR PRA

The PRA data parameters provided in the PBMR PRA database will include the following data categories:

a. Failure rates and unavailabilities for active components unique to Gas-cooled Reactors (GCRs) (e.g. gas blowers, gas-to-gas and gas-to-water heat exchangers, GCR control rod drives, gas system valves).
b. Failure rates and maintenance terms for active components common to LWRs (e.g. pumps and valves in water systems, water-to-water heat exchangers, diesel generators, breakers, Control and Instrumentation (C&I) components).
c. Common cause failure parameters for a limited set of redundant components, mostly in common cause groups of components common to LWRs.
d. Initiating event frequencies for HPB passive component failure modes (e.g. pipes, pressure vessels, weldments and pressure relief valves).
e. Initiating event frequencies for power conversion system failure modes (turbo-compressors, gas-turbine generators).
f. Initiating event frequencies for the same internal and external plant hazards as found in full-scope LWR PRAs (fires, floods, seismic events, transportation accidents).

The associated component failure modes for the parameters in the first category above are not normally risk significant due to the reliance on inherent characteristics and passive SSC to perform safety functions. These data parameters appear in the PRA primarily with respect to initiating event frequencies. In addition, the number of unique components in this category is rather small due to the increased reliance on passive safety systems. Data from GCR experience in the United Kingdom (e.g. control rods, gas blowers, and gas valves) is available to support some of these component failure rates. These parameters will be addressed via engineering judgement and will be assigned larger uncertainty bands.

For Category 2 parameters, the existing LWR PRA databases and supporting service experience are directly applicable.

For Category 3 parameters, there are only a few systems that employ redundancy, and those are for the most part Category 2 components which are supported by the LWR failure rate and common cause parameter database.

For Category 4, the PBMR design uses piping and pressure vessel components designed with materials and design codes common to LWRs. Although there are internal components that are exposed to helium temperatures as high as 900 °C, the external pressure boundary is kept at temperatures somewhat lower than that for LWR pressure boundary components during normal plant operation.

Some pipe damage mechanisms such as welding defects, thermal fatigue and vibration fatigue are applicable to modular HTGRs, while others such as internal corrosion mechanisms are minimized due to the high purity requirements for the circulating helium which is necessary to protect the fuel and graphite components from oxidation phenomena. Estimates of failure rates and rupture frequencies for PBMR HPB components have been derived from LWR service experience, taking into account the applicable failure mechanisms [29]. The applicability of LWR piping system failure mechanisms to modular HTGR HPB components as assessed in [29] is shown in Table 7.

As an example, a set of failure rates as a function of rupture severity for welds in carbon steel pipe on the PBMR pressure boundary is presented in Figure 9. For this PBMR HPB component, the service data from BWR main steam system piping was found to be applicable to the design codes and service conditions for the PBMR. In assessing the conditional probabilities of different rupture sizes that are used in these estimates, use was made of the results of a recent expert elicitation that was performed to update estimates of loss of coolant accident initiating event frequencies for LWRs [30]. Although there are expected to be positive benefits of the use of a highly pure and chemically inert coolant in the PBMR, the initial PBMR PRAs will use piping failure rates and rupture frequencies for the HPB that are not much different than for comparable LWR piping.

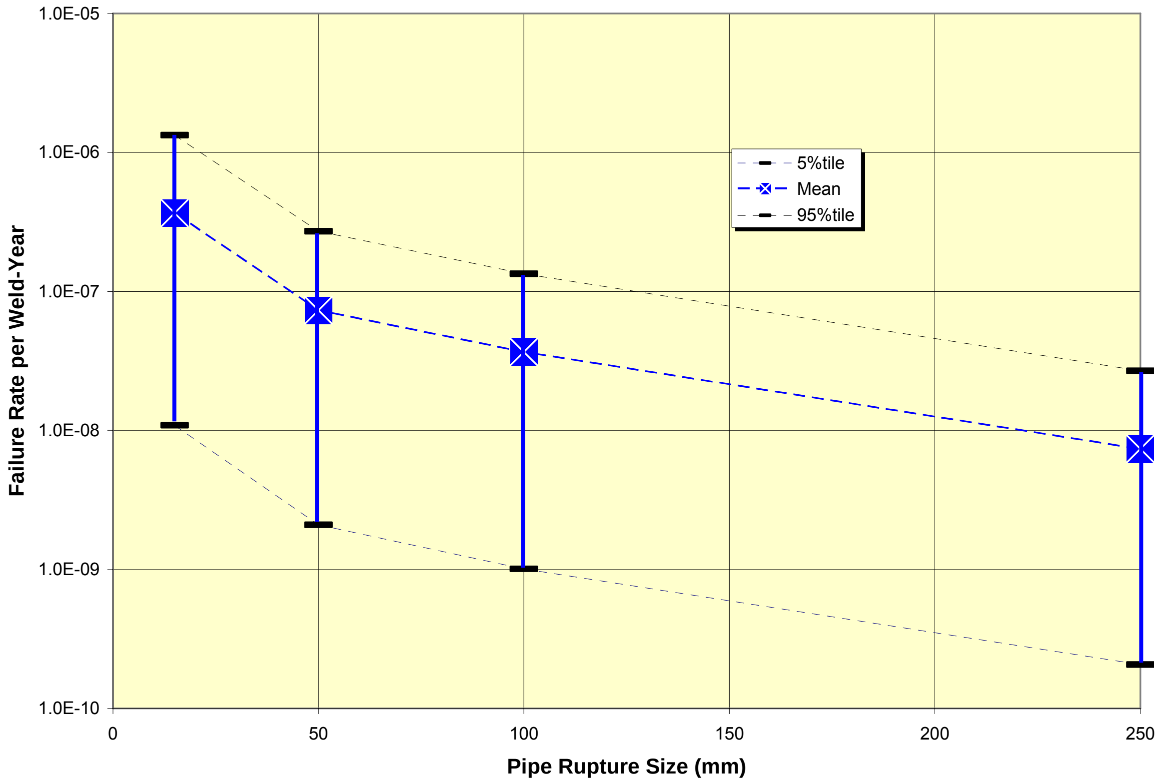


Figure 9: Failure Rate vs Rupture Size for 250 mm Carbon Steel Pipe Weld on PBMR Helium Pressure Boundary

For Category 5 events, there are significant differences in the design of the power conversion systems among the various modular HTGR designs, as well as differences with standard LWR steam cycle -designs. This and the lack of service experience with direct gas turbine cycle designs will be taken into account in the treatment of uncertainty for this category of data parameters. There are some relevant data from fossil-fuelled gas turbine and combined cycle plants that can be used to support this category. Meanwhile, the combination of expert opinion and conservative assumptions will be relied upon in the quantification of data parameters for this category.

For Category 6, there is essentially no difference between LWRs and HTGRs with respect to initiating event frequencies in these categories. However, there are significant differences that need to be taken into account when assessing the impacts of fires, floods, and seismic events on the operability of unique HTGR SSC. These unique impacts are reflected in the treatment of safety functions, success criteria, and deterministic analyses to simulate the plant response as discussed previously.

Table 7: Applicability of LWR Pipe Damage Mechanisms to PBMR HPB

Damage Mechanism		Applicability to PBMR
Symbol	**Description**	
CF	Corrosion-fatigue	Internal corrosion not applicable; external corrosion applicable
COR	Corrosion	Internal corrosion not applicable; external corrosion applicable
D&C	Design and Construction	Applicable
E/C	Erosion-corrosion	Not applicable
E-C	Erosion-Cavitation	Not applicable
HE	Human Error	Applicable
SC	Stress Corrosion Cracking	Internal SC not applicable; external SC applicable
TF	Thermal Fatigue	Thermal cycling applicable; thermal stratification and striping - not applicable
UNR	Unreported/Unknown	Applicable
VF	Vibration Fatigue	Applicable
WH	Water Hammer	Not applicable
EXT	External Loads (e.g. seismic events, missiles)	Applicable

For some selected components and events, service experience with GCRs has proven to be useful in the estimation of component failure rates. For many of the components level data that is needed for the PBMR PRA, existing generic data from LWRs is available and will be used. Engineering judgment will be relied upon for unique PBMR components for which there are little or no service data available to derive failure rates from. In such areas, the PBMR PRA will emphasize the quantification of uncertainties in both the accident frequencies and consequences; the limited service data will be reflected in larger uncertainties than cases where there is more data available to support the estimates. In some cases, where the PRA results are insensitive to data assumptions, conservative assumptions may be used in lieu of full uncertainty treatment. The larger uncertainties expected for PBMR-specific and unique components and events are not expected to be a problem, as the PRA results and insights are not expected to sensitive to these parameters.

3.9.2 Relationship between Uncertainty and Amount of Service Experience

Current PRAs on LWRs are supported by several thousand years of operating experience with LWRs to support the estimation of the data parameters that are modelled in the PRA. As discussed in paragraph 3.9.1, this service experience is applicable to many of the PBMR data parameters that need to be quantified. Only one category of data parameters, Category 1, must be quantified without the benefit of this LWR or other substantial service experience. The amount of PBMR and relevant HTGR and GCR experience that will be available to support the PBMR PRA is comparatively small, yet it is comparable to the amount of experience that existed with LWRs when the WASH-1400 study [31] was performed in the mid-1970s. It is useful to review the development of the database that was used in that landmark study, the results and insights of which are still used today. This review is aimed at establishing the relationship between the amount of service experience and PRA data uncertainties.

The quantification of the event sequence frequencies in WASH-1400 was supported mostly by generic industry data from non-nuclear power plants. The initiating event frequencies for Loss of Coolant Accident (LOCA) frequencies, a critical data parameter for an LWR PRA, were based on data collected from gas-pipelines and from non-nuclear fossil fueled steam cycle power plants. Engineering judgement was applied to estimate the improvement in performance in the piping systems to be expected by applying the ASME nuclear codes and special treatments. Despite the lack of a firm statistical basis, these estimates of the LOCA frequencies were used for more than 20 years, during which time most of the current risk informed applications on LWRs were completed. Only recently have improved estimates of LOCA frequencies been developed that have materially benefited from the accumulation of many years of LWR service experience, such as those in [30] and [32]. These improved estimates serve to validate such WASH-1400 insights derived from generic data treatment that large LOCAs are less risk significant than small LOCAs.

Despite the lack of service data in the LOCA frequency estimates, the data was adequate to support the applications of those PRAs and the development of technically sound conclusions. That is, the profound insight from WASH-1400 that small LOCAs are more risk significant than large LOCAs has not been revised through the accumulation of extensive service experience.

The limitations of service experience in supporting estimates of rare event frequencies can be seen with the following example. Most LWR PRAs since WASH-1400 have used an estimate of large break LOCA frequencies of the order of 10^{-4} per reactor-year. Uncertainty in this estimate is typically characterized using a lognormal distribution with a range factor of 10, and the above estimate taken as the mean value of this distribution. By 2005 there have been nearly 10 000 reactor operating years of service experience with current generation LWRs worldwide with no observed medium or large break LOCAs, and no small break LOCAs that challenged a full set of LOCA mitigation functions such as Emergency Core Cooling System recirculation switchover. In Figure 10 we show the results of a Bayes' update of the above generic estimate of large break LOCA frequency as a function of the number of years without an event that is accumulated since the service data is collected. As seen in this figure, the updated estimates do not change appreciably either in terms of the mean value or the uncertainty percentiles until the experience exceeds approximately 1 000 reactor years. This helps to explain why the LOCA frequency estimates originally developed in WASH-1400 have not changed very much in spite of the fact that nearly 10 000 reactor years of LWR service experience has been incorporated into the most recent estimates. The current estimates are still highly dependent on expert opinion.

When such large amounts of service experience are accumulated, it is often not possible to use it all to try and justify a low failure probability, because all the service experience may not be homogenous. Design changes and changes in operational conditions and inspection practices may lead to a need to exclude much of the service experience for statistical analysis purposes. This tends to limit the capability to aggregate large plant population exposure data sets, even when a large number of reactor years of experience are available. For the PBMR as well as the LWR the event sequences that are expected to be risk significant will exhibit large uncertainties despite how much service experience is available.

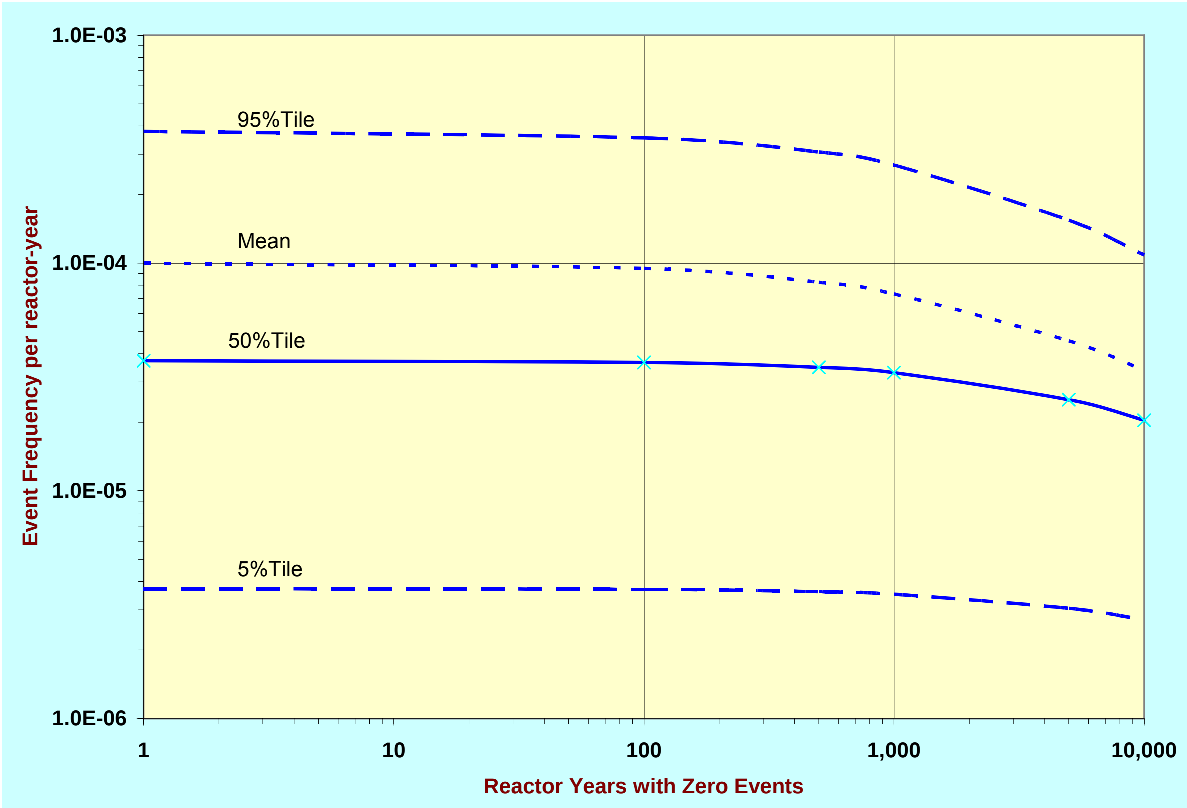


Figure 10: Bayes' Update of Large LOCA Frequency Estimate

3.10 RELATIONSHIP OF PRAs FOR THE DCA AND DPP

The DCA PRA will make use of the models, analyses, and data developed to support the DPP PRA where applicable. This is expected to add to the credibility of the assumptions that will be needed to perform a site-independent design stage PRA for the DCA. The DCA PRA will account for differences in design including the design of a multi-module plant with shared systems, and will account for the integrated risk using the approach described earlier in this chapter.

As with any PRA that is performed in support of design certification under 10 CFR Part 52, including those for evolutionary and passive ALWRs, the DCA PRA will be performed for a design that is sufficiently detailed to meet the requirements of Part 52, but many of the details of the design for an operating reactor, such as the interface with a site-specific heat sink, will not be defined. In addition, the design certification is expected to be applicable to a range of possible US sites, and hence a site-specific assessment of external hazards cannot be included in the PRA, as would be required by the available PRA standards. This design stage aspect of the PRA will be addressed in the following manner:

- Assumptions that are made in lieu of as-built and as-operated design details will be documented so that the impact of these assumptions on the selection of LBE can be explicitly evaluated.
- Where appropriate, the models used in PRA will be based on design details and models established for the DPP. Differences in the design details between those assumed in the DCA PRA and those for the DPP in order to increase the range of applicable sites will be identified and documented.

- The seismic PRA will be performed using an assumed seismic hazard curve that is selected to bound a range of US sites, uniform hazard spectra, and general fragilities for plant SSC.
- Internal fires and internal floods will be given as full of a PRA treatment as can be accomplished for a design stage PRA.
- External plant hazards will be addressed using a conservative bounding treatment that is sufficient to account for a range of US sites for the purpose of identifying external hazards for the LBEs.
- The seismic, internal fire and internal flooding PRAs will be performed using 3D computer models of the plant layout and general arrangement for the purpose of addressing plant walk-down requirements that can be met during the design stage.
- Uncertainties due to the lack of design details and site-specific characteristics will be taken into account in the assignment of uncertainties using expert opinion. Sources of uncertainty that cannot be quantified will be identified for consideration in the selection of LBEs.

3.11 KEY INTERFACES WITH DETERMINISTIC SAFETY ANALYSIS

The PBMR PRA has been developed and will continue to be developed in conjunction with a technically sound deterministic engineering basis that supports the safety case. The uses of the PBMR PRA results as part of the licensing basis are regarded as examples of a ‘risk-informed’ as opposed to a ‘risk-based’ process. As the PRA cannot be separated from the underlying deterministic bases, the interfaces with the deterministic analysis are discussed here. This is important to highlight the role of the PRA in a risk-informed design process that is integrated with the traditional deterministic approaches. The role of deterministic safety analyses has not been diminished by this use of the PRA, but rather has been strengthened as outlined in the following paragraphs.

In the development of the PBMR, probabilistic and deterministic safety assessments have developed concurrently and in an integrated fashion. The safety design philosophy itself is rooted in deterministic safety assessment principles. Key design parameters such as the core size and shape, power density, reactor cavity configuration, the fuel particle design and manufacturing specification are based on the principle of preventing core damage and large releases from the fuel using deterministic methods and means. Important aspects of the PBMR safety design philosophy, such as the importance placed on inherent and passive means to implement safety functions, are based on sound deterministic design principles. The design calculations that were made to establish these parameters were based on a conservative deterministic engineering analysis for a set of enveloping events and boundary conditions, and in accordance with the defense-in-depth philosophy.

The systematic selection of initiating events was performed for the PRA and the deterministic analysis will be performed in an integral fashion. The applicable knowledge that is available to support the selection of possible initiating events for both the probabilistic and deterministic safety analysis is applied for this purpose. This knowledge base is systematically developed by application of Failure Modes and Effects Analysis, Hazard and Operability (HAZOP) investigations, and by reviews of list of events that had been considered for other PBMRs, HTGRs, GCRs and LWRs. The need for a systematic, comprehensive, and reproducible set of initiating events is viewed to be fundamental to both the probabilistic as well as the traditional deterministic approaches to the selection of LBEs.

The development of a prediction of a sequence of events that could occur in the PBMR in response to the initiating events is fundamentally a deterministic process. Such deterministic knowledge is essential to the task often used in developing ESDs and event trees for the PRA as it is for listing the sequence of events for Chapter 15 of the DCA. Deterministic analyses must be applied to determine the plant response to initiating events and the event sequences resulting from success/failure combinations of SSC defined in the event sequence models. Deterministic models must also be applied to determine the success criteria for SSC along each event sequence, and for determining the ends states and mechanistic source terms. The sequences and conditions that are derived by the traditional deterministic rules for selecting DBAs are imbedded in the PRA, as well as the sequences and conditions that are excluded from the traditional DBAs. So the net effect of the PRA approach is to bring in a more complete enumeration of event sequences into consideration as LBEs. Either approach must address the plant responses to the event sequences that are permitted into the analysis by the application of verified and validated deterministic models to predict the plant response to initiating events. The PRA approach to selecting LBEs is regarded as robust because it yields a more complete set of scenarios to consider compared with the deterministic rules such as the limitation imposed by the single failure criterion.

The PRA also requires the use of deterministic models to determine the criteria for successfully terminating the event sequences and for determining the end states. Development of mechanistic source terms is also an area where the deterministic computer models play an important role.

One area where the roles of probabilistic and deterministic analyses may be contrasted is in the treatment of uncertainty. Traditional approaches to safety analysis have approached uncertainty with such concepts as safety margins, defense-in-depth, and conservative assumptions in the safety analyses. In PRA, sources of uncertainty are exposed in the context of quantifying the risks of events, including those within and outside the design basis envelope. Uncertainties are not introduced by the PRA nor are they introduced by the deterministic safety analysis, but rather are properties of our state of knowledge as to how the plant responds during these rare events. Judgments in the face of uncertainty will need to be made independent of the safety analysis approach being used**.** These different approaches to the treatment of uncertainty will be applied to the PBMR in a complementary fashion. Sources of uncertainty that are identified in the course of quantifying the risks of the event sequences in the PRA will be identified for proper deterministic treatment in the selection of LBEs and the formulation of regulatory design criteria for the SSC that perform required safety functions.

It is expected that sources of uncertainty will be identified in the course of performing both the probabilistic and deterministic safety analysis for the PBMR, and that these uncertainties will be addressed by making appropriate judgements to apply safety margins and conservative assumptions, and deterministic requirements. The use of PRAs provides a uniform framework for assessing uncertainties, applying conservatisms and evaluating margins, defense-in-depth and the value of additional mitigating features. The most effective way to identify these sources of uncertainty is to subject the plant to a state-of-the-art PRA supported by a technically sound deterministic safety analysis. The roles of deterministic and probabilistic approaches as elements of this risk-informed licensing process for the PBMR is more fully explained as part of the LBE Selection paper.

As is explained more fully in the LBE Selection and SSC Safety Classification papers, once the LBEs are selected based on input from the PRA, the safety classification of SSC is performed using a deterministic method. This safety classification is then subjected to a rigorous and conservative deterministic safety analysis to demonstrate that the safety classified SSC are sufficient to ensure that the dose criteria for the DBA are met with sufficient safety margin. This provides a balanced blend of deterministic and probabilistic approaches. PRA is not performed in place of deterministic analysis, but rather provides a risk-informed logic structure for deciding which deterministic analyses to perform. The deterministic safety analysis is thus integrated with the PRA process. This integration affords the opportunity to incorporate the event sequences that are the most risk significant into the design basis.

3.12 PRA GUIDANCE, STANDARDS AND APPROACH TO TECHNICAL ADEQUACY

The purpose of this paragraph is to describe the approach to using available guides and standards to assure the technical adequacy of the PBMR PRA. Comparisons to LWR PRAs are made to establish the similarities and differences between the PRAs for these reactor types in order to assist the NRC in planning for the DCA PRA review.

The applications envisioned for the PBMR PRA require the ultimate resolution of reactor accident consequences in a manner similar to that supported by an LWR Level 3 PRA. However, the means of dividing the LWR PRA into a Level 1-2-3 structure is not applied to the PBMR for the reasons explained earlier in this paper.

By design, PBMR has no damage states analogous to LWR core damage state in which a large fraction of the fission product inventory is released from the fuel as is postulated to occur in more severe core damage events that are modelled in typical LWR PRAs.

The PBMR PRA is structured to expose the appropriate damage states for the reactor in a manner that determines the level of risk of events and helps define the limiting LBEs that are appropriate for the PBMR.

LERF is not a useful risk metric for HTGR PRAs. The PBMR PRA has not yielded a credible scenario that would release large enough quantities of fission products, nor early enough, to approach the definition of a large early release.

The following aspects of current NRC guidance on PRAs will be modified to support their application to the PBMR PRA:

- The current quality initiatives focus on PRAs that are used to calculate CDF and LERF. However, the core damage end state has a definition that is specific to LWRs and is not directly applicable to the PBMR, which is subject to fundamentally different types of damage states. By replacing CDF and LERF with the frequencies of specific event families, the PBMR can utilize the vast majority of the technical requirements in the PRA standards in a straightforward manner. PBMR specific event families will be defined in a manner analogous to accidents for LWRs by specifying appropriate combinations of initiating events and successful and unsuccessful operation of SSC and operator actions to fulfil plant-specific safety functions.
- As noted in the previous section, it is neither appropriate nor necessary to fit the PBMR PRA into the mould of the Level 1-2-3 framework. Instead, an integrated PRA that develops sequences from initiating events all the way to source terms and consequences will be developed.
- Also, as noted in the previous section, it is not necessary to perform a completely different set of PRA models for full-power versus low-power and shutdown. The PBMR lends itself to

an integrated treatment of accident sequences that covers all operating and shutdown modes.

- Unlike most current LWR PRA applications that are adequately supported by a Level 1 PRA, the initial PBMR application to select LBEs will require quantification of site boundary dose consequences to be able to apply the LBE selection criteria for frequency and dose. To the extent supported by the anticipated relatively small magnitude source terms, simplified models to estimate off-site health effects will be applied, that are sufficient to apply the frequency dose criteria that are used to select LBEs and to demonstrate that NRC safety goal QHOs are met. Simplified bounding calculations using site boundary dose analyses will suffice for this purpose, as was the case for the MHTGR [28]. Evidence to support this conclusion will be included in the PRA.
- The calculation of site boundary doses is supported by a mechanistic accident progression and source term analysis that includes a quantification of uncertainties. The technical basis for the PBMR mechanistic source terms will be included as part of the DCA. There are no available PRA standards for mechanistic source terms, and most of the available guidance for establishing their adequacy is based on LWR-specific source terms and associated phenomena. Hence the criteria for acceptance of the PBMR mechanistic source terms need to be established, and this issue is identified as an issue to address during the preapplication phase of PBMR design certification.
- In view of the applications envisioned for the PBMR PRA, a full-scope treatment of internal events and internal and external hazards is anticipated including events both within and beyond the design basis for these hazards. The recently developed industry/NRC work on internal fire PRA methodology in NUREG/CR-6850 [33] will be used as guidance to support this task. For the PRA that is included with the DCA, some generic treatments of internal and external hazards will be necessary, as explained more fully below.
- Assumptions that are made to support the PRA development for the DCA will be identified and documented. These assumptions will be evaluated and taken into account in the uncertainty and sensitivity analysis. As explained more fully in the LBE Selection paper, deterministic approaches will be applied in the selection of LBEs to make the selection of LBEs rather insensitive to the expected differences in PRA results due to differences between the DCA design and site-specific designs.
- The PRA to support the DCA will be based on a site-independent design that is sufficiently detailed to meet the requirements of 10 CFR Part 52. As with other designs that have been certified according to Part 52, it will be necessary to perform a generic treatment (i.e. generic to a range of sites but PBMR design specific) of certain external hazards such events as seismic events and aircraft crash, the risk characteristics of which are known to exhibit significant site to site variability. Such generic treatment in the DCA is intended to bound the risk impacts of external hazards for a range of specific sites and to be sufficient to identify an appropriate set of LBEs associated with such hazards. When a COL application is made based on the certified design, a review will be performed that compares the site-specific hazards against the generic treatment of such hazards in the DCA. This review will also include a comparison of the site-specific design details against the design assumptions made in the DCA. The focus of these reviews is to confirm that the selection of LBEs made from the DCA PRA results has not been impacted. The logic of this process is viewed as being consistent with the treatment of site-specific PRAs in SECY 93-087 and the associated SRM as discussed in Chapter 2.

With these adjustments, the applicable LWR PRA standards and peer review process will be used as an approach to ensure adequate PRA quality for the DCA. An evaluation of the applicability of these standards to each PBMR PRA element is provided in Table 8. Note that the ASME standard proposes three Capability Categories to address PRA requirements for different applications. The process described in Section 3 of the ASME PRA Standard will be followed to determine the appropriate Capability Level to apply for each requirement. It is expected that Capability Level II will be appropriate for many requirements, but where appropriate, Capability Level III requirements will be applied. Due to the fact that the DCA PRA will be performed at the design stage and will be site independent, some requirements for Capability Category I will not be met at that time, but this is no different than for other design certifications under 10 CFR Part 52. As part of the PRA documentation, the interpretations of the PRA standards that were assumed in the PRA development and the assumed PRA Capability Category and its basis will be documented.

Table 8: Comparison of PBMR PRA Technical Elements and Applicable PRA Standards

Technical Elements	Applicable PRA Standards	Comments
1. Definition of Plant Operating States	ANS Draft Low Power and Shutdown (LPSD) PRA standard for low-power and shutdown states. [25]	One set of plant operating states will be defined to cover all envisioned plant operating and shutdown states for the Main Power System radionuclides. Appropriate states will be defined for other sources of radioactivity.
2. Initiating Events Analysis	ASME PRA Standard [23] and ANS LPSD PRA standard - Initiating Events Analysis	PBMR and LWR PRAs are essentially equivalent for this element; needs to be addressed for all modelled operating states.
3. Accident Sequence Definition	ASME PRA Standard and ANS LPSD PRA standard - Accident Sequence Analysis	Event trees will be developed for the response of SSC in the performance of plant-specific safety functions for each source; core damage end states would be replaced by PBMR-specific end states.
4. Success Criteria Development	ASME PRA Standard - Success Criteria and Supporting Engineering Analysis	Success criteria will be specific to PBMR inherent characteristics and end states.
5. Thermal and Fluid Flow Analysis	ASME PRA Standard - Success Criteria and Supporting Engineering Analysis	The physical and chemical processes that govern core reactivity, fuel temperatures, and all factors influencing radionuclide transport are fundamentally specific to HTGRs and will be addressed using deterministic computer models.
6. Systems Analysis	ASME PRA Standard - Systems Analysis	PBMR and LWR PRAs are essentially equivalent for this element except that the PBMR has fewer systems to analyse, the safety functions are different, and there is greater reliance on passive design principles.
7. Human Reliability Analysis	ASME PRA Standard - Human Reliability Analysis	PBMR and LWR PRAs are essentially equivalent for this element except that there are fewer actions to consider and the scenarios tend to progress slowly.

Technical Elements	Applicable PRA Standards	Comments
8. Data Analysis	ASME PRA Standard - Data Analysis	PBMR and LWR PRAs are essentially equivalent for this element. As there is less relevant operating experience for some SSC, the treatment of uncertainty will be an important issue.
9. Internal Flooding Analysis	ASME PRA Standard - Internal Flooding Analysis	PBMR and LWR PRAs are essentially equivalent for this element except that there are fewer flooding sources and consequences of flooding will be in the context of a PBMR specific event sequence model.
10. Internal Fires Analysis	ANS Draft Standard for Internal Fires Analysis [26]; NUREG/CR-6850 [33]	PBMR and LWR PRAs are essentially equivalent for this element except that there are fewer cables and the consequences of fires will be assessed in the context of a PBMR specific event sequence model.
11. Seismic Analysis	ANS PRA Standard External Events Analysis [34]	PBMR and LWR PRAs are essentially equivalent for this element; the consequences of seismic failures will be assessed in the context of a PBMR-specific event sequence model. For the DCA, a site-independent seismic PRA similar to that performed for the DOE MHTGR that bounds the seismic hazard risk impacts for a large set of US sites will be performed.
12. Other External Events Analysis	ANS PRA Standard External Events Analysis	PBMR and LWR PRAs are essentially equivalent for this element; the consequences of external events will be assessed in the context of a PBMR-specific event sequence model. For the DCA, an assessment of external hazards, which bounds the external hazard risk impacts for a large set of US sites and uses insights from the site-specific DPP PRA, will be performed.
13. Event Sequence Quantification	ASME PRA Standard Quantification	LWR separation of accident sequences into Level 1-2-3 structure is not appropriate for PBMR; scope of accident sequences include doses at the site boundary, and risk importance measures to be developed and analysed for each major plant-specific accident category.
14. Mechanistic Source Term Analysis	No corresponding PRA standard	This task is functionally similar to the mechanistic source terms analysis in an LWR Level 2 PRA; mechanistic source term phenomena and barrier design are specific to PBMR safety design approach.
15. Accident Consequence Analysis	No corresponding PRA standard	This task is similar to the consequence analysis in an LWR PRA which is not currently covered in LWR PRA standards, except that only site boundary doses are needed; detailed modelling of emergency response, health effects and land contamination may not be needed due to low source terms, as bounding estimates for these should suffice.

Technical Elements	Applicable PRA Standards	Comments
16. Risk Integration and Interpretation	No corresponding PRA standard	This task is needed to integrate the frequency and consequence information into a frequency-consequence format. Risk importance metrics will be normalized to PBMR-specific accident families and end states.
17. Peer Review	ASME PRA Standard and ANS LPSD PRA standard Requirements for peer review; NEI PRA Peer Review Process [34]	A peer review will performed for each major PRA phase that supports the various stages of the design process; periodic updates during plant operation will be performed as needed to support the certification decision.

4. ISSUES FOR PREAPPLICATION RESOLUTION

The issues addressed in this paper are framed in terms of the following questions about the PRA that will be performed to support the PBMR DCA. The PBMR position on the appropriate response to these questions is discussed in detail in Chapter 3 and summarized following the listing of each question.

1. **Are the intended uses of the PBMR PRA, including the selection of LBEs, input to the safety classification of SSC and the development of special treatment requirements for SSC, acceptable?**

PBMR Response: These intended uses are generally consistent with those described in Issue #4 of SECY 2003-0047. There is no reason in principle why these intended uses should not be acceptable. They are generally consistent with NRC's technology neutral licensing framework initiative, and more extensive use is made of risk-informed design and licensing.

2. **Given the intended uses described in Issue 1, what is the appropriate scope of the PRA?**

PBMR Response: A full-scope, all modes, and all hazards PRA as described in Chapter 3 will be performed for certification of the PBMR. Risk insights derived from such an approach will produce a balanced perspective for selecting LBEs and safety classification of SSC.

3. **How will the PBMR safety design approach such as the inherent and passive PBMR safety characteristics be taken into account?**

PBMR Response: As outlined in the event sequence framework in Chapter 3, the PBMR PRA is characterized by the systematic identification of PBMR-specific initiating events, definition and analysis of PBMR safety functions, delineation of all SSC that provide either required or supportive safety functions, technically sound deterministic engineering analyses, and mechanistic source terms.

4. **How will the PRA be used to understand the adequacy of the defense-in-depth approach in the design, construction and operation of a PBMR plant?**

PBMR Response: Information from the PRA will be used to identify the roles of each PBMR SSC responsible for preventing and mitigating each LBE that makes a significant contribution to the risk of a release of radioactive material [35]. Prevention will be analysed in terms of how the reliability characteristics of SSC contribute to the frequency of initiating events and the probability of failure of SSC that fail to perform their functions in response to an initiating event. Mitigation will be analysed in terms of the retention fractions of the radionuclide source inventories within each of the barriers to release including the fuel particle, graphite matrix, plate-out surfaces, HPB, and reactor building SSC. The roles that redundancy, diversity, independence and safety margins play in managing the risks of event sequences will be examined in this investigation. Deterministic approaches that are taken to address uncertainties will also be identified. This approach to using the information from the PRA to address defense-in-depth will be explained more fully in the Defense-in-Depth paper

5. **Are risk metrics such as core damage frequency appropriate for the PBMR? What risk metrics will be applied in the PRA?**

PBMR Response: Core damage frequency is not an appropriate metric because the core damage state as defined for LWRs is precluded by the PBMR safety design approach by deterministic means. The available definitions of core damage, such as those referred to in the ASME PRA standard, refer to LWR properties such as liquid levels in the reactor vessel, LWR fuel temperatures in relation to oxidation properties of zircalloy, and potential for large releases from the fuel, none of which apply to the PBMR. PBMR-specific accident families and release categories will be used as a basis to define PBMR-specific risk metrics that relate directly to the safety design approach of the PBMR and are expressed in terms of frequency of off-site radiological consequences.

6. How will the PRA use and interface with deterministic safety analyses?

PBMR Response: The PBMR safety design approach is deeply rooted in deterministic engineering principles. The PBMR PRA will be developed on a foundation of technically sound deterministic engineering analyses in a fashion that is functionally similar to the situation in LWR PRAs. The areas in which deterministic analyses will play a role include the definition of PBMR safety functions and success criteria, the prediction of the plant response to initiating events, and the development of mechanistic source terms. The deterministic and probabilistic analyses will be done in a coordinated and integrated manner. Once LBEs and safety classifications of SSC have been established, the licensing approach will include conservative deterministic safety analysis of DBEs to demonstrate that the selection of safety-related SSC is sufficient, and in this respect similar to that found in Chapter 15 of the DCA for current LWRs.

7. How will the limited PBMR service data be addressed in the development of PRA data?

PBMR Response: A technically sound database for the PRA will be developed by:

- Identifying SSC that are the same or similar to SSC and events in LWRs.
- Utilizing PRA data developed for these items.
- Identifying other SSC and events such as the pressure boundary components that are made of the same materials and use the same design codes as for LWRs, and applying the corresponding service data after considering the applicable failure mechanisms.
- Identifying SSC and events that are similar to those with applicable HTGR or GCR service experience and using that information.
- Identifying SSC and events that are unique to the PBMR and carefully applying expert judgement.

Uncertainties will be assessed and included in the PRA data.

8. How will source terms for event sequence be developed? How are uncertainties addressed?

PBMR Response: PBMR will include mechanistic source terms and a treatment of uncertainty in the development of these mechanistic source terms as part of the DCA PRA submittal. These source terms will account for the reliability of the fuel manufacture process, and for fuel performance during normal plant operation and burn-up, and during transient and accident conditions. Also reflected in the mechanistic source terms are the core reactivity behaviour, diffusion and oxidation phenomena, heat transport phenomena, fluid flow phenomena, and all relevant radionuclide transport phenomena. Uncertainties in the estimation of these source terms will be addressed.

9. How will modular reactor aspects of the design be accounted for in the PRA?

PBMR Response: The PRA will be developed for a multi-module design to be provided with the DCA. Event sequences involving single reactor and multiple reactor source terms will be explicitly developed. Event sequence frequencies will be calculated on a per multi-module plant year basis. The capabilities to support a fully integrated risk assessment will be available pending the outcome of ongoing policy discussions among the NRC commissioners, staff and ACRS on the integrated risk issue.

10. What is the approach to ensure the adequacy of the PRA quality for the intended applications?

PBMR Response: An approach to using existing LWR PRA standards and guidance as described in Chapter 3 and an independent peer review will be utilized to help ensure adequacy of the PRA for the use in PBMR certification. The DCA will advise the NRC how the existing standards were interpreted for application to the PBMR PRA. Certain requirements in the PRA standards and guidance that require knowledge of the as-built and as-operated design details and accumulation of service experience will not be met until a design certified PBMR is actually built. This issue is no different, however, than for any PRA in support of a DCA under 10 CFR Part 52.

11. What is the approach in the PRA for treating uncertainties associated with as-procured, as-built, site-specific and as-operated information that is not available when the DCA is submitted? How will these uncertainties be accounted for in the selection of a stable set of LBEs for operating plants?

PBMR Response: The PRA that is submitted for the DCA will be adequate to support the selection of LBEs for future operating PBMRs. The PRA itself is based on a deterministic foundation which is expected to be fundamental to future operating plants. The PRA uncertainties may be larger than for an existing operating plant, to account for the fact that equipment has not yet been procured, the plant has not yet been built, the site-specific features are not known, and the plant is not yet operating.

These uncertainties will be reflected in relatively large error bounds in the PRA results. Even given these large error bands, there will be sufficient margins between the PRA results and the frequency-dose criteria that will be used to select the LBEs. Additionally, the LBE selection process also has deterministic elements to make the final decisions on LBEs rather insensitive to numerical changes in PRA results.

As a result of the conservative treatment of uncertainties, the use of deterministic elements in the PRA and the LBE selection process, and the large margins between the PRA results and the frequency dose criteria for selecting the LBEs, there is confidence that the these design and site assumptions will not impact the selection LBEs. A design review is expected to be required during the COL stage that compares the site and design against the site and design assumptions in the DCA PRA to confirm that the selection of LBEs is appropriate.

12. What preapplication activities can be defined to set the stage for a successful PRA review for the DCA?

PBMR Response: It is expected that preapplication activities that are needed to set the stage for a successful PRA review will be identified as a result of a review of this paper and in-depth discussions in subsequent workshops. Examples of the preapplication activities that should be considered in these discussions include the following:

a. PBMR submittal and NRC review of analyses that have been performed to estimate HPB failure rates and rupture frequencies for pressure boundary failure rates used in the PRA based on LWR piping service experience and expert elicitation.

b. PBMR submittal and NRC review of a paper on the development of mechanistic source terms for the PBMR.

5. PREAPPLICATION OUTCOME OBJECTIVES

The objective of this paper and the follow-up workshops and paper revisions is to obtain NRC agreement on the list of issues for the use of PRA to support PBMR licensing as well as agreement on the approach to solving these issues. Specifically, PBMR would like the NRC to agree with the following statements, or provide an alternative set of statements that they agree with.

1. The scope of the PBMR PRA outlined in this paper (and discussed further in the LBE Selection, SSC Classification, and Defense-in-Depth papers) is appropriate for the intended uses of the PRA in the design certification of the PBMR. These uses include input to the selection of LBEs, input to the safety classification of SSC, and derivation of regulatory design criteria and special treatment requirements for SSC.
2. The PBMR PRA framework is a reasonable approach to capture the unique and specific elements of the PBMR safety design approach, and to delineate the elements of the PBMR PRA that are common to PRAs for LWRs.
3. The PBMR approaches to initiating event selection, event sequence development, end state definition, and risk metrics are appropriate to support the intended uses of the PRA and to account for the PBMR safety design approach.
4. The approach to the treatment of inherent characteristics and passive SSC is reasonable and consistent with the current state-of-the-art of PRA.
5. The approach to the use of deterministic engineering analyses to provide the technical basis for predicting the plant response to initiating events and event sequences, success criteria, and mechanistic source terms yields an appropriate blend of deterministic and probabilistic approaches to support the PBMR design certification.
6. The approach to the development of a PRA database as outlined in this paper, including the use of applicable data from LWRs, use of expert opinion, and treatment of uncertainty is a reasonable approach for the PBMR PRA.
7. The objectives for the development of a mechanistic source term provide a reasonable approach for this PRA application.
8. The approach for the PRA treatment of single and multiple reactor accidents in a multi-module design is sufficient to support certification of the basic single module of the PBMR for multi-module configurations.
9. The approach to using current guides and standards for LWR PRA quality and independent peer review, taking into account the differences due to the PBMR's safety design approach, is an acceptable approach to determining the adequacy of the PBMR PRA for its intended uses outlined above.
10. The PRA that is submitted for the DCA should be adequate to support the selection of LBEs for future operating PBMRs. The DCA PRA will account for uncertainties associated with as-procured, as-built, site-specific, and as-operated information in a conservative and bounding manner to provide assurance that the LBEs derived from the DCA PRA will be appropriate for as-built and as-operated plants.

It is requested that the NRC take the following steps:

Step 1 NRC review the paper for agreement on the list of issues and the PBMR approach proposed.

Step 2 The holding of a workshop on the issues identified in the paper and a discussion of the approach that is proposed for resolution.

Step 3 NRC issuance of preliminary comments and requests for additional information to clarify points not understood or adequately developed in the paper.

Step 4 PBMR preparation of a revised paper which identifies any Request for Additional Information (RAI) that can be addressed in the near term as well as requested information that will be included with the DCA PRA submittal. This will include a plan for preapplication activities that are agreed upon in the workshops as being necessary for a successful DCA PRA review.

Step 5 NRC issuance of an evaluation report on its findings related to the PRA and its intended use along with inputs to the DCA format and content guide that PBMR will use for the DCA.

6. REFERENCES

[1] Regulatory Guide 1.174, 'An Approach for Using Probabilistic Risk Assessment in Risk-informed Decisions on Plant Specific Changes to the Current Licensing Basis,' U.S. Nuclear Regulatory Commission, Revision 1, November 2002.

[2] 'Report on The President's Commission on the Accident at Three Mile Island', October 1979.

[3] Mitchell Rogovin, et al, 'Three Mile Island – A Report to the Commissioners and the Public', July 1980.

[4] U.S. Nuclear Regulatory Commission, '10 CFR Part 52--Early Site Permits; Standard Design Certifications; And Combined Licenses For Nuclear Power Plants' Published by Office of the Federal Register National Archives and Records Administration, January 1, 2004.

[5] PBMR (Pty) Ltd., 'U.S. Design Certification – Licensing Basis Event (LBE) Selection for the Pebble Bed Modular Reactor (PBMR)', June 2006.

[6] U.S. Nuclear Regulatory Commission, 'Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement,' Federal Register, Vol. 60, No. 158, pg. 42622-42629, August 16, 1995.

[7] U.S. Nuclear Regulatory Commission, 10 CFR Part 50, Appendix A. General Design Criteria for Nuclear Power Plants [63 FR 1897, January 13, 1998].

[8] SECY 2003-0047, 'Policy Issues Related to Licensing Non-Light Water Reactor Designs', U.S. Nuclear Regulatory Commission, March 28, 2003.

[9] SRM 2003-0047, 'Staff Requirements Memorandum for SECY 03-0047 – Policy Issues Related to Licensing Non-Light Water Reactor Designs', U.S. Nuclear Regulatory Commission, June 26, 2003.

[10] U.S. Nuclear Regulatory Commission, 'Draft Preapplication Safety Evaluation Report for the Modular High Temperature Gas-Cooled Reactor', NUREG-1338, March 1989.

[11] ACRS Letter, Subject: Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements, ACRSR-1509, February 19, 1993.

[12] SECY-93-087, 'Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs', U.S. Nuclear Regulatory Commission, April 2, 1993.

[13] SRM-93-087, 'Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs', U.S. Nuclear Regulatory Commission, July 21, 1993.

[14] U.S. Nuclear Regulatory Commission, 'Regulation of Advanced Nuclear Power Plants; Statement of Policy,' Federal Register, Vol. 51, No. 130, pg. 24643-14648, July 8, 1986.

[15] U.S. Nuclear Regulatory Commission, 'Safety Goals for the Operations of Nuclear Power Plants; Policy Statement', Federal Register, Vol. 51, No. 149, pp.28044-28049, August 4, 1986 (republished with corrections, Vol. 51, No. 160, pg. 30028-30023, August 21, 1986).

[16] Standard Review Plan 19.1, 'Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities', NUREG-0800, U.S. Nuclear Regulatory Commission, February 2004.

[17] Regulatory Guide 1.200, 'An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities', U.S. Nuclear Regulatory Commission (for trial use) February 2004.

[18] Regulatory Guide 1.183, 'Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors', U.S. Nuclear Regulatory Commission, July 2000.

[19] Exelon Generation Company Letter, Subject: Proposed Licensing Approach for the Pebble Bed Modular Reactor in the United States, January 31, 2002.

[20] U.S. Nuclear Regulatory Commission Letter, Subject: NRC Staff's Preliminary Findings Regarding Exelon Generation's (Exelon's) Proposed Licensing Approach For The Pebble Bed Modular Reactor (PBMR), March 26, 2002.

[21] U.S. Department of Energy, 'Preliminary Safety Information Document for the Standard MHTGR', DOE-HTGR-86-024, September 1988.

[22] NUREG/CR-2300, 'PRA Procedures Guide', U.S. Nuclear Regulatory Commission, January 1983.

[23] American Society of Mechanical Engineers, 'Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications', ASME-RA-Sb-2005.

[24] American Nuclear Society, 'External Events PRA Methodology' ANSI/ANS-58-21-2003, March 3, 2003.

[25] American Nuclear Society, 'Low-Power and Shutdown PRA Methodology Standard' Draft 6c, March 21, 2005.

[26] [American Nuclear Society, 'Fire PRA Methodology Standard', BSR/ANS 58.23 Draft Version of 10 March 2005.

[27] Fleming, K.N., et al, 'HTGR Accident Initiation and Progression Analysis Status Report – Phase II Risk Assessment', General Atomic Report No. GA-A15000, April 1978.

[28] U.S. Department of Energy, DOE-HTGR-86-011, 'Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor, Revision 5, April 1988.

[29] Bengt Lydell and Karl Fleming, 'PBMR Passive Component Reliability – Helium Pressure Boundary Components', Prepared by Technology Insights for PBMR (Pty) Ltd., PBMR Proprietary Data, Final Report, to be published in 2006.

[30] NUREG-1829 'Estimating Loss-of-Coolant-Accident (LOCA) Frequencies Through the Elicitation Process, U.S. Nuclear Regulatory Commission (Draft for Comment) June 2005.

[31] WASH-1400 (NUREG 75/014), 'Reactor Safety Study,' U.S. Nuclear Regulatory Commission, October 1975.

[32] NUREG/CR-5750, 'Rates of Initiating Events at U.S. Nuclear Power Plants', U.S. Nuclear Regulatory Commission, 1999.

[33] NUREG/CR-6850, 'EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities' (2 Volumes), U.S. Nuclear Regulatory Commission, September 2005.

[34] NEI-00-02, ‘Probabilistic Risk Assessment (PRA) Peer Review Process Guidance’, Revision 1, May 2006.

[35] Fleming, Karl N. and Fred A. Silady, ‘A Risk-informed Framework for Defense-in-depth for Advanced and Existing Reactors’, Reliability Engineering and System Safety, Elsevier Publishing Company, 78 (2002) pp. 205–225.