US DESIGN CERTIFICATION

LICENSING BASIS EVENT SELECTION FOR THE PEBBLE BED MODULAR REACTOR

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ABSTRACT

This preapplication paper outlines the relevant regulatory policy and guidance for the spectrum of Licensing Basis Events (LBEs) to be considered, defines licensing issues associated with LBE definition, describes the Pebble Bed Modular Reactor (PBMR) approach for the selection of the LBEs, and sets forth certain facts for review and discussion in order to facilitate an effective submittal leading to a PBMR design certification under 10 CFR Part 52.

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ABBREVIATIONS

This list contains the abbreviations used in this document.

Abbreviation or Acronym	Definition
ACRS	Advisory Committee on Reactor Safeguards
ALARA	As Low As Reasonably Achievable
ANPR	Advance Notice of Proposed Rulemaking
AOO	Anticipated Operational Occurrence
ATWS	Anticipated Transient Without Scram
BDBE	Beyond Design Basis Event
CAB	Controlled Area Boundary
CCS	Core Conditioning System
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
DBA	Design Basis Accident
DBE	Design Basis Event
DCA	Design Certification Application
DCD	Design Control Document
DPP	Demonstration Power Plant
EAB	Exclusion Area Boundary
EPA	Environmental Protection Agency
EPZ	Emergency Planning Zone
GDC	General Design Criteria
HPB	Helium Pressure Boundary
HTGR	High Temperature Gas-cooled Reactor
HX	Heat Exchanger
LBE	Licensing Basis Event
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LPZ	Low Population Zone
LWR	Light Water Reactor
MHTGR	Modular High Temperature Gas-cooled Reactor
MPS	Main Power System
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Commission report
PBMR	Pebble Bed Modular Reactor
PCU	Power Conversion Unit
PRA	Probabilistic Risk Assessment
QHO	Quantitative Health Objective
RAI	Request for Additional Information

Abbreviation or Acronym	Definition
RCCS	Reactor Cavity Cooling System
RCS	Reactivity Control System
RSS	Reserve Shutdown System
SBO	Station Blackout
SBS	Start-up Blower System
SRM	Staff Requirements Memorandum
SSC	Structures, Systems, and Components
TEDE	Total Effective Dose Equivalent
TLRC	Top Level Regulatory Criteria

1. INTRODUCTION

1.1 SCOPE AND PURPOSE

The PBMR Design Certification Application (DCA) will include a safety evaluation of a set of Licensing Basis Events (LBEs). As the term is used in this paper, LBEs are defined as the events that are considered in the licensing process and used to derive regulatory requirements for the PBMR design certification. LBEs include normal plant operation; events anticipated to occur in the life of the plant and off-normal events as required by 10 CFR Part 52, including infrequent Design Basis Events (DBEs); and rare events beyond the design basis. This paper outlines the relevant regulatory policy and guidance for the spectrum of events to be considered, defines licensing issues associated with LBE definition, describes the PBMR approach for the selection of the LBEs, and sets forth certain facts for review and discussion in order to facilitate an effective submittal leading to a PBMR design certification under 10 CFR Part 52.

The Nuclear Regulatory Commission's (NRC's) current regulations governing LBEs classified as Design Basis Accidents (DBAs), including the category of events referred to as 'Loss of Coolant Accidents' (LOCAs), were developed by the NRC for the licensing of Light Water Reactors (LWRs) and before the application of Probabilistic Risk Assessment (PRA) technology [1]. A large fraction of the General Design Criteria (GDC) refers to requirements to prevent and to mitigate LOCAs which are relevant to the safety design approach for LWRs. Before the GDC can be effectively reviewed for applicability to the PBMR, an appropriate set of PBMR-specific LBEs needs to be developed. The PBMR will apply state-of-the-art tools to select the LBEs as part of a risk-informed licensing approach, while maintaining the current NRC policies and guidance on the application of deterministic design criteria and the use of PRA [2], [3].

As the current set of licensing requirements was being developed, various rulemaking proceedings extended the LBEs for currently licensed reactors to include such events as Station Blackout (SBO), anticipated transients without scram, and internal fires. Additionally, for existing and advanced LWRs, the NRC developed guidance for the treatment of severe core damage accidents for plant conditions that are beyond those of the DBEs. While some of the LBEs for LWRs are applicable to the PBMR, many others are not. Moreover, LWR LBEs do not address all of the events and safety issues that are specific to the PBMR. This paper provides a structured, systematic, performance-based, and risk-informed process for selecting and analyzing LBEs for the PBMR.

PBMR's approach to the development of LBEs is risk-informed, and as such is based on both deterministic and probabilistic elements. The PBMR safety design approach is rooted in deterministic engineering principles. The PBMR PRA, which provides important probabilistic input to the selection of LBEs, is built on a foundation of deterministic principles and engineering evaluations which establish success criteria, predict the plant response to events, and establish the basis for the mechanistic source terms. Technical issues associated with the PBMR PRA are covered in a companion paper [4]. Once the LBEs have been defined and the safety classification of Structures, Systems, and Components (SSCs) has been made, the LBE selection and SSC safety classification decisions will be subjected to conservative deterministic safety evaluation to confirm that NRC's deterministic safety analysis requirements are still met.

While the PRA will provide important input to the selection of LBEs, the proposed licensing approach includes deterministic safety principles. The deterministic and probabilistic elements of the safety analysis will be integrated and developed concurrently and both play a role in final design and licensing.

The risk-informed licensing approach proposed for the PBMR includes the definition of Top Level Regulatory Criteria that provide frequency and dose limits for the LBEs, and in this respect determine ***what*** must be met for licensing approval. The selection of the LBEs answers the question of ***when*** the Top Level Regulatory Criteria (TLRC) are to be met. Other elements of the PBMR licensing approach answer the questions of ***how*** and ***how well*** the TLRC are to be met, as described more fully in the SSC Safety Classification and Defense-in-Depth papers.

The PBMR selection of LBEs provides a systematic, reproducible, and comprehensive enumeration of all the events that need to be considered in the certification of the PBMR.

This paper provides a summary of key issues associated with LBE selection and associated safety analyses to be identified and resolved prior to the submittal of the DCA.

1.2 STATEMENT OF THE ISSUES

The issues addressed in this paper are framed in terms of the following questions regarding the selection of LBEs to support the PBMR DCA:

1. What is an appropriate, systematic, and reproducible approach for selecting LBEs for the PBMR?
2. What is the appropriate blend of probabilistic and deterministic approaches for the selection and analysis of LBEs for the PBMR? What requirements must be applied to the PRA and supporting deterministic evaluations in order to support LBE selection and evaluation for the PBMR?
3. What categories of LBEs need to be considered?
4. What are the acceptable public consequences and analysis bases for each LBE category?
5. What is the frequency range for each LBE category?
6. At what frequency are events sufficiently low that they are not selected as LBEs?
7. What kinds of events and phenomena should the PRA include in the selection of a comprehensive set of event sequences?
8. How are the deterministic DBAs of Tier 2 of the Design Control Document (DCD) modeled, and analyzed?
9. How will uncertainties in SSC or operator performance be taken into account in the LBE selection?

The regulation and policy foundation for deriving this list of issues is developed in Section 2 of this paper. The PBMR approach to selection of LBEs is outlined in Section 3 and will be discussed at future NRC workshops. Section 4 examines how the PBMR approach meets the existing regulatory foundation in Section 2 and the guidance and precedents in this area.

1.3 SUMMARY OF PREAPPLICATION OUTCOME OBJECTIVES

The objective of this paper (and the follow-up workshops that are anticipated) is to get NRC agreement on the list of issues for the selection of LBEs to support PBMR certification as well as agreement on the approach to resolving these issues. Specifically, PBMR would like the NRC to agree with the following statements, or provide an alternative set of statements with which they agree:

1. The structured process for selecting LBEs using input from the PRA and supported by an integrated blend of deterministic and probabilistic elements is an acceptable approach for defining the PBMR LBEs.
2. The integrated blend of deterministic and probabilistic elements described in this paper establishes an appropriate performance-based and risk-informed approach for structuring the safety analyses that will be included in the DCA.
3. LBEs cover a comprehensive spectrum of events from normal operation to rare, off-normal events. Each LBE is defined as a family of individual event sequences where each family has a common initiating event, safety function response, and end state. This includes an appropriate definition of LBEs to support the integrated risk from a multi-module plant. There are three categories of LBEs:
    - Anticipated Operational Occurrences (AOOs) which encompass planned and anticipated events. The doses from AOOs are required to meet normal operation public dose requirements. AOOs are utilized to set operating limits for normal operation modes and states.
    - Design Basis Events (DBEs) encompass unplanned, off-normal events not expected in the plant's lifetime, but which might occur in the lifetimes of a fleet of plants. The doses from DBEs are required to meet accident public dose requirements. DBEs are the basis for the design, construction, and operation of the SSCs during accidents. Separate from the design certification, DBEs are also evaluated in developing emergency planning measures.
    - Beyond Design Basis Events (BDBEs) which are rare, off-normal events of lower frequency than DBEs. BDBEs are evaluated to ensure that they do not pose an unacceptable risk to the public. Separate from the design certification, BDBEs are also evaluated in developing emergency planning measures.

    The LBEs in all three categories will be evaluated individually to support the tasks of assessing the performance of SSCs with respect to safety functions in response to initiating events and collectively to demonstrate that the integrated risk of a multi-module plant design meets the NRC Safety Goals.
4. Acceptable limits on the event sequence consequences and the analysis basis for the LBE categories are as follows:
    - AOOs – 10 CFR Part 20: 100 mrem Total Effective Dose Equivalent (TEDE) mechanistically modeled and realistically calculated at the Controlled Area Boundary (CAB).
    - DBEs – 10 CFR §50.34: 25 rem TEDE mechanistically modeled and conservatively calculated at the Exclusion Area Boundary (EAB).

- BDBEs – NRC Safety Goal Quantitative Health Objectives (QHOs) mechanistically and realistically calculated at 1 mile (1.6 km) and 10 miles (16 km) from the plant.

5. The frequencies of LBEs are expressed in units of events per plant-year where a plant is defined as a collection of up to eight reactor modules having certain shared systems. The limits on the frequency ranges for the LBE categories are as follows:
   - AOOs – event sequences with mean frequencies greater than 10^{-2} per plant-year.
   - DBEs – event sequences with mean frequencies less than 10^{-2} per plant-year and greater than 10^{-4} per plant-year.
   - BDBEs - event sequences with mean frequencies less than 10^{-4} per plant-year and greater than 5×10^{-7} per plant-year.
6. The frequency below which events are not selected as LBEs is 5×10^{-7} per plant-year. The PRA examines events to 10^{-8} per plant-year to assure that there are none just below this *de minimus* frequency.
7. The kinds of events, failures, and natural phenomena that are evaluated include:
   - Multiple, dependent and common cause failures to the extent that these contribute to LBE frequencies.
   - Events affecting more than one reactor module.
   - Internal events and internal and external plant hazards that occur in all operating and shutdown modes and potentially challenge the capability to satisfactorily retain any source of radioactive material.
8. The deterministic DBAs for Chapter 15 of Tier 2 of the DCD are derived from the DBEs by assuming that only SSCs classified as safety-related are available to mitigate the consequences. The public consequences of deterministic DBAs are based on mechanistic source terms and are conservatively calculated. The upper bound consequence of each deterministic DBA must meet the 10 CFR §50.34 consequence limit at the EAB.
9. Uncertainty distributions are evaluated for the mean (statistical) frequency and the mean consequence for each LBE. The mean frequency is used to determine whether the event sequence family is an AOO, DBE, or BDBE. If the upper or lower bound (95%-tile or 5%-tile of the uncertainty distribution) on the LBE frequency straddles two or more regions, then the LBE is compared against the consequence criteria for each region. The mean, lower, and upper bound consequences are explicitly compared to the consequence criteria in all applicable LBE regions. The upper bound (95%-tile) for the DBE and deterministic DBA consequences must meet the 10 CFR §50.34 dose limit at the EAB.

1.4 RELATIONSHIP TO OTHER PREAPPLICATION FOCUS TOPICS/PAPERS

This paper on the selection of the LBEs is linked to the companion PRA paper as noted above. Subsequent papers on defense-in-depth and the SSC safety classification elements of the PBMR licensing approach are dependent on the selection of the LBEs.

Inherent in the PBMR safety design and licensing approach is the development and quantification of mechanistic source terms for the spectrum of LBEs. Papers on the fuel, the reactor unit materials, and verification and validation of the analytical models and computer codes are key inputs to the mechanistic source terms.

In addition, the papers on the fuel and reactor unit materials demonstrate for key PBMR SSCs the use of the LBE selection, the safety classification and the defense-in-depth elements of the PBMR licensing approach.

2. REGULATORY FOUNDATION

NRC regulations, policies, and guidance that are relevant to the definition of LBEs and their treatment are discussed in this section. These regulatory criteria are examined to investigate two aspects of the proposed risk-informed certification approach for the PBMR. The first is the process of enumerating and selecting the LBEs, and the second is the development of the TLRC which establishes limits on the frequencies and public radiological consequences used to classify and evaluate the LBEs.

2.1 REGULATORY FOUNDATION FOR THE SELECTION OF LICENSING BASIS EVENTS

2.1.1 NRC Regulations

NRC regulations and guidance for the design of currently licensed reactors divide LBEs into three categories: 1) normal operations including AOOs, 2) unplanned transients and Design Basis Accidents (DBAs), and 3) BDBEs, including severe accidents.

For normal operations including AOOs, the NRC regulations are for the most part generic and apply to the PBMR as discussed in Section 2.2.1.

However, for unplanned transients and accidents, the regulations that have evolved are LWR-specific. For LWRs, the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 indicate that Loss-of-Coolant Accidents (LOCAs) must be postulated in designing systems, and the GDC define the types of design considerations that apply to the design of SSCs that prevent or mitigate postulated accidents. For example, the GDC typically indicate that safety systems must be able to perform their design basis functions given a single active failure and a concurrent loss of offsite power.

NRC regulations do not define DBAs and transients in probabilistic terms. However, as provided in regulations such as GDC 35 and 10 CFR §50.46, LWRs must be designed with emergency core cooling systems to prevent any significant fuel damage (including fuel melting) in the event of a LOCA. Additionally, safety-related SSCs must be designed to withstand external events such as seismic events.

With limited exceptions, NRC's regulations do not have criteria to limit the consequences or frequency of BDBEs. The exceptions pertain to limited categories of events, such as Anticipated Transients Without Scram (ATWS) which are addressed in 10 CFR §50.62, Station Blackout (SBO) which is addressed in 10 CFR §50.63, and functions such as combustible gas control which is addressed in 10 CFR §50.44.

Several important observations pertain to the applicability of the above listed regulations to the PBMR. First, because the safety design philosophy of the PBMR does not depend on an inventory of the helium coolant and there is no SSC that is either needed or provided that performs the coolant inventory control functions of an emergency core cooling system as is the case with LWRs, LOCAs are not applicable to the PBMR. Although LBEs for the PBMR do not include LOCAs, they do include leaks and breaks in the Helium Pressure Boundary (HPB). Such breaks are referred to as depressurization events rather than LOCAs.

A second observation is that the treatment of LWR LBEs is prescriptive and the prescription is made in terms of LWR-specific failure modes and safety functions. The PBMR will utilize a systematic approach to produce a set of LBEs that are both comprehensive with respect to nuclear safety and demonstrate an adequate level of protection of the public health and safety.

2.1.2 NRC Policy Issues

The PBMR is subject to policy related to non-LWR applications. It is the intent of the PBMR design certification application to comply with this guidance and meet these policy requirements. The PBMR design responds to the Advanced Reactor Policy that encouraged innovative licensing approaches [5]

SECY 2003-0047, 'Policy Issues Related to Licensing Non-Light Water Reactor Designs' [6] offers staff recommendations on seven relevant policy issues that had been originally defined in an earlier policy statement, 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 treatment of LBEs and are discussed herein. The Staff Requirements Memorandum for SECY 2003-0047 [7] stated the Commissioners approval of the staff recommendations on both of these issues.

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' will be addressed by other papers on the fuel, reactor materials, analytical model and computer code verification and validation and the mechanistic source terms as part of the PRA and the deterministic safety analysis of the DBEs. The PRA will utilize deterministic analyses including those incorporated into the safety design approach, those for the development of success criteria and end states, those for the determination of the plant response to events, and those for the development of mechanistic source terms. The classification of SSCs will be based on their role in preventing and mitigating DBEs and will be subjected to deterministic requirements as described in another paper on the safety classification of SSCs.

With respect to Issue 5, 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 behavior under a wide range of scenarios and on ensuring fuel and plant performance is maintained over the life of the plant.

SECY 2003-0047 notes that in NUREG-1338 [8], the draft Pre-application Safety Evaluation Report on the Modular High Temperature Gas-cooled Reactor (MHTGR), 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 Advisory Committee on Reactor Safeguards (ACRS) stated in a letter dated February 19, 1993 [9] 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 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 PRA will include mechanistic source terms as well as radiological doses. Mechanistic source terms will also be used in the deterministic safety analysis for the deterministic design basis accidents that is part of the DCA. The DCA will establish 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.

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 1 as stated in the SECY. PBMR intends to select LBEs with a PRA of the multi-module plant. This provides flexibility for utilizing the design certification of the single reactor module as a single unit or in various multiples.

2.1.3 NRC Guidance

NRC guidance provides more detail on the type of accidents that constitute design basis accidents and transients for the currently licensed LWRs. In particular, NUREG-0800, Standard Review Plan (SRP) [10], identifies the types of AOOs and DBAs that must be postulated for LWRs. There is no comparable guidance for MHTGRs. As discussed above, the PBMR approach for selection of LBEs utilizes PRA.

LWR NRC guidance does not define DBAs and transients in probabilistic terms. However, in its June 26, 1990 Staff Requirements Memorandum (SRM) on SECY-90-16 [11], the Commission endorsed a Core Damage Frequency (CDF) goal of 10^{-4} per year for advanced reactors. Since accidents involving core damage are BDBAs, this implies that DBAs in general have a collective frequency greater than 10^{-4} per year (recognizing that some DBAs, such as double-ended guillotine breaks of large pipes, may have significantly lower frequencies than this value). For the PBMR LBE selection, event sequences with frequencies greater than 10^{-4} per plant-year[1] will be defined as DBEs. The basis for this frequency limit is discussed in Section 3 of this paper.

The NRC has not established a lower bound for the frequency of severe accidents that need to be considered. However, in general, the NRC does not require consideration of accidents that are not deemed to be 'credible'. Additionally, Regulatory Guide 1.174, Section 2.2.4, states that an increase in CDF of less than 10^{-6} per year and an increase in Large Early Release Frequency (LERF) of less than 10^{-7} per year are considered 'very small' and consistent with the Commission's Safety Goal Policy. These criteria are repeated in Section III.2.2.5 of SRP 19, *Use of Probabilistic Risk Assessment in Plant-Specific Risk-Informed Decision-making: General Guidance* [12]. Additionally, SRP 19 states that a PRA may have a Truncation Limit that, depending on the level of PRA detail (module level, component level, or piece-part level), may be from 10^{-12} to 10^{-8} per reactor-year. Similarly, Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities* [13], Section 1.2.5 states that an external event may be screened out of a PRA if it can be shown that the mean value of the frequency of the corresponding design-basis hazard used in the plant design is less than 10^{-5} per year and that the conditional core-damage probability is less than 10^{-1}, given the occurrence of the design-basis hazard. These guidelines indicate that events that have a frequency lower than ~10^{-6} or 10^{-7} per year do not need to be evaluated, and that events with a probability of less than about 10^{-8} may be screened from the PRA.

For the PBMR, all events and event sequences with frequencies greater than 5×10^{-7} per plant-year are considered candidates for LBEs via an all modes and all events and hazards PRA, using screening criteria consistent with Regulatory Guide 1.200.

On May 4, 2006, the NRC published an Advance Notice of Proposed Rulemaking (ANPR) in the Federal Register [14], identifying possible approaches to establish risk-informed performance-based requirements for nuclear power reactors. The ANPR contemplates the possibility of establishing subsidiary risk objectives of 10^{-5} per plant-year for accident prevention (i.e. prevention of major fuel damage) and 10^{-6} per plant-year for accident mitigation (i.e. prevention of releases that could cause early fatalities offsite). The ANPR also identifies a possible cut-off frequency for analysis of rare events of 10^{-7} per plant-year. However, the ANPR leaves open the question whether these frequencies should apply to individual reactors or all reactors at a site. Although the values being proposed for the PBMR are somewhat different than those proposed in the ANPR, we believe that the PBMR values are generally consistent with the ANPR, especially considering that the values for the PBMR are collective frequencies that will account for up to eight modules at a site.

[1] Frequencies per plant-year should not be confused with frequencies per reactor-year. For an eight-module plant, the frequency per plant-year of an event that impacts each reactor independently is eight times as great as the frequency per reactor-year.

2.2 REGULATORY FOUNDATION FOR ESTABLISHING TOP-LEVEL REGULATORY CRITERIA

The focus of this section is to define criteria that establish limits on the frequencies or consequences of LBEs and LBE categories that must be considered in the design and operation of a nuclear power plant in order to assure public safety and to assess the adequacy of the performance of SSCs that perform safety functions during these LBEs.

The following primary sources have been identified as containing criteria that establish limits on the risk or consequences of potential radiological releases from nuclear power plants in the U.S.

- **Reactor Safety Goal Policy Statement (51 FR 28044):** This policy limits public safety risk resulting from nuclear power plant operation. Limits are stated in the form of the maximum allowable risk of immediate death and the risk of delayed mortality from exposure to radiological releases of all types from nuclear power plants.
- **10 CFR Part 20, 'Standards for Protection against Radiation (Subpart D, Radiation Dose Limits for Individual Members of the Public)':** These criteria limit the dose consequences of releases associated with relatively high frequency events that occur as part of normal plant operations.
- **10 CFR Part 50, Appendix I, 'Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents':** This appendix provides explicit limits on doses from planned discharges that meet the NRC's definition of 'as low as is reasonably achievable'.
- **40 CFR Part 190, 'Environmental Radiation Protection Standards for Nuclear Power Operations':** These standards provide the generally applicable exposure limits for members of the general public from all operations except transportation and disposal or storage of spent fuel associated with the generation of electrical power by nuclear power plants.
- **10 CFR Part 100, 'Reactor Site Criteria (Subpart B, Evaluation Factors for Stationary Power Reactor Site Applications on or After January 10, 1997)':** §100.20 defines the Exclusion Area Boundary and Low Population Zones of a nuclear reactor site, and requires that the combination of the site and reactor located on that site be capable of meeting the dose and dose rate limitations set forth in 10 CFR §50.34(a).
- **10 CFR §50.34(a)(ii)(D), 'Contents of Applications: Technical Information (Radiological Dose Consequences)':** This section of the regulation specifies dose limits for evaluating the acceptance of the engineered safety features that are intended to mitigate the radiological consequences of accidents. These dose limits are consistent with those utilized in 10 CFR Part 100 for determining the extent of the EAB and Emergency Planning Zone (EPZ).

Each of these primary sources is discussed in greater detail below. They have been grouped into three sets of criteria, consistent with the category of event(s) to which they apply.

2.2.1 TLRC Related to Normal Operation and Anticipated Operational Occurrences

The U.S. Environmental Protection Agency (EPA) is the agency given the authority to set generally applicable regulations governing the acceptable level of radiological exposure to members of the public. Specifically, 40 CFR §190.10(a) states that the annual dose equivalent to a member of the general public from planned uranium fuel cycle operations shall be < 25 mrem to the whole body, < 75 mrem to the thyroid, and < 25 mrem to any other organ. Portions of these exposure limits must be allocated to the various elements comprising the uranium fuel cycle (e.g. uranium mining and milling, fuel production, and reactor operations to produce electrical power). While the definition of ‘uranium fuel cycle operations’ specifically references the production of electric power by LWRs, the inclusion of the alternative term ‘nuclear fuel cycle’ in the definitions section of the regulation, as well as its formal title, ‘Environmental Radiation Protection Standards for Nuclear Power Operations’, can be inferred to mean that it applies to other types of nuclear fuel as well as other types of reactors. Variances from these limits are allowed for unanticipated occurrences that still fall within the category of normal operations.

The NRC is the agency directly responsible for regulating the operation of nuclear power plants. As such, it is authorized to develop its own regulations, consistent with the requirements of 40 CFR Part 190, to ensure the health and safety of the general public. In exercising this responsibility, the NRC has promulgated a number of regulations that limit doses to the public from anticipated and unanticipated events during normal reactor operations.

10 CFR §50.34, 10 CFR Part 20, and 10 CFR Part 50, Appendix I all provide guidance on the limits for radiological releases from reactors during normal operations.

10 CFR §50.34(b)(3) states that the means for controlling and limiting effluent releases and radiation exposures during operation shall be capable of meeting the requirements set forth in 10 CFR Part 20. 10 CFR §20.1301 requires that the Total Effective Dose Equivalent (TEDE) for a member of the public be limited to 100 mrem per year and 2 mrem in any one hour, in unrestricted areas. This regulation provides the applicable criteria for limiting dose to the general public from anticipated and unanticipated events associated with the normal (non-accident) operation of a nuclear power plant.

10 CFR Part 50, Appendix I identifies dose and dose rate limits and limits on planned releases from the operation of nuclear power plant radwaste systems during normal operation, to maintain exposures As Low As Reasonably Achievable (ALARA). These criteria provide implementation guidance for applying the requirements of 10 CFR §50.34(a) and §50.36(a), for planned releases from the radwaste systems of nuclear power plants to the general environment to be ‘as low as is reasonably achievable.’ These ALARA limits are small fractions of the limits imposed by 10 CFR Part 20.

In setting its own quantitative limits on radiation exposure limits for the operation of nuclear power plants, the NRC has made use of the variance provided in 40 CFR §190.11 discussed earlier. The higher exposure limits set by 10 CFR Part 20 are associated with events still considered to lie within the regime of normal operations, but which are not considered to be ‘planned’ events as that term is used in 40 CFR Part 190.

The regulations do not define the term ‘normal operation’ in quantitative terms, i.e. the expected frequency of specified anticipated occurrences. However, Appendix A to 10 CFR Part 50 defines Anticipated Operational Occurrences (AOOs) as ‘those conditions of normal operation... expected to occur one or more times during the life of a nuclear power plant’.

2.2.2 TLRC Related to Design Basis Accidents

10 CFR §50.34(a)(1) contains NRC's regulations governing the design of new reactors and the means provided to protect against DBAs. This regulation requires that any reactor be designed such that:

- An individual located at any point on the EAB would not receive a radiation dose in excess of 25 rem TEDE for any two-hour period following the onset of a postulated fission product release.
- An individual located at any point on the outer boundary of the Low Population Zone (LPZ), exposed to the radioactive cloud resulting from a postulated fission product release, would not receive a radiation dose in excess of 25 rem TEDE.

10 CFR §50.34(a)(ii)(D) requires that these consequence limits are to be used when evaluating the acceptability of the features included in the plant design (i.e. engineered safety features and fission product barriers) for mitigating accident radioactive releases. The footnote pertaining to this section states that the fission product release to be assumed should be based 'upon a major accident... postulated from consideration of possible accidental events'. 10 CFR §100.21(c)(2) 'Reactor Site Criteria: non-seismic site criteria' requires that the radiological dose consequences of postulated accidents meet the criteria stated in 10 CFR §50.34(a)(1) for the type of facility located at the site in question.

In general, NRC's regulations do not define the type of events that comprise the category of DBAs. For LWRs, the General Design Criteria (Appendix A to 10 CFR Part 50) indicates that LOCAs must be considered as postulated accidents when designing safety systems. However, the category of 'postulated accidents' may include other types of events.

Nor do the regulations define DBAs in terms of their expected frequencies of occurrence, but 10 CFR §50.34(a)(i)(2) articulates the expectation that the design, construction and operation of nuclear power reactors will be such as to produce an 'extremely low probability of occurrence' for accidents that could release significant quantities of radioactive fission products. No quantitative definition of the term 'extremely low probability' is provided in the regulation.

2.2.3 TLRC Related to Beyond Design Basis Events

Current policy and guidance require that certain events outside the scope of the normal operation and DBE categories be considered in the design of nuclear power plants.

The NRC's *Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants* [15] states the Commission's intent to 'take all reasonable steps to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur'. As noted earlier, this policy statement specifically addresses the Commission's intent to resolve safety issues associated with 'accidents more severe than design basis accidents'. This policy statement also makes the following points with respect to the design and licensing of new nuclear power plants:

- New plants are expected to achieve a higher standard of severe accident safety performance than existing plants.
- Innovative, cost-effective ways of achieving improved overall reliability for systems that prevent or mitigate the consequences of severe accidents are supported by the NRC.

- Analyses of events beyond the design basis should be as realistic as possible, and make use of the insights provided by PRA.

In addition to its Severe Accident Policy, the Commission has issued a policy entitled *Safety Goals for the Operations of Nuclear Power Plants* [16]. Two qualitative safety goals are used to express the Commission's policy regarding the acceptable level of radiological risk from nuclear power plant operation as follows:

> *Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.*
>
> *Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.*

The following quantitative health objectives were identified as the basis for determining achievement of the above safety goals:

> *The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.*
>
> *The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.*

The statement of risks provided in the Safety Goal Policy envelops the spectrum of allowable risk associated with the operation of a nuclear power plant. As such, it clearly defines the outermost boundaries of acceptable risk associated with any event that has the potential to produce a radiological release affecting the environment or the health and safety of the general public.

2.3 SUMMARY

There are a number of legally-binding NRC criteria that explicitly constrain the risk and/or allowable consequences of radiological releases from nuclear power plants. These criteria include requirements to evaluate the adequacy of the proposed design of the plant against specific limits.

NRC regulations and policies also recognize the categorization of events that (generically termed Licensing Basis Events or LBEs in this paper) fall into three distinct categories: normal operation or Anticipated Operational Occurrences (AOOs); Design Basis Accidents (DBAs); and Beyond Design Basis Events (BDBEs).

All but one of the criteria described in this section provide their guidance in terms of consequence or release limits, not in the context of individual radiological risk as such. The exception is the Safety Goal Policy, which specifies two quantitative risk metrics. Each of the other regulatory requirements discussed in this section may, however, be regarded as providing useful risk limitation guidance, because it is possible to associate types of events with particular frequency and consequence or release limits. For example:

1. 10 CFR Part 20, 10 CFR Part 50 Appendix I, 40 CFR Part 190, and 10 CFR §50.34(b) pertain to normal operations and AOOs. These types of events would be expected to occur sometime during the operational life of a plant;
2. 10 CFR §50.34(a) pertains to DBEs that are not expected to occur during the life of a plant but could reasonably occur within the operational life of a large fleet of plants; and
3. The Safety Goal Policy generally pertains to accidents within and beyond the design basis of the plant.

Additional sources of information on the applicability of the criteria cited above exist in other forms, such as implementation guidance provided by NRC Regulatory Guides or in NUREGs. These sources offer additional relevant information for development of LBEs for the PBMR. One other source of information that is useful in defining the PBMR basis is the record of licensing submittals and decisions pertaining to the MHTGR.

3. PBMR APPROACH

Each of the TLRC identified in the regulations and policy of Section 2 have associated with them an explicit or implicit frequency range. The assignment of frequency ranges is discussed in Section 3.1. A PRA is utilized as the best available tool to assure a comprehensive understanding of off-normal events. A brief summary of the PBMR approach to performing a comprehensive PRA is provided in Section 3.2. The subsequent sections provide a detailed description with examples of the method for selecting the three categories of LBEs. The method builds on the LBE selection methodology presented in prior modular HTGR preapplication interactions [17], [18], and [19]. Section 3.5 describes the derivation of deterministic DBAs that are the basis for showing regulatory compliance of the conservatively calculated consequences of accidents, typically in Chapter 15 of a Safety Analysis Report. These deterministic DBAs are an essential element of the PBMR certification approach, and provide an important complement to the LBEs selected with the PRA. Finally, Section 3.6 examines the full spectrum of LBEs in comparison to the NRC Safety Goal QHOs.

3.1 FREQUENCY RANGES OF THE TOP LEVEL REGULATORY CRITERIA

The spectrum of potential accidental radioactive releases from the PBMR plant is divided into three regions of a scenario frequency versus consequence chart. The regions include those associated with:

- Anticipated Operational Occurrences
- Design Basis Events
- Beyond Design Basis Events

An examination of the entire frequency range and the identification of one or more of the TLRC as being applicable for each region provide assurance that the selected criteria are adequately established.

3.1.1 Anticipated Operational Occurrences Region

AOOs are those conditions of plant operation which are expected to occur one or more times during the life of the plant. Current plants were licensed to operate for an initial 40-year period; however, with the advent of license renewal, operating licenses of conventional plants have been increased for some plants by 20-year increments. Therefore, a conservative value of 1×10^{-2} is used to establish the lower bound of the AOO region. For this region, 10 CFR Part 20 provides the applicable criteria, as it specifies the numerical guidance to assure that releases of radioactive material to unrestricted areas during normal reactor operations, including AOOs, are maintained ALARA.

3.1.2 Design Basis Event Region

The DBE region encompasses releases that are not expected to occur during the lifetime of a single nuclear power plant, but may be encountered during the lifetime of a population of nuclear power plants. Therefore, a value of 1 x 10^{-4} per plant-year (or 1.25 x 10^{-5} per reactor-year in the case of a commercial eight-reactor module PBMR) is used to establish the lower bound of this region. There is no need to require 10 CFR §50.34 to be met at frequencies lower than 10^{-4} per plant-year to meet the NRC Safety Goal QHOs; at 10^{-4} per plant-year they are met with margin.

As discussed in Section 2, 10^{-4} per plant-year as the lower frequency for the design basis region is consistent with LWR regulatory guidance for the design goal frequency of core damage events which are not DBEs.

For the DBE region, the 25 rem TEDE criterion in 10 CFR §50.34a provides the quantitative dose guidance for accidental releases for siting a nuclear power plant to ensure that the surrounding population is adequately protected. The combination of the selected frequency limits and dose limits for the DBE region ensures that the NRC Safety Goal QHOs for individual risk of latent cancer fatality is met by several orders of magnitude for all event sequences within the DBE region.

3.1.3 Beyond Design Basis Event Region

The BDBE region, comprising improbable events that are not expected to occur during the lifetime of a large fleet of nuclear power plants, should be considered to assure that the risk to the public from low probability events is acceptable. The frequency cutoff implicit in the acute fatality risk goal in NUREG-0880 is taken as the lower frequency boundary of the BDBE Region. NUREG-0880 notes that the individual mortality risk of prompt fatality in the U.S. is about 5 x 10^{-4} per year for all accidental causes of death. The prompt mortality risk design objective limits the increase in an individual's annual risk of accidental death to 0.1% of 5 x 10^{-4}, or an incremental increase of no more than 5 x 10^{-7} per year. If the frequency of a scenario or set of scenarios is at or below this value, it can be assured that the individual risk contributions from these scenarios would still be within the safety goal, independent of the magnitude of the consequences with a significant residual margin.

3.2 USE OF PROBABILISTIC RISK ASSESSMENT

The PBMR PRA will provide a logical and structured method to evaluate the overall safety characteristics of plants [20]. This will be 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 will capture the dependencies and interactions among SSCs, human operators, and the internal and external plant hazards that may perturb the operation of the plant. The quantification of both frequencies and consequences will address uncertainties, especially those associated with the potential occurrence of rare events. 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 against the TLRC.

A full scope PRA for the PBMR Demonstration Power Plant (DPP) is under development [21]. The scope of the PBMR PRA will be as comprehensive and sufficiently complete as would be covered in a full-scope, all modes, Level 3 PRA covering a full set of LWR internal and external events. For the DCA, the PRA will be revised to correspond to the multi-module PBMR DCA design and will be modified to apply to a spectrum of U.S. sites.

3.3 SELECTION OF ANTICIPATED OPERATIONAL OCCURRENCES

Figure 1 depicts the frequency range for AOOs. As shown, the range covers very frequent events expected to occur several times a year to events that are as rare as once in a hundred years. The ordinate in the figure is the mean or expected frequency per plant-year to include releases from one or more reactors or sources within the plant. The abscissa is the mean or expected consequence to the public, so it is measured at the Controlled Area Boundary (CAB) as required by 10 CFR §§20.1301 and 20.1302. 10 CFR Part 20 is plotted in the figure as the limiting TLRC to encompass an acceptable and an unacceptable region for AOOs. 10 CFR Part 20 is shown with a break in the line at the frequency of once per year, since it is an annualized limit, that is, the sum of all releases in a given year should not exceed 0.1 rem to the whole body gamma dose. For example, only two events with a release of one half of the 0.1 rem limit can occur per year. Events with frequencies less than once per year must each meet the 0.1 rem limit.

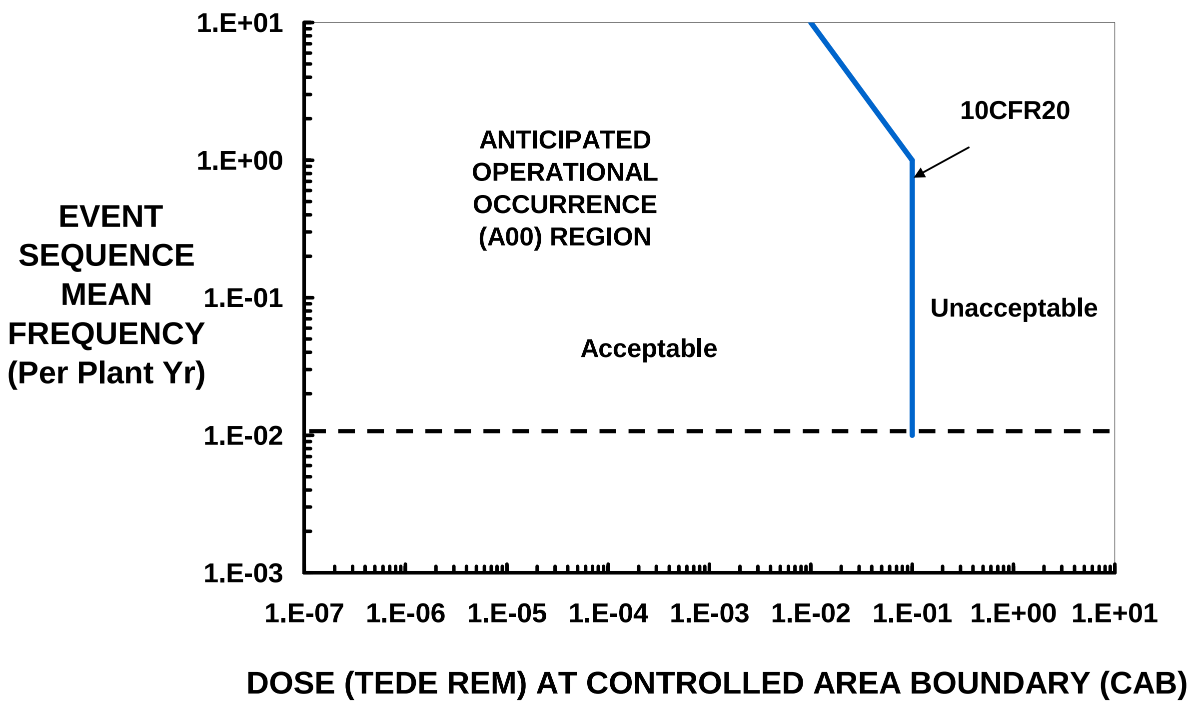


Figure 1: Frequency-Consequence Chart for Anticipated Operational Occurrences

The next step is to plot the events from the PRA on the chart. An abbreviated PRA event tree for the loss of Power Conversion Unit (PCU) from the earlier 268 MWt PBMR DPP is provided in Figure 2. The figure is a stylized event tree presented to illustrate the dominant branches in the PRA that result from a transient in which the normal operation PCU is lost. The initiating event frequency is shown as 3.4 x 10^{-2} per reactor-year. Note that the demonstration plant is one reactor module. The event sequences start with the initiating event and proceed to the full event sequence, that is, if there are additional failures after the initiating event, they are included in the event sequence. The first branch in the tree is related to the function to trip the reactor by automatically or manually inserting the Reactivity Control System (RCS) control rods or the Reserve Shutdown System (RSS) backup poison material (small absorber spheres). As shown, successful functions are in the upward direction. The succeeding branches all relate to the core heat removal function. Three systems are available in the earlier 268 MWt demonstration plant for this function: the Start-up Blower System (SBS), the Core Conditioning System (CCS), and the Reactor Cavity Cooling System (RCCS). The first two involve the forced convection of the helium coolant through the core to heat exchangers to transport the heat to water cooling systems and from there to plant site heat sinks. The RCCS relies on passive natural convection, conduction, and radiation heat transfer from the core through the uninsulated reactor vessel to surrounding water tanks. The RCCS has an active mode in which the water in the tanks is circulated through a heat exchanger to a separate water cooling system and a passive mode in which the water in the tanks boils off. Figure 2 shows the event sequence families for each of these three cases, labelled AOO 1a, AOO 1b, and DBE 1a, respectively.

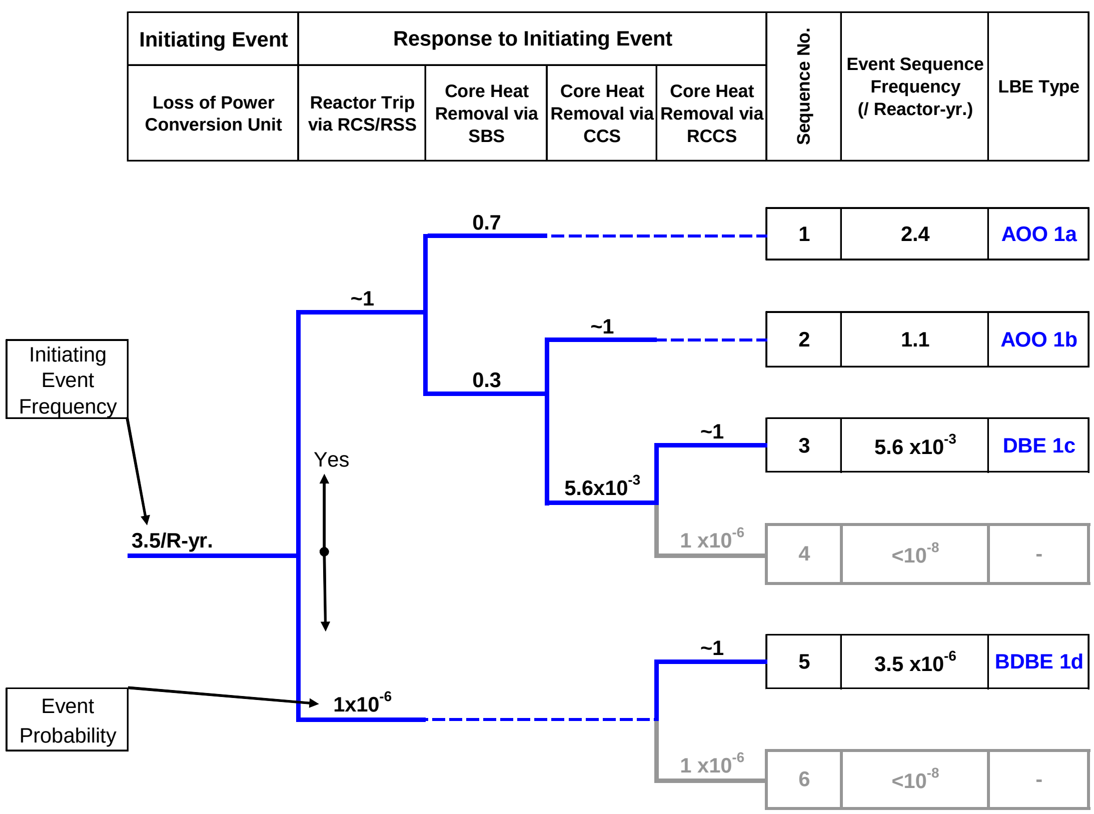


Figure 2: Abbreviated Event Tree for Loss of Power Conversion System (PCU)

As discussed in the PRA paper, 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 a combination would mask the definition of unique challenges to the SSCs 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.
- 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.

A common situation that yields accident 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 the Figure 2 loss of PCU initiating event tree, separate initiating events were developed for the main contributors to the initiating event, e.g. turbine failure, bypass valve failure, loss of secondary water flow, but since the event sequences follow the same event tree logic and result in the same end states, they are aggregated into a family.

Without the use of event sequence families, the level of detail in the initiating event categories and event 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 structure of the event sequence model does not impact the LBE classification.

Figure 3 takes the PRA results from the event trees and shows the event frequencies and consequences on the Figure 1 chart. The events with the same initiating event are labeled with the same numerical designation and depicted in the same color. For example, the Loss of the Power Conversion System example are all LBEs labeled 1 and are shown in blue font. Different sequences within a given initiating event tree are denoted by an alphabetical letter following the initiating event number; that is, in AOO 1a the SBS provides forced core cooling for decay heat removal, whereas in AOO 1b the CCS performs the forced core cooling function.

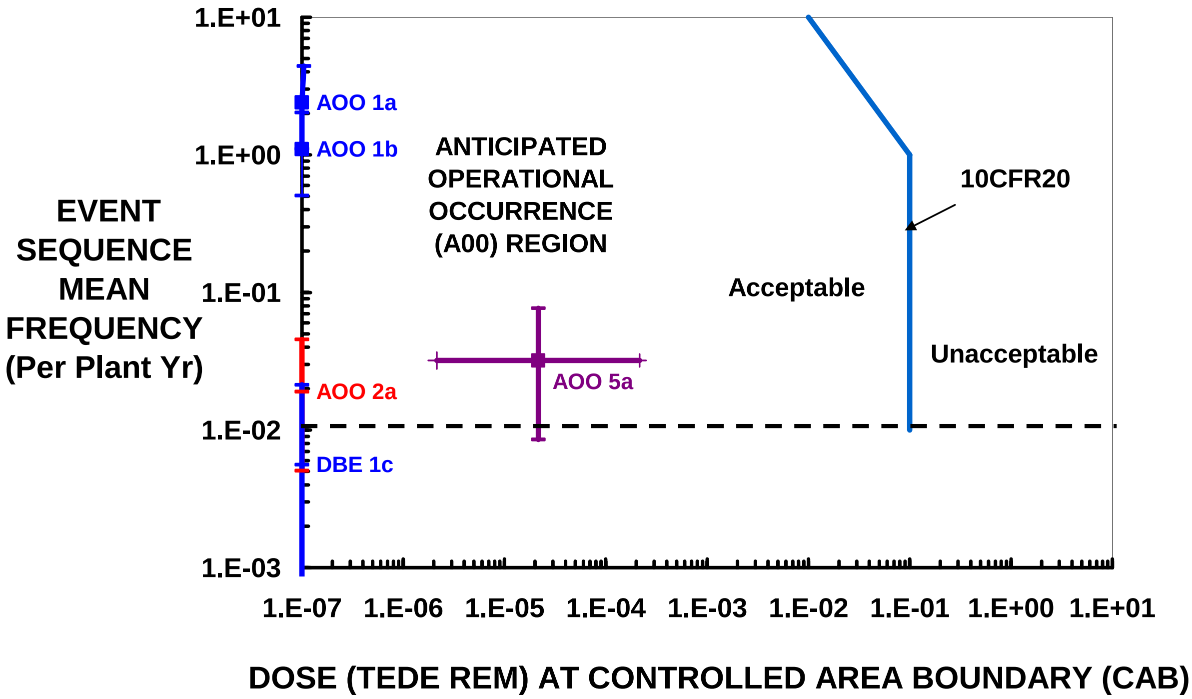


Figure 3: Use of PRA to Select Anticipated Operational Occurrences

AOOs are selected from those families of events whose mean frequency falls within the AOO region, as shown on the risk criteria chart in

Figure 3, and that would exceed the 10 CFR Part 20 criteria on a mean value basis if it were not for design selections that control radionuclide release. Those that meet this condition are designated as AOOs. Although AOO 1a and AOO 1b do not have releases, it has been determined that they would be in the unacceptable region if it were not for a number of design selections that include successful decay heat removal.

Events may have significant uncertainties in the estimate of their frequencies and consequences. The consideration of these uncertainties is necessary to ensure that all events will be assessed against the appropriate criteria. The mean value of frequency, which involves an integral over the complete uncertainty spectrum, is the selected parameter for accounting for frequency uncertainties. In addition, the upper and lower uncertainty bands are shown on the event points in

Figure 3. If the upper bound frequency of an event is above the 10^{-2} per plant-year lower frequency range of AOOs, it is evaluated as an AOO as well as a DBE. DBE-1c is the example whose event sequence is shown in red in the Figure 2 abbreviated event tree in which a loss of the CCS leads to heat removal with the RCCS. Since it has an upper bound frequency that exceeds the 10^{-2} per plant-year, it is shown on the AOO plot.

AOOs typically have relatively small consequences associated with them. Only one of the AOOs shown in

Figure 3, AOO 5a, has a non-zero consequence. (Note that AOO 5a event sequence is shown in Figure 6.) Furthermore, in general, uncertainties in the consequences of AOOs are relatively small, and are monitored and reduced during the life of the plant. Therefore, although the PRA assessment provides the entire consequence distribution, including the mean, and upper and lower bound doses, it is appropriate that the consequences of AOOs meet 10 CFR Part 20 criteria on a mean-value basis.

3.4 SELECTION OF DESIGN BASIS EVENTS

Figure 4 is utilized for the selection of DBEs. The figure plots 10 CFR §50.34 on a frequency-consequence chart over the frequency range of 10^{-2} to 10^{-4} per plant-year. There are a number of similarities with the corresponding AOO chart: the event sequence frequency is measured in events per plant-year and the dose is measured at the EAB. However, 10 CFR §50.34 doses are measured in TEDE units, which take into account the relative importance of the various doses such as whole body gamma and thyroid in arriving at a weighted health effect to an individual. Since DBEs are not expected in the plant lifetime, there is no summation of frequent events as in the 10 CFR Part 20 case for AOOs. The 10 CFR §50.34 limit is 25 rem TEDE per event sequence. However, recognizing that more frequent events in the upper frequency range of DBEs should have a lower risk, a value of 10% of the limit has been assigned to the top of the DBE region.

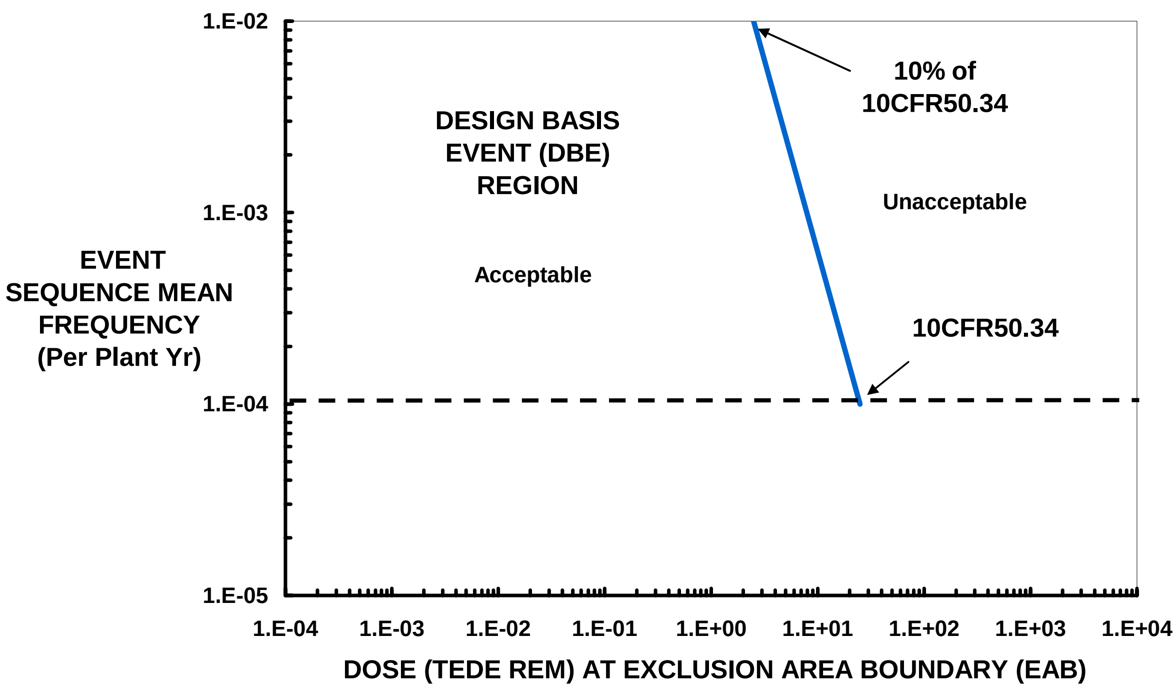


Figure 4: Frequency-Consequence Chart for Design Basis Events

Figure 5 shows the next step in the process for DBE selection with the PRA results plotted on the DBE frequency-consequence chart. In a similar fashion as for AOOs, DBEs are selected from those families of events whose mean frequency falls within the DBE region and that would exceed the 10 CFR §50.34(a) dose criteria on a mean value basis if not for design selections that control radionuclide release. Those that meet this condition are designated as DBEs. The designations of the initiating event families shown in Figure 5 are provided in Table 1.

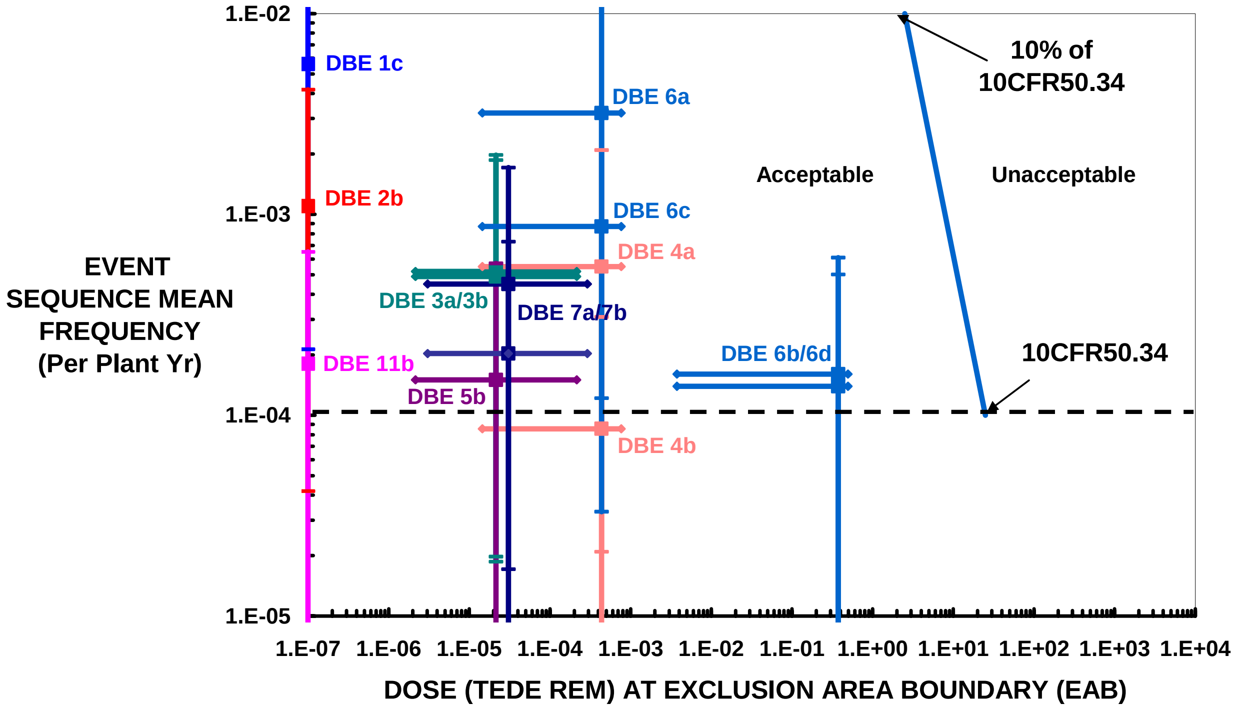


Figure 5: Use of PRA to Select Design Basis Events

The preliminary PRA results shown in Figure 5 indicate that the DBE dose limit was met. Note however, the larger uncertainties, particularly in the frequencies. As before, all DBEs must meet the 10 CFR §50.34 limit. PBMR does not take the most limiting event with respect to dose, for example DBE-6, and assume that because it meets the 10 CFR §50.34 limit that the safety design of the DBEs is complete. Because different DBEs may rely upon different SSCs to achieve acceptable results, PBMR proposes to use the results of its analyses of all DBEs to define a complete set of the SSCs that impact the range of events.

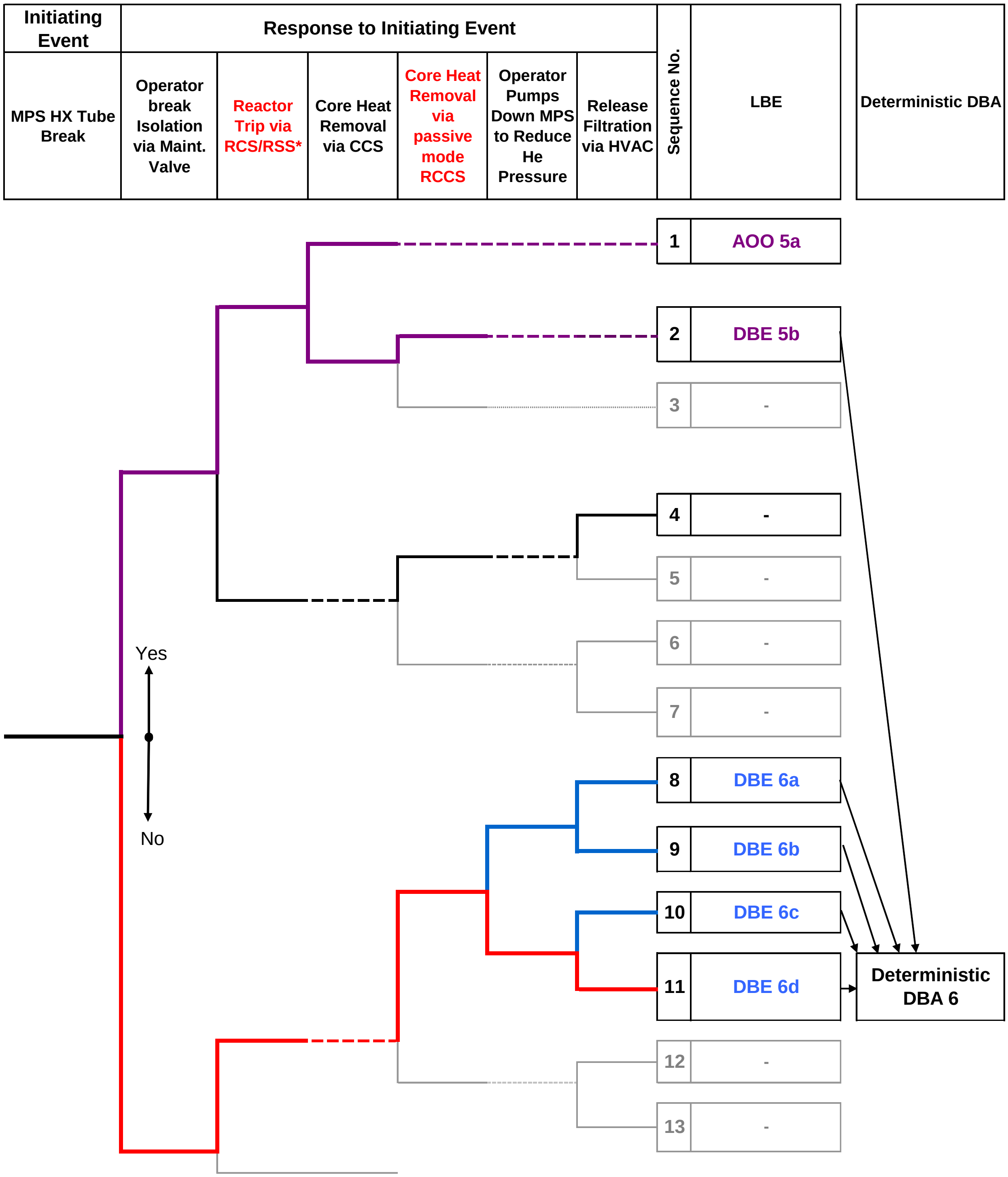
Table 1: Identification of DBE Initiating Events in Preliminary PRA for the 268 MWt PBMR Demonstration Power Plant

DBE Designation	Initiating Event
DBE-1	Loss of Power Conversion System
DBE-2	Control Rod Group Withdrawal
DBE-3	Small Isolated HPB Break
DBE-4	Small Unisolated HPB Break
DBE-5	MPS Heat Exchanger Tube Leak Isolated
DBE-6	MPS Heat Exchanger Tube Leak Unisolated
DBE-7	Medium Isolated HPB Break
DBE-11	Safe Shutdown Earthquake

3.5 DETERMINISTIC DBAS FOR SAR CHAPTER 15 EVALUATION

PBMR plans to identify deterministic DBAs from the DBEs by assuming that only SSCs classified as safety-related are available to perform the safety functions required to meet 10 CFR §50.34. After the safety-related SSCs are selected, all of the DBEs are reanalyzed deterministically with only the safety-related SSCs responding in a mechanistically conservative manner.

The deterministic DBAs generally do not have the same sequence of events as the corresponding DBEs, since the latter consider the expected plant response with all SSCs responding whether safety-related or not. Figure 6 shows the deterministic DBA-6 superimposed on the Main Power System (MPS) Heat Exchanger (HX) break abbreviated event tree. The safety-related SSCs are shown in red font. For this example the SSCs indicated are assumed to perform the required functions for reactor shutdown and core heat removal. The other SSCs are assumed to not be available. The approach to safety classification is the subject of another paper. Thus, deterministic DBA-6, which considers only the response of the safety-related SSCs, is identical to DBE-6d and bounds DBE-5b and the other DBE-6a, -6b, and 6c. Figure 7 provides another example for the loss of PCU initiating event presented earlier. In this case, deterministic DBA-1 is identical to DBE-1c.



* safety-related SSCs shown in red font

Figure 6: Illustration of Deterministic Design Basis Accident for PBMR Main Power System Heat Exchanger Initiating Event

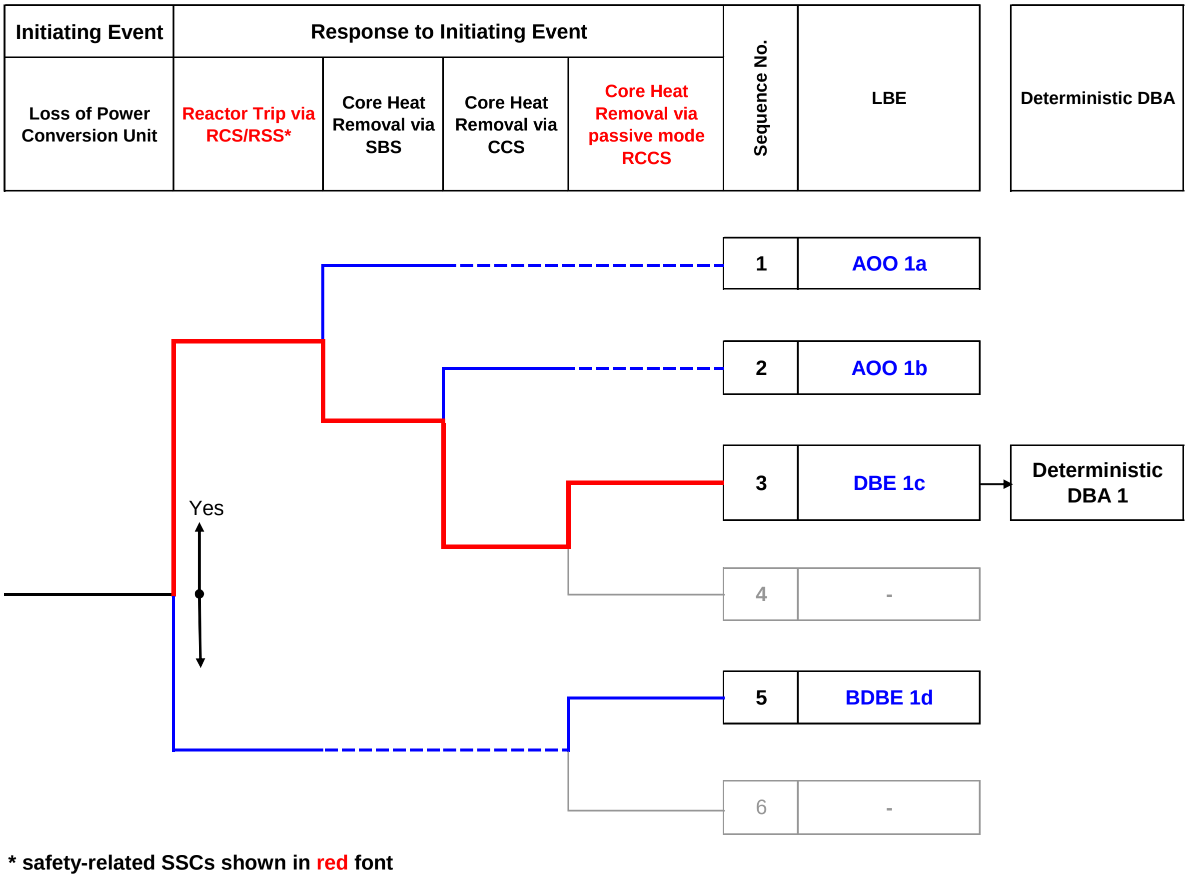


Figure 7: Illustration of Deterministic Design Basis Accident for PBMR Loss of Power Conversion Unit Initiating Event

Note that the deterministic DBAs are the analog of the traditional LWR DBAs analyzed in Chapter 15 of the Safety Analysis Report. A key advantage is that the safety-related SSCs with their basis rooted in PRA are designed for the expected response of the entire plant (for the DBE sequence families) as well as the safety-related response (the deterministic DBAs). Furthermore, the approach allows the transition to be made to the traditional deterministic response with only safety-related SSCs responding to deterministic DBAs and all SSCs responding to DBEs, so that both the conservative and expected plant behaviour are understood.

Table 2 provides the list of deterministic DBAs and their relation to the DBEs. For example, DBE 5b is the event sequence family in which the HX break is manually isolated, whereas in the corresponding deterministic DBA 6, the safety-related response of the plant does not have this operator action.

Table 2: Relation of PBMR Deterministic Design Basis Accidents to Design Basis Events

DBE Designation	Design Basis Event	Deterministic DBA Designation	Deterministic Design Basis Accident
DBE-1c	Loss of PCU w/core conduction cooling to RCCS	Deterministic DBA-1	Loss of PCU w/core conduction cooling to passive mode of RCCS
DBE-2b	Control rod withdrawal w/CCS forced cooling	Deterministic DBA-2	Control rod withdrawal w/core conduction cooling to passive mode of RCCS w/unfiltered release
DBE-3a	Small, auto isolated HPB break w/SBS forced cooling	Deterministic DBA-3	Small, unisolated HPB break w/core conduction cooling to passive mode of RCCS w/unfiltered release
DBE-3b	Small, manually isolated HPB break w/CCS cooling		
DBE-4a	Small, unisolated HPB break w/pumpdown w/RCCS cooling	Deterministic DBA-4	Small, unisolated HPB break w/core conduction cooling to passive mode of RCCS w/unfiltered release
DBE-4b	Small, unisolated HPB break w/o pumpdown w/RCCS cooling		
DBE-5b	HX tube break, manually isolated w/RCCS cooling	Deterministic DBA-6	HX tube break, unisolated w/core conduction cooling to passive mode of RCCS w/unfiltered release
DBE-6a	HX tube break unisolated w/pumpdown w/RCCS cooling w/filtered release		
DBE-6b	HX tube break unisolated w/pumpdown w/RCCS cooling w/unfiltered release		
DBE-6c	HX tube break unisolated w/o pumpdown w/RCCS cooling w/filtered release		
DBE-6d	HX tube break unisolated w/o pumpdown w/RCCS cooling w/unfiltered release		
DBE-7a	Medium, auto isolated HPB break w/SBS cooling	Deterministic DBA-7	Medium, unisolated HPB break w/core conduction cooling to passive mode of RCCS
DBE-7b	Medium, isolated HPB break w/CCS cooling		
DBE-11a	Safe shutdown earthquake w/SBS cooling	Deterministic DBA-11	Safe shutdown earthquake w/core conduction cooling to passive mode of RCCS
DBE-11b	Safe shutdown earthquake w/CCS cooling		

Even with the consolidation of the DBEs into a smaller number of deterministic DBAs, there is still a spectrum of challenges that must be addressed based on the initiating event and on the progression of the events. Furthermore, as with the DBEs, a deterministic DBA with no consequences such as DBA-1 is just as important as the one with the highest predicted consequences in terms of identification of SSCs that should be classified as safety-related.

3.6 SELECTION OF BEYOND DESIGN BASIS EVENTS

While the BDBEs are not part of the DBAs, the BDBEs do form an important element of the licensing basis to assure that the overall risk to the public is acceptably low and that adequate emergency planning is in place. BDBEs are selected from those families of events with doses whose mean frequency falls within the BDBE region as shown in Figure 8. The mean public consequences of the DBEs and BDBEs are the relevant measure of acceptance. As discussed in prior sections, the limiting TLRC for DBEs is the dose limit of 10 CFR §50.34. There is not a corresponding dose limit shown on the figure for BDBEs on a per event basis. However, BDBEs together with the AOOs and DBEs are evaluated in a cumulative manner and compared in terms of individual risk to the Reactor Safety Goals as discussed in the next section. The NRC Safety Goal acute fatality is shown on the plot as a bounding limit in that it is plotted at the EAB. If each BDBE meets this limit, the cumulative risk of all LBEs will meet the NRC Safety Goals with large safety margins as discussed in the next section.

Events below the 5 x 10^{-7} per plant-year BDBE region are examined to assure that the residual risk is negligible with respect to the latent mortality safety goal as discussed in the next section, and to provide general assurance that no potentially high consequence events go unnoticed.

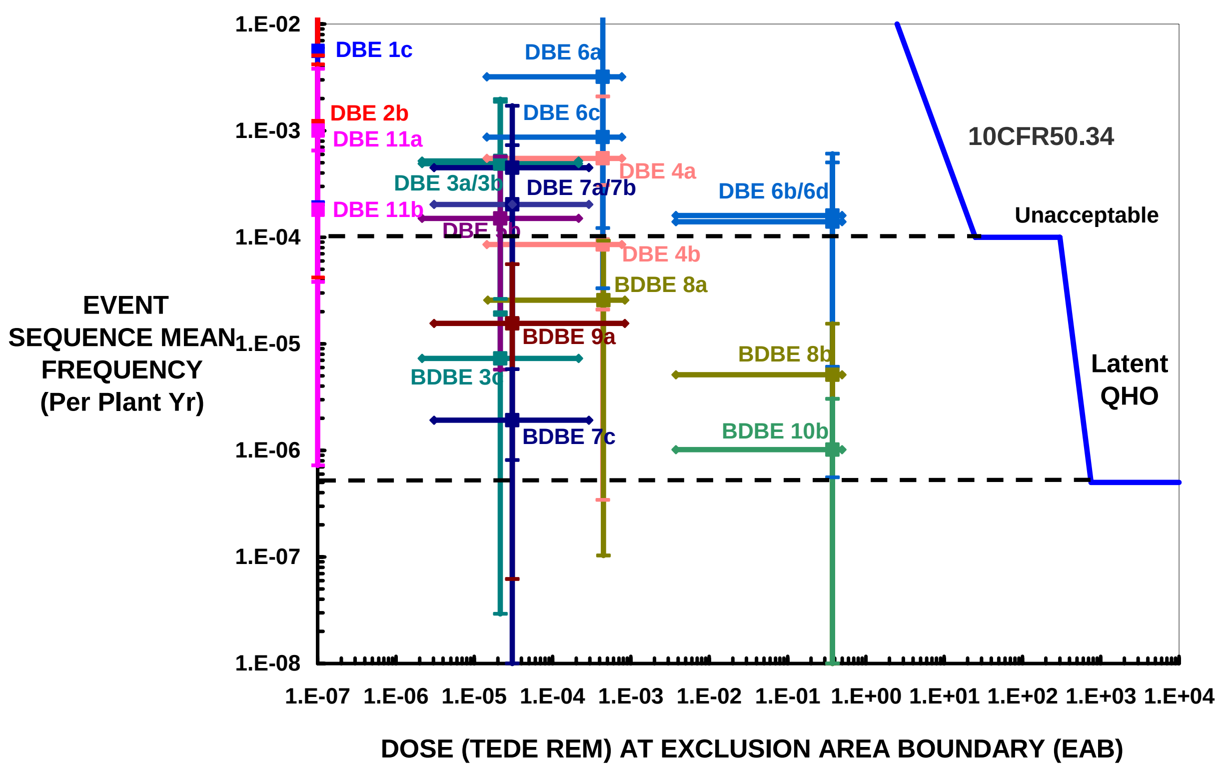


Figure 8: Use of PRA to Select Beyond Design Basis Events

3.7 EVENT CONSEQUENCE EVALUATION

The preceding sections have discussed the selection of the LBEs and deterministic DBAs. Although the frequencies and consequences of the LBEs have been shown on the frequency-consequence charts, the emphasis of the discussion has been on the frequency of the event sequence families in relation to the three LBE regions. This section summarizes the approach to the consequence evaluation.

For each LBE and deterministic DBA, mechanistic source terms will be developed that evaluate the realistic response of the plant to the initiating event. The initial radionuclide inventories during the modes of normal operation will include those in the fuel, the circulating activity, the plateout activity within the HPB, the spent and used fuel, and the radwaste sources. For each of these inventories, the response to the initiating event of the barriers and that of the passive and active SSCs that protect those barriers will be modeled. The transport of the radionuclides from their source through the PBMR barriers, depending on the sequence, to the public will be mechanistically modeled with uncertainty distributions on the expected values. Thus, temperatures, flows, pressures, and concentrations with uncertainty bands will be combined statistically to provide expected and upper and lower bounds at intermediate points along the transport path to the EAB. Thus, the constituents of the uncertainty bands on the TEDE doses shown in the previous figures will be presented. Mechanistic source terms for the consequences for deterministic DBAs will also be evaluated in the same fashion, although for these deterministic accidents only safety-related SSCs respond.

The appropriate measure of acceptance varies for each category of event are as follows:

- **Anticipated Operational Occurrences:** The consequence distribution for each AOO will be compared to the 10 CFR Part 20 public consequence limit shown in Figure 1. The acceptance criterion is that the expected or mean consequence of the AOO must be less than the 100 mrem TEDE limit at the CAB as indicated in the figure. Frequent events predicted to occur more than once a year must meet a fraction of the limit.
- **Design Basis Events:** The consequence distribution for each DBE is compared to the 10 CFR §50.34 public consequence limit. The acceptance criterion is that the upper bound of the mean consequence of each DBE must be less than the 25 rem TEDE limit at the EAB as indicated in the figure. Events with frequencies closer to the AOO region must meet a fraction of the limit.
- **Deterministic Design Basis Accidents:** The consequence distribution for each deterministic DBA is compared to the 10 CFR §50.34 public consequence limit. The acceptance criterion is that the upper bound of the mean consequence of each DBA must be less than the 25 rem TEDE limit at the EAB.
- **Beyond Design Basis Events:** The consequence distribution for each BDBE is, together with the AOOs and DBEs, compared to the Safety Goals as discussed in the next section.

For events involving more than one reactor module, the consequences from each involved reactor module are summed prior to comparison to the acceptance criterion.

3.8 CUMULATIVE LICENSING BASIS EVENT EVALUATION

Figure 9 presents all three categories of LBEs on one chart in comparison to the respective TLRC. A final acceptability demonstration is to sum the risks from all of the events and to compare the results to the Reactor Safety Goals.

The Reactor Safety Goals are discussed in NUREG-0880 in terms of quantitative health objectives. PBMR expects, based on the DPP safety analyses to date, that the DCA will show that there are no LBEs with doses sufficiently high to cause an acute fatality at the EAB. The latent fatality QHO limits the increase in an individual's annual risk of death to 0.1% of 2 x 10^{-3} per person-year, or an incremental increase of no more than 2 x 10^{-6} per person-year. The evaluation process is to sum the risks (the product of the frequency and consequence) from each LBE, adjust the dose from the EAB to midway in the annular region between the EAB and 10 miles from the plant, and convert the health effect doses to fatalities. To aid in the visualization of the evaluation, iso-risk lines have been superimposed on Figure 9 to highlight the highest risk points from the preliminary PBMR results. As shown, DBEs-6b and -6d will dominate the risk summation by over an order of magnitude. The explicit equation for the PBMR risk of latent cancer fatality is given by

$$R_{LCF} = \sum_j F_{RCj} \sum_k p_k \sum_l d_{jkl} n_l r_{LCFd} < 2x10^{-6} \, / \, person - year$$

n_l = fraction of the total population that is within sector l (the annulus between the EAB and 10 miles from the plant is divided up into sectors in a polar coordinate grid with radius ρ and angle θ)

r_{LCFd} = probability of latent fatality given radiation exposure of dose level d

Because the PBMR consequences are very low for an individual at the site boundary (less than the 10 CFR §50.34 dose limit), the consequences to an average individual within 10 miles is extremely low. The result of this summation of products for the preliminary PBMR results is a risk that is five orders of magnitude less than the QHO individual latent cancer risk limit of 2 x 10^{-6} per person-year. Another way of stating the result is that if all events including the BDBEs meet the dose limits at the site boundary, then the QHOs within the larger distances from the site are met with large margins.

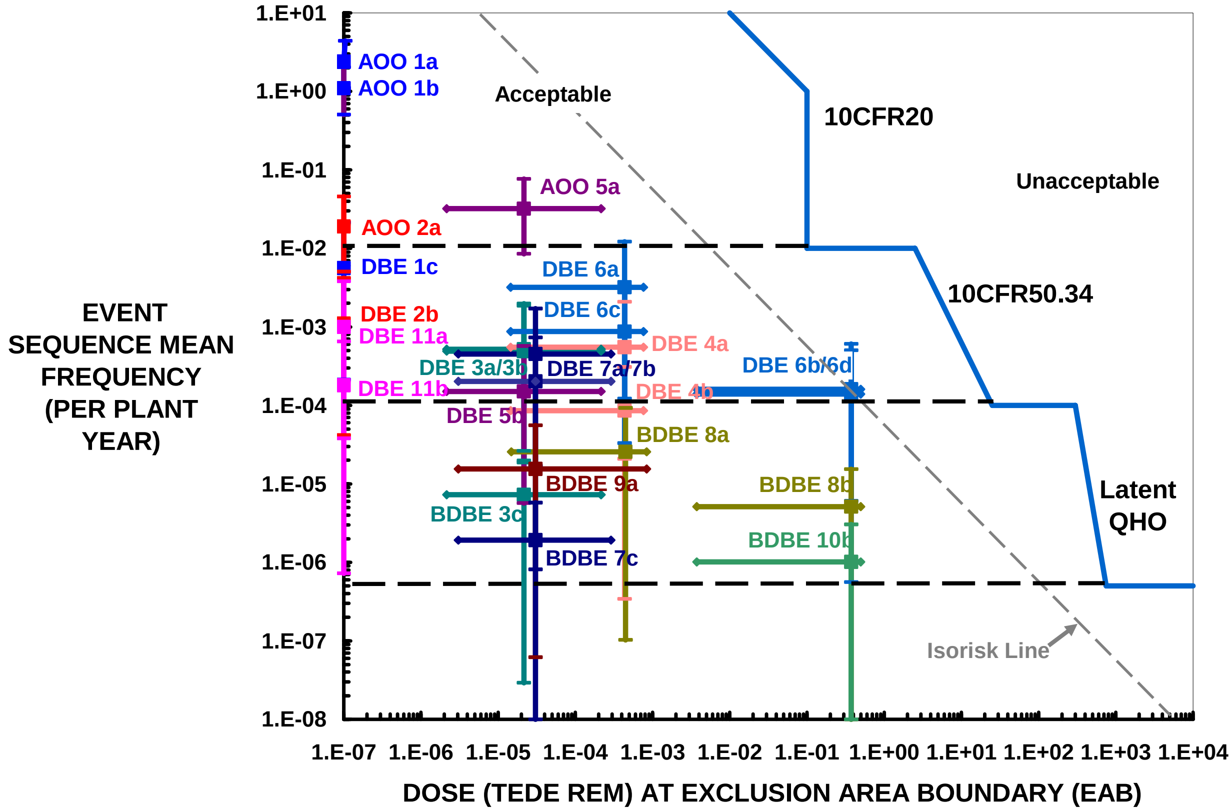


Figure 9: Frequency-Consequence Chart for all Three Categories of Licensing Basis Events

4. ISSUES FOR PREAPPLICATION RESOLUTION

The issues addressed in this paper are framed in terms of the following questions about the selection of LBEs that will be performed to support the PBMR DCA. The PBMR position on the appropriate response to these questions has been discussed in detail in Section 3 and summarized below, following the listing of each question.

1. **What is an appropriate, systematic, and reproducible approach for selecting LBEs for the PBMR?**

   PBMR Response: An appropriate approach is to derive a set of LBEs from the PBMR PRA that is performed according to the approach described in the PRA Paper. Each LBE will be defined as a family of event sequences from the PRA having a similar initiating event, the same plant response, in terms of which safety functions are successfully provided and which are not, and a common end state that justifies the application of the same mechanistic source term.

2. **What is the appropriate blend of probabilistic and deterministic approaches for the selection and analysis of LBEs for the PBMR? What requirements must be applied to the PRA and supporting deterministic evaluations in order to support LBE selection and evaluation for the PBMR?**

   PBMR Response: An appropriate blend includes a design-specific PRA as described in the PRA Paper and the following deterministic elements: the deterministic safety design approach of the PBMR, an engineering analysis of the plant response to each initiating event using deterministic and verified computer models, deterministic success criteria, deterministic methods used to predict the mechanistic source term, deterministic selection of safety-related SSCs, conservative deterministic safety analyses of design basis accidents in Chapter 15 of Tier 2 of the DCD, and development of deterministic regulatory design requirements for the safety-classified SSCs. Both the probabilistic and deterministic analysis will be supported by a comprehensive and systematic search for initiating events including internal events and internal and external plant hazards that could occur during all operating and shutdown modes, and covering the sources of radioactive material.

3. **What categories of LBEs need to be considered?**

   PBMR Response: The three categories of LBEs and their purposes adhere to the regulations and policy. The LBEs for the PBMR include AOOs, DBEs, and BDBEs used in NRC regulatory policy and guidance. DBEs, as well as AOOs and BDBEs, are selected through the use of the PRA and are based on a realistic response of the entire plant response. This is the necessary foundation for understanding the safety functions and the SSCs available to perform them. This leads to the safety classification of PBMR SSCs, which is the subject of another paper. Once the SSC safety classification is known, the deterministic DBAs are derived from the DBEs by demonstrating success paths relying solely on their response of PBMR safety-related SSCs, as in the conventional regulatory practice.

4. **What are the acceptable public consequences and analysis bases for each LBE category?**

   PBMR Response: The acceptable public consequences have been taken directly from the existing regulations and policy in Section 2. In summary, the limits in 10 CFR Part 20 are applied to AOOs, and the limits in 10 CFR §50.34 are applied to the DBEs and the deterministic DBAs. The Safety Goal QHOs are applied to all the LBEs in a cumulative manner. The analyses bases follow the conventional practice for each of the LBE categories and respective TLRC.

5. **What is the frequency range for each LBE category?**

   PBMR Response: The frequency ranges for each category are not explicitly stated in NRC regulation or guidance. For AOOs, PBMR proposes a lower frequency limit of 10^{-2} per plant-year. For DBEs selected with PRA, PBMR proposes a lower frequency range for event sequences of 10^{-4} per plant-year, which meets the NRC Safety Goals and is consistent with LWR regulatory practice. For BDBEs, PBMR proposes a lower limit of 5×10^{-7} per plant-year.

   To account for the modular nature of the PBMR, PBMR proposes that the frequency be stated on a 'per plant-year' basis. For an eight (8) reactor module plant, the frequency for a 10^{-4} per reactor-year event impacting only one of the eight reactor modules is 1.25×10^{-5} per reactor-year. For events impacting more than one and up to all eight reactor modules, such as earthquakes, the frequency is 10^{-4} per plant-year and the consequences will take into account all eight reactor modules. By setting the lower bound of the DBE region at 10^{-4} per plant-year, PBMR is committing to designing for all events with higher frequency, whether impacting one reactor module or up to all eight.

6. **At what frequency are events sufficiently low that they are not selected as LBEs?**

   PBMR Response: BDBEs will meet the NRC Safety Goals at the prescribed distances from the plant. PBMR proposes 5×10^{-7} per plant-year, since lower frequency events by definition meet the NRC Safety Goal QHO for acute individual risk of fatality.

7. **What kinds of events and phenomena should the PRA include in the selection of a comprehensive set of event sequences?**

   PBMR Response: The PRA will be a full scope, all modes evaluation. The PRA Paper discusses this topic in greater detail.

8. **How are the deterministic DBAs of Tier 2 of the DCD derived from the LBEs, modeled, and analyzed?**

   PBMR Response: The deterministic DBAs will be derived from the DBEs by considering only the response of SSCs classified as safety-related. The consequences of deterministic DBAs will be based on mechanistic source terms and will be conservatively calculated. The upper bound consequence of each deterministic DBA will meet the 10 CFR §50.34 consequence limit at the EAB.

9. **How will uncertainties in SSC or operator performance be taken into account in the LBE selection?**

   PBMR Response: Uncertainty distributions will be evaluated for the mean (statistical) frequency and the mean consequence for each LBE. The mean frequency will be used to determine whether the event sequence family is an AOO, DBE, or BDBE. If the upper or lower bound (95% confidence) on the LBE straddles two regions, then the LBE will be compared against the consequence criteria for each region. The mean, lower, and upper bound consequences will be explicitly compared to the consequence criteria in all LBE regions.

5. PREAPPLICATION OUTCOME OBJECTIVES

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

1. The structured process for selecting LBEs using input from the PRA and supported by an integrated blend of deterministic and probabilistic elements is an acceptable approach for defining the PBMR LBEs.

2. The integrated blend of deterministic and probabilistic elements described in this paper establishes an appropriate performance-based and risk-informed approach for structuring the safety analyses that will be included in the DCA.

3. LBEs cover a comprehensive spectrum of events from normal operation to rare, off-normal events. Each LBE is defined as a family of individual event sequences where each family has a common initiating event, safety function response, and end state. This includes an appropriate definition of LBEs to support the integrated risk from a multi-module plant. There are three categories of LBEs:

   - Anticipated Operational Occurrences (AOOs) which encompass planned and anticipated events. The doses from AOOs are required to meet normal operation public dose requirements. AOOs are utilized to set operating limits for normal operation modes and states.
   - Design Basis Events (DBEs) encompass unplanned, off-normal events not expected in the plant's lifetime, but which might occur in the lifetimes of a fleet of plants. The doses from DBEs are required to meet accident public dose requirements. DBEs are the basis for the design, construction, and operation of the SSCs during accidents. Separate from the design certification, DBEs are also evaluated in developing emergency planning measures.
   - Beyond Design Basis Events (BDBEs) which are rare, off-normal events of lower frequency than DBEs. BDBEs are evaluated to ensure that they do not pose an unacceptable risk to the public. Separate from the design certification, BDBEs are also evaluated in developing emergency planning measures.

   The LBEs in all three categories will be evaluated individually to support the tasks of assessing the performance of SSCs with respect to safety functions in response to initiating events and collectively to demonstrate that the integrated risk of a multi-module plant design meets the NRC Safety Goals.

4. Acceptable limits on the event sequence consequences and the analysis basis for the LBE categories are as follows:

   - AOOs – 10 CFR Part 20: 100 mrem Total Effective Dose Equivalent (TEDE) mechanistically modeled and realistically calculated at the CAB.
   - DBEs – 10 CFR §50.34: 25 rem TEDE mechanistically modeled and conservatively calculated at the EAB.
   - BDBEs – NRC Safety Goal Quantitative Health Objectives (QHOs) mechanistically and realistically calculated at 1 mile (1.6 km) and 10 miles (16 km) from the plant.

5. The frequencies of LBEs are expressed in units of events per plant-year where a plant is defined as a collection of up to eight reactor modules having certain shared systems. The limits on the frequency ranges for the LBE categories are as follows:
   - AOOs – event sequences with mean frequencies greater than 10^{-2} per plant-year.
   - DBEs – event sequences with mean frequencies less than 10^{-2} per plant-year and greater than 10^{-4} per plant-year.
   - BDBEs – event sequences with mean frequencies less than 10^{-4} per plant-year and greater than 5×10^{-7} per plant-year.
6. The frequency below which events are not selected as LBEs is 5×10^{-7} per plant-year. The PRA examines events to 10^{-8} per plant-year to assure that there are none just below this *de minimus* frequency.
7. The kinds of events, failures, and natural phenomena that are evaluated include:
   - Multiple, dependent, and common cause failures to the extent that these contribute to LBE frequencies.
   - Events affecting more than one reactor module.
   - Internal events and internal and external plant hazards that occur in all operating and shutdown modes and potentially challenge the capability to satisfactorily retain any licensed source of radioactive material.
8. The deterministic DBAs for Chapter 15 of Tier 2 of the DCD are derived from the DBEs by assuming that only SSCs classified as safety-related are available to mitigate the consequences. The consequences of deterministic DBAs are based on mechanistic source terms and are conservatively calculated. The upper bound consequence of each deterministic DBA must meet the 10 CFR §50.34 consequence limit at the EAB.
9. Uncertainty distributions are evaluated for the mean (statistical) frequency and the mean consequence for each LBE. The mean frequency is used to determine whether the event sequence family is an AOO, DBE, or BDBE. If the upper or lower bound (95% confidence) on the LBE straddles two regions, then the LBE is compared against the consequence criteria for each region. The mean, lower, and upper bound consequences are explicitly compared to the consequence criteria in all LBE regions. The upper bound (95% confidence value) for the DBE and deterministic DBA consequences must meet the 10 CFR §50.34 dose limit at the EAB.

The process of gaining agreement on the issues is expected to involve the following steps:

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

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 address any Requests for Additional Information (RAIs) that can be addressed in the near term and identification of requested information that will be included with the DCA submittal.

Step 5 NRC issuance of a safety evaluation report on its findings related to the selection of LBEs and their intended use.

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