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08-Oct-2021

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2021-XE-NRC-013

Project No. 99902071

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

Submission of X Energy, LLC (X-energy) Xe-100 Licensing Topical Report: Transient and Safety Analysis Methodologies Framework

The purpose of this letter is to submit the subject licensing topical report (LTR) to the U.S Nuclear Regulatory Commission (NRC) on behalf of X Energy, LLC ("X-energy"). This submission describes the approach taken by X-energy to develop evaluation models and analysis methods used for transient and safety analyses for the Xe-100 reactor. It is provided for NRC review and approval as indicated in the report and is expected to be referenced in future Xe-100 licensing applications. The specific review schedule will continue to be developed with X-energy's NRC project manager; however, we request that acceptance review and schedule planning occur within 60 days of commencement and a review duration of 12 months be considered.

This report contains commercially sensitive, proprietary information and, as such, we are requesting that this information be withheld from public disclosure in accordance with 10 CFR 2.390, "Public inspections, exemptions, request for withholding," paragraph (a)(4). Enclosure 1 is the Non-Public version of the report, which contains non-redacted sensitive information. This proprietary information is appropriately marked, and the affidavit to support this request is enclosed as Enclosure 2. A redacted copy of the report that contains non-proprietary content is included as Enclosure 3.

Additionally, certain information in this report was determined to contain Export Controlled Information (ECI). This information must be protected from disclosure pursuant to 10 CFR 810. The ECI is also appropriately discussed in the affidavit in Enclosure 2, marked in Enclosure 1, and redacted in Enclosure 3.

This letter contains no commitments. If you have any questions or require additional information, please contact Ingrid Nordby at inordby@x-energy.com.

Dr. Martin van Staden
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Enclosures:

1) Xe-100 LTR: Transient and Safety Analysis Methodologies Framework (Proprietary)
2) Affidavit
3) Xe-100 LTR: Transient and Safety Analysis Methodologies Framework (Non-Proprietary)

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Enclosure 1
X Energy, LLC Xe-100 Licensing Topical Report: Transient and Safety Analysis Methodologies Framework
(Proprietary)

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Enclosure 2
Affidavit

X Energy, LLC
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Affidavit Supporting Request for Withholding from Public Disclosure (10 CFR 2.390)

I, Martin van Staden, Vice President, Xe-100 Program Manager, of X Energy, LLC (X-energy) do hereby affirm and state:

1. I am authorized to execute this affidavit on behalf of X-energy. I am further authorized to review information submitted to or discussed with the Nuclear Regulatory Commission (NRC) and apply for the withholding of information from disclosure. The purpose of this affidavit is to provide the information required by 10 CFR 2.390(b) in support of X-energy's request for proprietary treatment of certain commercial information submitted in Enclosure 1 to X-energy's letter XE00-R-R1ZZ-RDZZ-X-000714 from myself to the NRC which provides a topical report that provides an approach to safety analysis methodology for X Energy, LLC's Xe-100 reactor.

2. I have knowledge of the criteria used by X-energy in designating information as sensitive, proprietary, confidential, and export-controlled.

3. Pursuant to the provision of paragraph (b)(4) of 10 CFR 2.390, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

a. The information sought to be withheld from public disclosure in Enclosure 1 is owned by X-energy. This information was prepared with the explicit understanding that the information itself would be treated as proprietary and confidential and has been held in confidence by X-energy.

b. The information sought to be protected in Enclosure 1 is not available to the public.

c. The information contained in Enclosure 1 is of the type that is customarily held in confidence by X-energy, and there is a rational basis for doing so. The information X-energy is requesting to be withheld from public disclosure includes technical information related to the design, analysis and operations associated with our Xe-100 high-temperature, gas-cooled, pebble bed advanced reactor design that directly impact our business development and commercialization efforts. X-energy limits access to this proprietary and confidential information in order to maintain confidentiality.

d. Enclosure 1 contains information about the planned activities of X-energy related to the development of the Xe-100 design bases, forecast design development timeframes, and relate to the commercialization strategy for our Xe-100 advanced reactor. Public disclosure of the information contained in Enclosure 1 would create substantial harm to X-energy because it would reveal valuable technical information regarding X-energy's design development, competitive expectations, assumptions, current position and strategy. Its use by a competitor could substantially improve the competitor's position in the design, manufacture, licensing, construction and operation of a similar competing product.

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e. Additionally, Enclosure 1 is assessed to contain certain information that is considered Export Controlled Information (ECI) under the provisions of 10 CFR 810. I have personal knowledge of the criteria used by X-energy to evaluate documents for ECI and affirm that this information should be withheld from public disclosure.

f. The Proprietary Information contained in Enclosure 1 is transmitted to the NRC in confidence and under the provisions of 10 CFR 2.390; it is to be received in confidence by the NRC. The information is properly marked.

I declare under the penalty of perjury that the foregoing is true and correct. Executed on October 8, 2021.

Sincerely,

Dr. Martin van Staden
Vice President, Xe-100 Program Manager
X Energy, LLC

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Enclosure 3
X Energy, LLC Xe-100 Licensing Topical Report: Transient and Safety Analysis Methodologies Framework
(Non-Proprietary)

Xe-100 Licensing Topical Report

Transient and Safety Analysis Methodologies Framework

Configuration Classification	**:**	**XE00-R-R1ZZ-RDZZ-X**
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Project	**:**	**Xe-100**
Project Phase	**:**	**Preliminary**

E-SIGNATURES: DOCUMENT APPROVAL

Preparer:
Electronically signed by Raymond C Wang
29-Sep-2021 15:19

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29-Sep-2021 16:50

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Electronically signed by Martin Van Staden
29-Sep-2021 21:09

Department of Energy Acknowledgement and Disclaimer
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EXECUTIVE SUMMARY

This licensing topical report provides X Energy, LLC's (X-energy) approach to develop evaluation models and analysis methods used for transient and safety analyses for the Xe-100 reactor. The Xe-100 is a 200 MWt (80 MWe) pebble bed high temperature gas-cooled reactor design.

This report follows the evaluation model development and assessment process described in U.S. Nuclear Regulatory Commission Regulatory Guide 1.203, "Transient and Accident Analysis Methods," to prepare the Xe-100 plant evaluation model. Regulatory Guide 1.203 includes the following four elements, which are addressed within the report:

1. Establish requirements for evaluation model capability
2. Develop assessment base
3. Develop evaluation model
4. Assess evaluation model adequacy

Additionally, the approach X-energy has provided for developing the transient and safety analysis methodology and performing analyses for the Xe-100 plant includes elements of the risk-informed, performance-based methodology documented in Nuclear Energy Institute 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development." Combining elements of the risk-informed methodology with NRC's standard process, X-energy has established an approach for defining and categorizing licensing basis events, developing evaluation models, modeling the plant, and performing analyses.

In additional to evaluation model development, this report describes what kinds of codes will be used to model the Xe-100, along with the necessary phenomena to be modeled by the codes. The report also contains a list of quality assurance procedures applicable to code development and verification, as well as analysis performance and documentation. Finally, this report documents the performance of the safety analysis, using the evaluation model and code methodology described herein for the following two events:

1. Depressurized loss of forced circulation
2. Spurious withdrawal of reactivity control system control rods

It is acknowledged that this report does not contain the complete technical basis that would be expected in a full transient and safety analysis methodology report. As a starting point, this report provides the overall framework and approach taken by X-energy to develop specific analysis methodologies, code verification and validation, and other supporting bases. As documented throughout the report, several sections describe actions that will be taken by X-energy, and as that information is available, it will be provided to the NRC staff through revisions to this report, supplemental documents, and continued engagements.

X-energy is seeking U.S. Nuclear Regulatory Commission review and approval of the proposed approach to performing transient and safety analyses for the Xe-100 reactor. This information will be used as content for future safety analysis reports to fulfill the regulatory requirements for prospective Xe-100 licensing applications under 10 CFR 50, 10 CFR 52, and/or a future 10 CFR 53.

CONFIGURATION CONTROL

Document Change History

Rev.	Date	Preparer	Changes
1	September 29th, 2021	A. Spalding	For approval

Document Approval

Action	Designation	Name	Signature	Date
Preparer	Licensing Support, Westinghouse p.p. Senior Licensing Engineer	Amanda Spalding p.p. Paul Loza		
Contributor	Licensing Engineer	Raymond C. Wang		
Reviewer	Licensing Engineer	Stephen Vaughn		
Reviewer	Safety Analysis Lead	Kyle Metzroth		
Reviewer	Lead Licensing Engineer	Travis Chapman		
Approver	VP, Xe-100 Program Manager	Martin van Staden		

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ABBREVIATIONS

This list contains the abbreviations used in this document.

Abbreviation or Acronym	Definition
AGSS	Auxiliary Gas Services System
AOO	Abnormal Operational Occurrence
AR	Advanced Reactor
ARCAP	Advanced Reactor Content of Application Project
ARDC	Advanced Reactor Design Criteria
AVR	Arbeitsgemeinschaft Versuchsreaktor
BC	Boundary Conditions
BDBE	Beyond Design Basis Event
BUMS	Burnup Measurement System
CFR	Code of Federal Regulations
CI	Conventional Island
CLS	Core Loading System
COL	Combined Operating License
CP	Construction Permit
CUS	Core Unloading System
DBA	Design Basis Accident
DBE	Design Basis Event
DC	Design Certification
DCS	Distributed Control System
DID	Defense in Depth
DLOFC	Depressurized Loss of Forced Circulation
DOE	Department of Energy
EAB	Exclusion Area Boundary
EFPY	Effective Full-Power Years
EM	Evaluation Model
EMDAP	Evaluation Model Development and Assessment Process
ESP	Early Site Permit
F-C	Frequency Consequence

Abbreviation or Acronym	Definition
FHCS	Fuel Handling Control System
FHS	Fuel Handling System
FOM	Figure of Merit
FSS	Fuel Handling Support System
GDC	General Design Criteria
HPB	Helium Pressure Boundary
HTR	High Temperature Reactor
HTS	Heat Transport System
HVAC	Heating, Ventilation and Air Conditioning
IET	Integral Effects Test
IPS	Investment Protection System
IPyC	Inner Pyrolytic Carbon
ISF	Intermediate Storage Facility
LAR	License Amendment Request
LBE	Licensing Basis Event
LEU	Low Enriched Uranium
LMP	Licensing Modernization Project
LPZ	Low-Population Zone
LTR	Licensing Topical Report
LWA	Limited Work Authorization
LWR	Light-Water Reactor
MCR	Maximum Continuous Rating
MHTGR	Modular High Temperature Gas-Cooled Reactor
ML	Manufacturing License
MST	Mechanistic Source Term
MVR	Molecular Vapor Release
NEI	Nuclear Energy Institute
NEIMA	Nuclear Energy Innovation and Modernization Act
NGNP	Next Generation Nuclear Plant
NI	Nuclear Island

Abbreviation or Acronym	Definition
NRC	Nuclear Regulatory Commission
OL	Operating License
OPyC	Outer Pyrolytic Carbon
PDC	Principal Design Criteria
PIRT	Phenomena Identification and Ranking Table
PLOFC	Pressurized Loss of Forced Circulation
PRA	Probabilistic Risk Assessment
QA	Quality Assurance
QAP	Quality Assurance Plan
RB	Reactor Building
RCCS	Reactor Cavity Cooling System
RCS	Reactivity Control System
RCSS	Reactivity Control and Shutdown System
RG	Regulatory Guide
ROT	Reactor Outlet Temperature
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSF	Required Safety Function
RSS	Reserve Shutdown System
SAR	Safety Analysis Report
SARRDL	Specified Acceptable System Radionuclide Release Design Limit
SDA	Standard Design Approval
SDS	Software Design Specification
SEMP	Systems Engineering Management Plan
SET	Separate Effects Test
SFM	Safety Function Module
SFR	Sodium-Cooled Fast Reactor
SFSS	Spent Fuel Storage System
SiC	Silicon Carbide
SRNS	Sphere Recirculation System

Abbreviation or Acronym	Definition
SRP	Standard Review Plan
SRTS	Sphere Replenishment System
SSC	Structure, System, and Component
TEDE	Total Effective Dose Equivalent
TICAP	Technology-Inclusive Content of Application Project
TI-RIPB	Technology-Inclusive, Risk-Informed, and Performance-Based
TRISO	TRistructural ISOtropic
UCO	Uranium Oxy-Carbide
V&V	Verification and Validation
X-energy	X Energy, LLC

1. INTRODUCTION

1.1. PURPOSE

The purpose of this licensing topical report (LTR) is to describe the X-energy approach to develop evaluation models and analysis methods used for transient and safety analyses for the Xe-100 reactor. The evaluation models will be used to perform the various transient and accident analyses that make up the safety case for the Xe-100 reactor, which will be contained in the plant's Safety Analysis Report (SAR). The evaluation models and analysis methods provide the basis on which those analyses will be performed.

1.2. SCOPE

The evaluation models and analysis methods described herein apply to the transient and safety analysis performed to support the preparation of the SAR as defined in 10 Code of Federal Regulations (CFR) 50.34, "Contents of applications; technical information" [1]. At this stage in the Xe-100 program, this report provides a high level description of how X-energy plans to approach performing production safety analyses and how X-energy plans on addressing the methodology elements described in U.S. NRC Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods" [2]. This report does not provide the details of specific codes that will be used to perform safety analyses or how specific analyses have been categorized/performed. This information will be provided at a later time and/or in separate LTRs as the Xe-100 reactor development program progresses.

The transient and safety analysis evaluation models and analysis methods provided in this report are applicable for the Xe-100 reactor in deployments of a single-unit through multi-unit plants. X-energy intends to provide revisions to this report as the analysis methodology and code validations are developed as a means of providing the NRC staff the opportunity to perform reviews, audits, and inspections of transient and safety analysis-related activities. Once all transient and safety analysis methodology validation activities are complete, this report will form the technical basis for performing transient and safety analyses as part of the SAR for either a Construct Permit (CP) application (per 10 CFR 50.34(a)), an Operating License (OL) application (per 10 CFR 50.34(b)), or other licensing applications.

1.3. RELATIONSHIP TO OTHER DOCUMENTS

This report incorporates insights from several sources as described in the Reference section (Section 7) and throughout the report. The related mechanistic source term development approach, verification and validation of the safety analysis code suite, application of Nuclear Energy Institute (NEI) 18-04, and associated details about the qualification of TRistructural ISOtropic (TRISO) coated particle fuel as part of that validation are provided in separate LTRs. Reviewers are also advised to reference the Xe-100 Technology Description Technical Report [3] for details on the Xe-100 design and unit and plant operations. NRC staff review of that report is not requested at this time.

1.4. DOCUMENT LAYOUT

This report presents the high-level approach that X-energy is applying to develop evaluation models and perform analysis for transient and accident analyses for the Xe-100 plant for review by the NRC staff through the guidance of RG 1.203 [2]. It summarizes the relevant regulatory requirements and guidance, overall transient and accident analysis approach, Xe-100 plant modeling, and quality assurance for transient and safety analyses that will be performed to support the XE-100 plant. The layout of the report follows the elements described in RG 1.203. The methodology also relies on the technology-inclusive, risk-informed, and performance-based (TI-RIPB) processed for selection of licensing basis events (LBEs) described in NEI 18-04 [4]. Specific requests for NRC staff review and approval are provided in Section 6.

2. OVERVIEW OF REGULATORY REQUIREMENTS AND GUIDANCE

The NRC provides rules for the design, licensing, construction, operation, and decommissioning of reactors in order to provide reasonable assurance of adequate protection of public health and safety and to provide for the common defense and security. The majority of regulations associated with reactors are found in 10 CFR Parts 1-199, with a principle set of requirements found in Parts 50 and 52. The NRC also provides guidance to prospective applicants in the form of RGs that provide acceptable methods and approaches to demonstrate compliance with the regulations. Regulatory Guides may be stand-alone documents or issued as acceptance of a code, standard, or other non-NRC document as an acceptable means of demonstrating conformance. Prospective applicants are allowed to propose alternative approach to meeting regulatory requirements if appropriately justified.

The following sections provide a high-level overview of the types of documents in the U.S. regulatory framework and initial consideration for specific requirements and guidance for the Xe-100.

2.1. NRC REGULATIONS

2.1.1. 10 CFR 50

The regulatory requirements of 10 CFR 50 provide for the licensing of utilization facilities like the Xe-100 reactor plant. The principal licensing pathway used by most operating reactors is the two-step approach provided for in this part, namely the CP and OL applications. Part 50 provides requirements for processes, administration, application, review, and issuance of permits and licenses according to the NRC's mandate under the Atomic Energy Act of 1954, as amended. The regulations also provide the scope, kind, and level of maturity of information related to design and programs for preliminary and final safety analysis report contents. Part 50 maintains the core requirements for reactor licensing, is referenced in multiple sections by Part 52 licensing pathways and has captured many requirements through decades of experience and evolution in the regulatory framework of light water reactors (LWRs).

As described in SECY-15-0002 "Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications" [27], the NRC staff is undertaking rulemaking to align Parts 50 and 52 based on experience with new reactor licensing lessons learned in the 2000s and 2010s. X-energy is following this rulemaking activity to assess any impacts of the proposed rule on this report's subject matter.

2.1.1.1. 10 CFR 50.34

In 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 50.34, "Contents of Applications; Technical Information" (10 CFR 50.34) [1], specifies the requirements regarding applications for construction permits and/or licenses to operate a facility. 10 CFR 50.34(b) includes the following requirements for the final safety analysis report, which are applicable to this report:

> (b) *Final safety analysis report*. Each application for an operating license shall include a final safety analysis report. The final safety analysis report shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and shall include the following:

(2) A description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

(i) For nuclear reactors, such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent.

(4) A final analysis and evaluation of the design and performance of structures, systems, and components with the objective stated in paragraph (a)(4) of this section and taking into account any pertinent information developed since the submittal of the preliminary safety analysis report.

Transient and safety analysis methodologies are used to develop the associated information in the SAR describing the performance and safety analyses of structures, systems, and components.

2.2. NRC POLICY STATEMENTS

2.2.1. Policy Statement on the Regulation of Advanced Reactors (72 FR 60612, 2008) [5]

Reviewed and reissued in 2008, the Advanced Reactor Policy Statement provides overarching direction to advanced reactor developers like X-energy and informs the approach to regulation and addressing regulatory requirements. The final policy statement includes multiple design attributes that could assist the NRC to establish the acceptability and/or ability to license advanced reactor designs. These attributes inform the manner in which X-energy assessed the 10 CFR Part 50 technical requirements associated with design and programmatic elements of the Xe-100. An assessment of the Xe-100 with respect to each of the policy attributes is expected in future licensing application content.

2.3. NRC GUIDANCE/REFERENCES

The following guidance documents developed by the NRC staff and its contractors were used to develop the transient and accident analysis methods described in this topical report.

2.3.1. RG 1.203

RG 1.203 [2] describes a process that the NRC staff considers acceptable for use in developing and assessing evaluation models (EMs) that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.

The EM establishes the basis for methods used to analyze a particular event or class of events. This concept is described in 10 CFR 50.46 for LOCA analysis but can be generalized for all analyzed events described in the Standard Review Plan (SRP) and other regulatory guidance.

An EM is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event, as illustrated by the following examples:

1. Procedures for treating the input and output information (particularly the code input arising from the plant geometry and the assumed plant state at transient initiation).
2. Specification of those portions of the analysis not included in the computer programs for which alternative approaches are used.
3. All other information needed to specify the calculational procedure.

The entirety of an EM ultimately determines whether the results are in compliance with applicable regulations. Therefore, the development, assessment, and review processes must consider the entire EM.

To produce a viable model, certain principles should be addressed during the model development and assessment processes. Specifically, the NRC has identified the following six basic principles as important to follow in the process of developing and assessing an EM:

1. Determine requirements for the evaluation model. The purpose of this principle is to provide focus throughout the evaluation model development and assessment process (EMDAP). An important outcome should be the identification of mathematical modeling methods, components, phenomena, physical processes, and parameters needed to evaluate the event behavior relative to the figures of merit (FOM) described in the SRP and derived from the general design criteria (GDC) in Appendix A to 10 CFR Part 50. The phenomena assessment process is central to ensuring that the EM can appropriately analyze the particular event and that the validation process addresses key phenomena for that event.
2. Develop an assessment base consistent with the determined requirements. Since an EM can only approximate physical behavior for postulated events, it is important to validate the calculational devices, individually and collectively, using an appropriate assessment base. The database may consist of already existing experiments, or new experiments may be required for model assessment, depending on the results of the requirements determination.
3. Develop the evaluation model. The calculational devices needed to analyze the events in accordance with the requirements determined in the first principle should be selected or developed. To define an EM for a particular plant and event, it is also necessary to select proper code options, boundary conditions, and temporal and spatial relationships among the component devices.
4. Assess the adequacy of the evaluation model. Based on the application of the first principle, especially the phenomena importance determination, an assessment should be made regarding the inherent capability of the EM to achieve the desired results relative to the FOMs derived from the GDC. Some of this assessment is best made during the early phase of code development to minimize the need for later corrective actions. A key feature of the adequacy assessment is the

ability of the EM or its component devices to predict appropriate experimental behavior. Once again, the focus should be on the ability to predict key phenomena, as described in the first principle. To a large degree, the calculational devices use collections of models and correlations that are empirical in nature. Therefore, it is important to ensure that they are used within the range of their assessment.

5. Follow an appropriate quality assurance protocol during the EMDAP. Quality assurance standards, as required in Appendix B to 10 CFR Part 50, are a key feature of the development and assessment processes. When complex computer codes are involved, peer review by independent experts should be an integral part of the quality assurance process.
6. Provide comprehensive, accurate, up-to-date documentation. This is an expected requirement for a credible regulatory review. It is also clearly needed for the peer review described in the fifth principle. Since the development and assessment process may lead to changes in the importance determination, it is most important that documentation of this activity be developed early and kept current.

A summary of the EMDAP is shown in Figure 1.

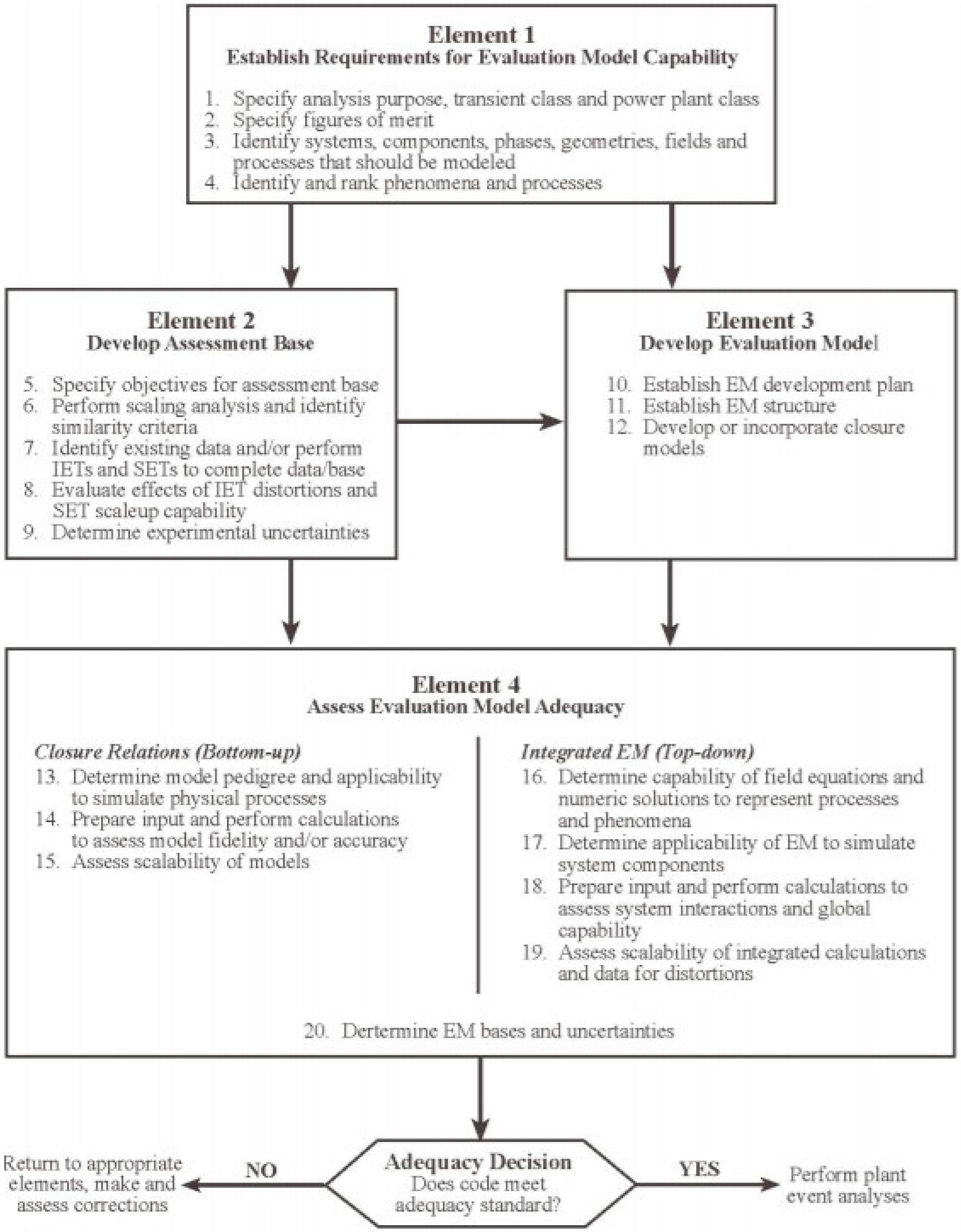

Figure 1: EMDAP [2]

This report describes the Xe-100 design's implementation of the process described in RG 1.203 for EM development.

2.3.2. RG 1.232

RG 1.232 [6] describes the NRC's proposed guidance on how the GDC in Appendix A, "General Design Criteria for Nuclear Power Plants," of 10 CFR Part 50 may be adapted for non-LWR designs. This guidance may be used by non-LWR reactor designers, applicants, and licensees to develop principal design criteria (PDC) for any non-LWR designs, as required by the applicable NRC regulations, for nuclear power plants. The RG also describes the NRC's proposed guidance for modifying and supplementing the GDC to develop PDC that address two specific non-LWR design concepts: sodium-cooled fast reactors (SFRs), and modular high temperature gas-cooled reactors (MHTGRs). Since the MHTGR design is closely aligned with the HTGR technology the Xe-100 is based upon, many of the mHTGR design criteria are relevant to the Xe-100's safety analyses.

RG 1.232 provides one approach on addressing PDC development. The ongoing Technology-Inclusive Contents of Application Project (TICAP) also provides insight into a RIPB approach to developing PDC using the guidance in NEI 18-04 [4]. The status of how PDC will be derived for the Xe-100 to determine FOMs for the safety analysis EMs is discussed within this report.

2.3.3. RG 1.233

RG 1.233 [7] provides the NRC staff's guidance on using a technology-inclusive, risk-informed, and performance-based methodology to inform the licensing basis and content of applications for non-LWRs, including, but not limited to, molten salt reactors, high-temperature gas-cooled reactors, and a variety of fast reactors. The RG is for use by non-LWR applicants applying for permits, licenses, certifications, and approvals under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

The selection of LBEs; classification and special treatments of structures, systems, and components (SSCs); and assessment of defense in depth (DID) are fundamental to the safe design of non-LWRs. These activities also support identifying the appropriate scope and depth of information non-LWR designers and applicants should provide in applications for licenses, certifications, and approvals. This RG endorses NEI 18-04, Revision 1, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development" [4] as one acceptable method for non-LWR designers to use when carrying out these activities and preparing their applications.

Regulatory Guide 1.233 endorses the guidance of NEI 18-04 as one acceptable method for informing the licensing basis and determining the appropriate scope and level of detail for parts of applications for licenses, certifications, and approvals for non-LWRs. The NRC staff had no significant exceptions to the guidance in NEI 18-04 but did provide clarifications and points of emphasis as detailed in the RG. X-energy provided responses to these clarifications in its topical report on the subject [10]. NEI 18-04 outlines an approach for use by reactor developers to select LBEs, classify SSCs, determine special treatments and programmatic controls, and assess the adequacy of a design in terms of providing layers of DID. The methodology described in NEI 18-04 and RG 1.233 also provides a general approach for identifying an appropriate scope and depth of information that applications for licenses, certifications, and approvals should provide.

As described herein, X-energy is implementing the NEI 18-04 approach and uses safety analysis methods as described herein as a means of evaluating plant performance and response to various LBEs. This is discussed further, as necessary, in this report.

2.3.4. RG 1.70

RG 1.70 [8] contains guidance on use of a standard format and content for SARs for nuclear power plants. The purpose of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants is to indicate the information to be provided in the SAR and to establish a uniform format for presenting the information. Use of this format helps ensure the completeness of the information provided, assists the NRC's staff and others in locating the information, and aids in shortening the time needed for the review process. While the guidance for SAR content has evolved further, RG 1.70 provides relevant information on the difference in scope, completeness, and level of detail between preliminary and final SAR content and was generally available to early HTGR licensees in formulating their SAR content.

X-energy's potential use of the standard format and content for the Xe-100 SAR will be discussed further in this report.

2.3.5. NUREG-2246

The NRC staff issued this draft NUREG [14] for industry feedback and public comment in June 2021 on fuel qualification for advanced reactors. The purpose of this report is to identify criteria that will be useful for advanced reactor designs through an assessment framework that would support regulatory findings associated with nuclear fuel qualification. The report begins by examining the regulatory basis and related guidance applicable to fuel qualification, noting that the role of nuclear fuel in the protection against the release of radioactivity for a nuclear facility depends heavily on the reactor design. The report considers the use of accelerated fuel qualification techniques and lead test specimen programs that may shorten the timeline for qualifying fuel for use in a nuclear reactor at the desired parameters (e.g., burnup). The assessment framework particularly emphasizes the identification of key fuel manufacturing parameters, the specification of a fuel performance envelope to inform testing requirements, the use of evaluation models in the fuel qualification process, and the assessment of the experimental data used to develop and validate evaluation models and empirical safety criteria.

X-energy has reviewed the draft NUREG for applicability and to provide insight into evaluation model development and code qualification. Further consideration is needed to determine whether there are opportunities to incorporate considerations from this report into the RG-1.203 process.

2.3.6. NRC Non-Light Water Reactor Review Strategy (Draft)

The NRC staff issued this draft white paper [28] in September 2019 for industry feedback and dialogue through the Advanced Reactor Program series of public engagements. The primary purpose of the white paper is to provide guidance to the NRC staff for review of advanced reactor licensing applications received before 2027 as a compliment to other review guidance and before the prescribed rulemaking for a technology-inclusive regulatory framework identified as 10 CFR Part 53 under the Nuclear Energy Innovation and Modernization Act (NEIMA) [40].

The white paper provides insight into the NRC staff's prioritization of technical content and high-level evaluation criteria for non-LWR technologies. The major elements of an NEI 18-04 safety design approach are provided for, as well as analysis and evaluation of integrated system design and use of risk assessment methods to inform the review scope.

X-energy reviewed the draft white paper for applicability and to provide insight into license application development. The scope, level of detail, and type of information necessary to address the technical and programmatic content expectations provided support to early annotated outline development for the Xe-100. The table of likely exemptions was also used to develop initial assessments of specific regulatory requirements.

2.3.7. NRC Analysis of Applicability of NRC Regulations for Non-Light Water Reactors

In September of 2019, and updated in September 2020, the NRC staff produced a draft white paper providing their analysis and assessment of the applicability of regulations to non-LWR technologies [30]. This effort provided a general overview of 10 CFR Part 50 and 52 regulations to facilitate "clear, open, and efficient review" of advanced reactor designs. The paper acknowledges that some regulations may be generally applicable but not be specifically necessary for non-LWR designs that incorporate unique design attributes and features. X-energy is using these insights to inform the performance of certain safety analyses that may not have relevant parallels to LWR transients and accident sequences, some of which are effectively codified in the CFR.

2.3.8. TICAP/ARCAP

Since the issuance of NEI 18-04 [4] in 2019 and its associated endorsement by the NRC staff in RG 1.233 [6], a coordinated activity to produce content of application guidance has commenced. The industry-led, U.S. Department of Energy (DOE)-funded TICAP, and NRC staff-led Advanced Reactor Content of Application Project (ARCAP) are focused on producing format, organization, scope, and level of detail guidance to support producing an advanced reactor licensing application that is based upon the NEI 18-04 licensing basis framework. X-energy participated in the tabletop demonstration of the TICAP guidance and has monitored the development of both guidance documents since mid-2020. Subsequently, NEI 21-07 [43] was issued in August 2021. While these projects continue to mature and are not yet fully endorsed, they often represent logical, technology-agnostic means of organizing the information within the SAR, and may be used by X-energy to provide a systematic means of documenting the transient and safety analyses of the Xe-100.

2.3.9. NRC Safety Review of Power Reactor Construction Permit Applications (Draft)

The NRC staff issued this draft white paper [31] in February 2021 for industry feedback and dialogue through the Advanced Reactor Program series of public engagements. It complements the draft white paper on non-LWR review strategy issued in September 2019 by focusing on the necessary information required at the preliminary safety analysis report stage of design development and licensing. As of August 2021, X-energy is working with the industry to provide observations to the NRC staff on this paper. The annotated outline development efforts for the Xe-100 are generally aligned with the scope, type, and organization of information described in the NRC's paper. Differences in expectation for level of detail will be resolved through early pre-application engagement for site-specific license applications.

2.4. U.S. HTGR PRECEDENTS

The X-energy team evaluated multiple reactor technology safety analysis reports and, when available, safety evaluation documents from domestic and international experiences. The extensive experience gained through the General Atomics' Modular HTGR review, as documented in NUREG-1338, and Next Generation Nuclear Plant (NGNP) licensing strategy, including white paper development for technical and policy considerations, provided useful insight for the Xe-100 approach to meeting regulatory requirements. Of particular interest are the phenomena identification and ranking tables (PIRT) of relevant phenomena common among HTGRs matured through the NGNP program. X-energy is undertaking an activity to screen the NGNP PIRTs to determine where further progress has been made in the state of knowledge and understanding of HTGR and TRISO fuel technology.

3. TRANSIENT AND ACCIDENT ANALYSIS APPROACH

3.1. IMPLEMENTATION OF NEI 18-04 GUIDANCE

X-energy has implemented the guidance found in NEI 18-04 [4] as the systematic, risk-informed and performance-based approach to establishing the LBEs for the Xe-100 [10]. The RIPB licensing basis approach LTR [10] summarizes how the Xe-100 approach adopts the NEI 18-04 methodology and addresses the Xe-100 approach to the NEI 18-04 clarifications established by RG 1.233. The RIPB LTR includes information describing how X-energy has:

- Defined the LBEs that will be used in safety analysis reports required for licensing applications in 10 CFR 50.34 (a)(4) and 10 CFR 50.34(b)(4) to meet the dose requirements of 10 CFR 50.34(a)(1)(D) and 10 CFR 50.34(b)(1) and the quantitative safety goals set by the NRC.
- Defined the safety classification of SSCs.
- Evaluated the adequacy of defense in depth provided by the plant capabilities and programmatic controls.

The details of the X-energy approach and methodology for developing a RIPB licensing basis are not replicated herein. The process described in [10] is being followed to develop the Probabilistic Risk Assessment (PRA) event sequences, select and categorize LBEs, define Required Safety Functions (RSFs), and classify SSCs. This information will inform the safety analyses and the layout and content in the SAR. Additionally, the use of the NEI 21-07 and NEI 18-04 guidance will support the development of PDC for the Xe-100, rather than relying only upon the Advanced Reactor Design Criteria (ARDC) defined in RG 1.232 [6]. X-energy will continue ongoing engagement with the NRC staff to discuss the proposed approach as PDC development progresses. This decision will have an impact on the documentation of the safety analyses but should not make a material difference in the development of EMs and the performance of the safety analyses.

3.2. DEVELOPING EVALUATION MODELS

3.2.1. RG 1.203 EMDAP Elements

The following subsections describe how X-energy has addressed (or plans to address) elements described in NRC RG 1.203 [2] for EMDAP for the Xe-100 safety analyses. A high-level summary of how EMDAP is being addressed for Xe-100 is provided in Figure 2. This section also documents further development to be completed or that will be documented in other LTRs.

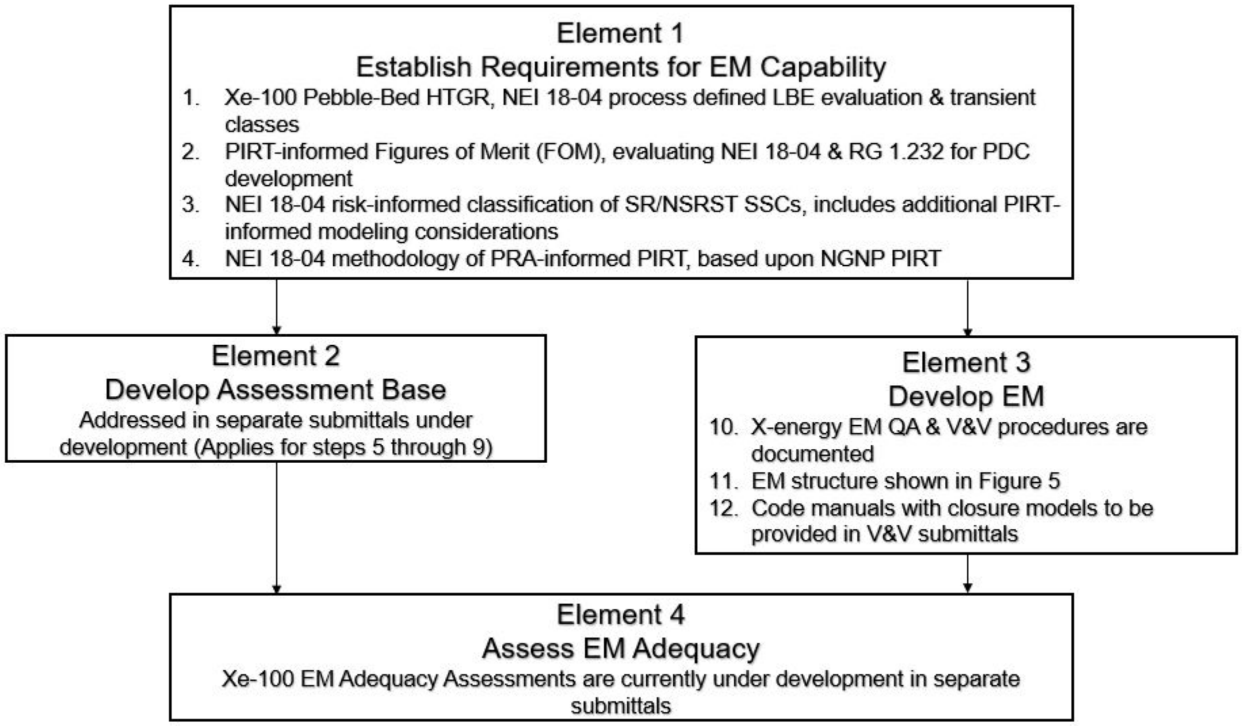

Figure 2: Xe-100 Specific EMDAP

3.2.1.1. Element 1: Establish Requirements for Evaluation Model Capability

Step 1: Specify analysis purpose, transient class, and power plant class

The purpose of the Xe-100 transient and safety evaluation model is to define the process by which the purpose of each specific transient analysis for the Xe-100 reactor is derived, and to comply with the regulations for this reactor (see Section 2 for specific regulatory requirements). The analysis of events and event classes will be determined through the Xe-100 PRA using the NEI 18-04 [4] methodology. This step will follow the process described in the Xe-100 RIPB LTR [10]. The following short description of the Xe-100 provides a high-level overview of the plant design for which the EM is applicable.

The Xe-100 is a Generation IV Advanced Reactor (AR) based on pebble bed HTGR technology, which utilizes U.S.-developed Uranium Oxy-Carbide (UCO) TRISO-coated fuel embedded in spherical fuel elements to form fuel pebbles. The reactor generates 200 MWt and produces high-quality, super-heated steam at 565°C and 16.5 MPa, which is suitable for many different energy applications.

The standard Xe-100 plant consists of four identical units, each containing a nuclear reactor, a steam generator, and a steam turbine generator. The nuclear steam supply system and safety-related SSCs are contained in the Nuclear Island (NI), while the steam turbine generator and all its associated systems are contained in the Conventional Island (CI). The Xe-100 plant is designed such that the CI systems (secondary or non-nuclear side) are commercially available or off-the-shelf and can be optimally configured to suit the end user's requirements. The NI, on the other hand, is the standardized part of the design and remains unchanged irrespective of the application.

A single 200 MWt reactor module configured for electricity generation produces approximately 80 MWe net power. The Xe-100 is highly maneuverable, it can ramp up and down at ~5% per minute within the 100% - 40% - 100% power range. The reactor can, however, operate at stable power levels as low as 25% but with limited maneuverability. The plant is designed to handle a trip from full power to house load conditions without causing a reactor trip. All four units of the Xe-100 plant are controlled from a single control room resulting in staffing efficiencies and potentially reduced operating costs. The Xe-100 analysis capabilities accommodate design transient analyses of such maneuvering as well as safety analyses that evaluate plant (individual units and 4-unit plant) response to various LBEs.

Step 2: Specify figures of merit

X-energy continues to assess both the NEI 18-04 [4] methodology and RG 1.232 [6] as mutually supportive means of deriving the Xe-100 PDC and associated FOMs for safety analyses. For example, RG 1.232 states the following for Abnormal Operational Occurrences (AOOs) related to ARDC Criterion 10:

> The concept of specified acceptable fuel design limits, which prevent additional fuel failures during AOOs has been replaced with that of the specified acceptable system radionuclide release design limits (SARRDLs), which limits the amount of radionuclide inventory that is released by the system under normal and AOO conditions. The term “system” refers to the fuel, the helium coolant circuit and all connected systems that are not isolated and may contribute to dose.
>
> Design features within the reactor system must ensure that the SARRDLs are not exceeded during normal operations and AOOs…
>
> The SARRDLs will be established so that the most limited LBE does not exceed the siting regulation dose limits criteria at the exclusion area boundary (EAB) and low-population zone (LPZ), and also so that the 10 CFR 20.1301 annualized dose limits to the public are not exceeded at the EAB for normal operation and AOOs.

At a high level, there will be FOMs for AOOs for the following:

- Circulating activity within helium pressure boundary (HPB)
- Plateout activity within HPB
- Delayed activity release due to AOOs

Additionally, per RG 1.233 [7], Design Basis Accidents (DBAs), which rely on the safety-related (SR) SSCs, should be performed using deterministic methods. The DBAs will use deterministic FOMs. For example, RG 1.233 states: “there should be no DBAs with significant releases from two or more modules or radionuclide sources.”

The development of event-specific FOMs is still in progress. X-energy will provide the NRC staff with additional details in future submissions and engagements.

Step 3: Identify systems, components, phases, geometries, fields, and processes that must be modeled

Information for some of the systems, subsystems, and components required for Step 3 and utilized in the Appendices B and C analyses are included below for the Xe-100 reactor. At this stage of Xe-100 development, the majority of SSCs that require modeling have been identified at the functional level and are being more fully developed through the preliminary design. The analyses of Appendix B and C do not provide an example the full scope that is required for RG 1.203 Step 3; X-energy will provide updated information in future submittals. The full scope of safety-related SSCs will be determined through the NEI 18-04 methodology [4], and a preliminary safety classification of all SSCs was performed to inform early safety analysis activities (some of which were performed to further evaluate the impact of safety-related classification determinations). As the Xe-100 PRA and safety analysis results become more mature for the Xe-100 reactor, the safety-classification of specific SSCs will become further defined.

Reactor Protection System (RPS) Action Assumptions

[[

]]P,E

The other functions are somewhat unique to HTGR technology. Stopping forced circulation provides a means of shutting down the nuclear reaction due to increasing core temperature, as well as protecting the pebble bed from any contaminants introduced into the system in the event of a HPB breech. Isolating the secondary system by closing the steam and feedwater isolation valves, and initiating a steam generator relief to allow egress of steam/water in the system, controls the degree of ingress of these fluids into the primary system in the event of a steam generator tube leak or rupture.

Distributed Control System

For DBAs, only the Safety-Related SSCs are credited with providing mitigation, unless crediting the action of other systems would make the consequences worse.

Reactor Cavity Cooling System (RCCS)

The RCCS provides cooling of the reactor building concrete during normal operation and contributes to the heat sink capability for accident conditions where forced convection heat removal from the fuel is lost. [[
]]P,E

Step 4: Identify and Rank Key Phenomena and Processes

A preliminary phenomena identification and ranking table (PIRT) has been compiled for Xe-100 [23]. However, at this stage in the Xe-100 development program, the PRA and safety analysis for the Xe-100 conceptual design were still in early stage development and therefore the progression of events for target accidents in the Xe-100 reactor and specific FOMs were not finalized. The information from [23] is constituted from existing PIRTs for NGNPs, which were modified based on knowledge of the Xe-100 conceptual design. Further maturation of the Xe-100 PIRT is expected to occur through the preliminary design phase and be provided in future submittals.

The existing PIRTs used as a basis for the Xe-100 are acknowledged to be generic in nature and therefore, all listed phenomena may not be fully applicable to the Xe-100 reactor. A process, described herein, was followed through which phenomena specific to the Xe-100 design were identified.

The following steps were executed to prepare the initial Xe-100 PIRT:

1. The events to be analyzed were identified.
2. A literature search was carried out to identify potential sources of PIRTs that may be relevant for the Xe-100.
3. The PIRTs identified in Step 2 were assessed.
4. Relevant information (for FOMs and SSCs related to Xe-100) was extracted and collated.
5. Information not explicitly relevant to the Xe-100 reactor, such as phenomena specific to prismatic fuel, were not included.
6. Phenomena identification numbers were updated to reflect areas of relevance.
7. The information from open literature sources is presented as found and no editorial revisions were executed. Original notes and references are maintained.

The generic PIRTs were used to create a Xe-100 specific PIRT for normal operating conditions and the initial Xe-100 deterministic safety analysis cases. The phenomena included in the Xe-100 PIRT:

- Were of High or Medium rank
- Occurred in components simulated during initial Xe-100 deterministic safety analysis cases

Moving forward, using the NEI 18-04 methodology and NEI 21-07 guidance, this step will be dictated by the Xe-100 PRA. The PRA will inform which systems are safety-significant and inform the definition of the FOMs. Maturation of the Xe-100 PRA continue to revise the PIRT and provide input to the activities for Step 1 of EMDAP methodology for safety analysis to evaluate.

3.2.1.2. Element 2: Develop Assessment Base

The information for the steps in Element 2 is maturing, based on past HTGR technology examples and current state of the field. This information will be captured in separate Verification & Validation (V&V) documents for the codes that will make up the Xe-100 evaluation model. These individual documents will be submitted to the NRC staff for review when they are available.

Step 5: Specify objectives for assessment base

Step 6: Perform scaling analysis and identify similarity criteria

Step 7: Identify existing data and/or perform integral effects tests (IETs) and separate effects tests (SETs) to complete the database

Step 8: Evaluate effects of IET distortions and SET scaleup capability

Step 9: Determine experimental uncertainties as appropriate

Since specific code selections for certain event sequence evaluation models are still under evaluation (e.g., in many cases, several combinations of codes could be used to model a given event; selection for production runs will be made to support SAR development), it is too early to fully describe each of these steps. The work of past HTGR technologies and licensing precedents continue to form a basis for establishing the V&V plans and methods. X-energy evaluates the effectiveness of each method and improvements to model use where the start of the art has matured. For instance, the Xe-100 benefits from extensive use of high-fidelity computational fluid dynamics (CFD) to model thermal phenomena across the plant as the methods and computational resources for this tool are much improved today compared to past HTGR development.

3.2.1.3. Element 3: Develop Evaluation Model

Step 10: Establish an evaluation model development plan

See Section 5 for a list and description of quality assurance (QA) and software V&V procedures applicable to the development and performance of the safety analysis evaluation model, including procedures for documentation requirements and configuration control.

Step 11: Establish evaluation model structure

An example Xe-100 EM high level analysis flow chart is documented in Figure 5. Code development and phenomena modeled for the codes that make up the Xe-100 evaluation model is discussed in Section 4.

Additionally, the plant models can be found in the basedecks for the codes that will make up the Xe-100 evaluation model. These basedecks and modeling details will be made available for NRC review once finalized.

Step 12: Develop or incorporate closure models

A full description of the closure models and the associated equations used in the Xe-100 evaluation model will be provided in code user manuals. X-energy is in the process of finalizing code evaluation and development and will provide users manuals for all codes utilized for safety analysis for NRC review.

3.2.1.4. Element 4: Assess Evaluation Model Adequacy

Detailed information for Steps 13-20 has not been fully developed for the Xe-100 evaluation model during the Conceptual Design phase. As evaluation models mature and this information becomes available through the remainder of Preliminary Design, X-energy will provide for NRC review.

Step 13: Determine model pedigree and applicability to simulate physical processes

Step 14: Prepare input and perform calculations to assess model fidelity or accuracy

Step 15: Assess scalability of models

Step 16: Determine capability of field equations to represent processes and phenomena and the ability of numeric solutions to approximate equation set

Step 17: Determine applicability of evaluation model to simulate system components

Step 18: Prepare input and perform calculations to assess system interactions and global capability

Step 19: Assess scalability of integrated calculations and data for distortions

Step 20: Determine evaluation model biases and uncertainties

3.3. DEMONSTRATION SAFETY ANALYSIS INITIAL CONDITIONS

The steady-state normal operating conditions that serve as initial conditions for the safety cases enclosed in Appendices B and C are calculated through computer codes that perform the following functions:

- Steady-state pre-accident neutron flux, power and temperature distributions
- Steady-state mass flow rate and heat balance across cores
- Steady-state isotope inventory, fission product release from fuel, radioactivity in helium heat transport system (HTS), leakage through HPB under normal steady-state conditions

The initial steady-state conditions for safety and transient analyses are captured in Table 1.

Table 1. Initial Steady-State Conditions. From Ref. [11].

Parameter	Value
Core Thermal Power	200 MW(th) (Footnote 1)
Core geometry	As in conceptual design
Average fuel and moderator temperature	[[ ]]^{P,E}
Core average power density	[[ ]]^{P}
Xenon load	[[ ]]^{P,E}
RCS rod insertion	[[ ]]^{P,E} (Footnote 2)
Reactivity worth of rod withdrawal	[[ ]]^{P,E}
RSS shutdown rod position	Withdrawn and poised
Total worth of RSS and RCS rod insertion	[[ ]]^{P,E}
Fuel U-235 Enrichment	15.5 wt%
Fuel average discharge burnup	[[ ]]^{P}
Pebble-average maximum fuel temperature	[[ ]]^{P,E}
He mass flow rate in pebble bed core	[[ ]]^{P,E}
He inlet temperature	[[ ]]^{P}
He outlet temperature	750°C (Footnote 1)
He primary circuit average pressure	6.0 Mpa
Main Steam temperature	565°C (Footnote 1)
Main Steam pressure	16.5 MPa (Footnote 1)
Decay heat	DIN 25485 standard
Material properties	30 years of fluence (Footnote 3)
Initial failed fuel fraction	[[ ]]^{P,E}

The initial conditions are as follows:

- For Appendices B and C cases, the initial condition is a core operating at 100% maximum continuous rating (MCR), which is 200 MWth. Analyses at part and 0% MCR are planned for future development pending revised PIRT.
- The [[ ]]^{P,E} control rods under Reactivity Control System (RCS) control are used to maintain the core critical. The rod insertion (expressed in terms of the position of the control rod lower tip relative to the bottom of the top reflector) is [[ ]]^{P}.

[1] Core thermal power, reactor outlet temperature, main steam temperature and main steam pressure are control parameters in the Distributed Control System (DCS), and the values here are their preliminary control setpoints.

[2] Distance from bottom of top reflector to bottom tip of control rods [[ ]]^{P}.

[3] Refers to graphite reflector properties.

- The core region contains TRISO fuel pebbles in 6 passes with uranium burnup ranging from 0 (just injected into the stream) to [[ ]]^{P,E} on discharge at the end of the [[ ]]^{P,E}.
- The neutronics design parameters are developed using Xe-100 Core Design methodology and reported in the Core Design Report [12].
- The steady-state mass and heat balance are developed by Xe-100 Design and reported in the Xe-100 Heat and Mass Balance Report, pending design finalization [13].
- The Distributed Control System (DCS) is assumed to be functioning normally prior to an accident, such that the core thermal power, reactor outlet temperature, main steam temperature and main steam pressure are at their control setpoints as listed in Table 1.

3.4. STATES FOR SAFETY ANALYSES

The examples of Xe-100 safety analysis methodology in this report's appendices were focused on a full power nominal operating state. The reactor is assumed to be generating 200 MWth and 80 MWe. Future work will include finalizing the methodology for running cases from lower initial power as necessary to demonstrate that the analysis covers all operating modes and states and considering uncertainties in the key parameters that may affect the analysis outcome, including but not limited to uncertainties in initial reactor power and variations in the initial values of plant operating parameters.

Analysis based on nominal operating conditions is important to provide an understanding of the expected plant response to AOOs, DBEs, DBAs, and Beyond Design Basis Events (BDBEs). Analysis from nominal operating conditions provides a baseline for understanding the impacts of uncertainties and variations in initial parameter values. For the included examples, the analyses provided helpful insights into the performance of SSCs in fulfilling the Xe-100's required safety function and their candidacy as safety-related SSCs. While the event sequences were highly deterministic (assuming multiple failures in both cases), the analyses fulfilled their purpose for preliminary SSC performance evaluation.

The current design of operating modes and states for the Xe-100 are shown in Figure 3. The operating modes and states have matured through the Conceptual Design phase and will be finalized in Preliminary Design. For the example analyses in Appendix B and C, the plant is operating in Mode 5. The figure shows the transition from a cold defueled state (1a), through startup and power raise, to full power operation (5c). It is noted that the shutdown action of the RPS is not credited in the example cases.

Figure 3: [[ ]]P,E

3.5. PLANT RESPONSE ACCEPTANCE CRITERIA

NRC RG 1.232 documents ARDC that could be used as the acceptance criteria for the Xe-100. Additionally, the X-energy RIPB LTR [10] documents how X-energy plans to use the Frequency-Consequence targets contained in NEI 18-04 [4] to categorize LBEs. This information can be used as a tool to inform the development of PDC and/or acceptance criteria for performing Xe-100 safety analysis. The development of PDC and acceptance criteria for the Xe-100 at the preliminary design stage are still in progress. The use of the NEI 21-07 and NEI 18-04 guidance will support the development of PDC for the Xe-100, as opposed to only using the Advanced Reactor Design Criteria (ARDC) defined in RG 1.232 [6]. X-energy will continue engagement with the NRC staff through this process.

3.6. EXECUTION AND EVALUATION OF AOOS, DBES, BDBES ANALYSES

Per the X-energy RIPB LTR [10], the NEI 18-04 methodology and NEI 21-07-informed processes will be used for the Xe-100 design and licensing bases. These methods do not require deterministic analyses to be performed for AOOs, Design Basis Events (DBEs), and Beyond Design Basis Events (BDBEs).

3.7. EXECUTION AND EVALUATION OF DBAS AND EXTERNAL HAZARDS ANALYSES

Per the X-energy RIPB LTR [10], the NEI 18-04 methodology and NEI 21-07 informed processes will be used for the Xe-100 design and licensing bases. These methods require that specifically-defined deterministic analyses be performed for DBAs. For the analyses identified as DBAs using NEI 18-04, the evaluation model and methods described in this report will be used.

External hazards will also be evaluated in the PRA. The same NEI 18-04 methodology will be applied to external hazards analyses. Any LBEs based on external hazards that rise to the level of DBA as defined in NEI 18-04 will also be run through the safety and transient analysis evaluation model described herein.

3.8. IDENTIFICATION OF ANALYSES REQUIRING SOURCE TERM ANALYSIS

X-energy is currently preparing a LTR to NRC describing the approach for performing mechanistic source term (MST) analysis for the Xe-100 [24]. This report summarizes:

> The Xe-100 design expects relatively minor post-accident dose consequences due to the robust TRISO fuel design. X-energy has developed a MST code to mechanistically model the transport of radionuclides that comprise the source term, from their birth in fuel to their potential release to, and transport in, the environment. These small source terms enable the small plant footprint and reduced accident radionuclide releases. The Licensing Modernization Project (LMP) [25] developed technology-inclusive, risk-informed, and performance-based non-LWR licensing methods applicable to the Xe-100. This licensing topical report on MST is one of several reports to be discussed with the NRC staff to demonstrate how the design will accomplish reactivity and power control, heat removal, and radioactive material retention. Xe-100 design licensing is aligned with RG 1.233 [7] methodology resulting from the LMP to select LBEs, classify SSCs, assess the adequacy of a design in terms of providing layers of DID, identify appropriate programmatic controls, and help determine the appropriate scope and level of detail. Upon defining the plant potential failure modes, the plant responses to these failures, and mitigating protective strategies, a design specific PRA is then used to confirm this selection of events, safety functions, key SSCs, and adequacy of DID. This PRA work is being conducted using ASME/ANS RA-S-1.4-2021 Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants [RA-S-1.4-2021] [26].

NEI 18-04 [4] contains the following discussions on how mechanistic source term analysis fits into the overall analysis process:

> The evaluation of the consequences of all LBEs are supported by mechanistic source terms.
>
> Thus, the LBEs can inform the determination of the limiting source terms and potential releases to be considered for operational radiation protection in normal operations as well as AOOs and DBEs that can then be used to identify design-specific shielding, filtering capability of the heating, ventilation, and air conditioning system, monitoring, and other requirements for different types of non-LWRs.

As described in the RIPB LTR [10], the Xe-100 will implement the NEI 18-04 methodology and MST into its evaluation model development and analyses.

3.9. IDENTIFICATION OF ANALYSES REQUIRING SPECIAL METHODS

Analyses will follow the same development methodology as described within this report. If and when any analysis is identified that requires special methods, those methods will be submitted to NRC for approval.

3.10. CONTROL SYSTEM RESPONSE

Specific assumptions of the Distributed Control System are provided in the example analyses in this report's appendices.

3.11. TREATMENT OF NSRST AND NST SSCS

The X-energy RIPB LTR states that NEI 18-04 guidance will be adhered to for categorization and treatment of NSRST and NST SSCs [10]. More detail on specific assumptions made for NSRST and NST SSC operations in LBE analyses will be provided in each evaluation model.

3.11.1. Boundary Conditions and Interfaces

Example boundary conditions and interfaces for evaluation model development are show in Figure 4.

Figure 4: [[ ]]P,E

In the safety analysis, a variety of systems, processes, and codes form boundary conditions (BC) and interfaces with the EM of interest. The final set of BC and interfaces for each EM will be outlined in future submittals to the NRC for review.

For system analyses, an example of BCs is shown in Figure 4, which documents a process flow diagram of Xe-100 showing the primary and secondary side. This figure illustrates the boundaries and interfaces for the various SSCs throughout the plant, as well as key parameter values. This information is to be used as input for the suite of codes that will make up the Xe-100 EM, to ensure that the codes all work together to accurately and seamlessly model the plant and perform its transient and safety analyses.

Note, that certain BCs are undergoing further refinement, thus final values are expected to deviate from the preliminary information presented here. The purpose of Figure 4 is merely to provide examples of candidate BCs and interfaces for the Xe-100 plant. This explains, for example, why some of the parameter values in Figure 4 differ slightly from the values in Table 1.

3.12. NORMAL OPERATIONS AND EXAMPLE ANALYSES

A description of how normal operations are modeled for the Xe-100 is contained in Appendix A. This appendix includes information on pre-equilibrium core and initial start-up core, internal and external hazards, and fuel handling. Additionally, examples of two BDBEs are contained in the Appendices B-C of this report to demonstrate X-energy's execution of the EMDAP. See Appendix B for the depressurized loss of forced circulation (DLOFC) event and Appendix C for spurious withdrawal of reactivity control system control rods. The appendices both contain the preliminary event sequence description, methodology and assumptions, results, and conclusions. In both cases, the analyses were conducted using mature conceptual design information from the Xe-100 to exercise safety analysis capability and perform preliminary assessments of SSC performance.

3.13. DOCUMENTATION AND REQUIREMENTS

With input from the NEI 18-04 methodology on classification of LBEs, those that are defined as DBAs will be run through the safety analysis methodology and EM discussed herein. The analyses will be required to meet the previously defined acceptance criteria for each analysis, as defined based on the PDC determined to be applicable for the Xe-100 reactor.

The results of these analysis will be documented in the SAR for NRC review. X-energy currently expects to use NEI 21-07[43] guidance for the format and content for SAR Chapters 1-8. However, that decision is not yet final, and X-energy will continue to keep NRC informed of any plans related to the Xe-100 SAR.

As decisions are made and the overall safety analysis process is more defined for the Xe-100, X-energy will continue communications with NRC. As needed, this report will be amended and resubmitted, or the information herein will be supplemented with other LTR submittals.

4. XE-100 PLANT MODELING

4.1. XE-100 CODE DEVELOPMENT FOR TRANSIENT AND SAFETY ANALYSES

4.1.1. Mechanistic Source Term Code

A code will be used to perform fully integrated MST calculations. Its calculational scope ranges from steady-state and transient reactor core simulation to modeling of fuel performance (i.e., probability of fuel particle failure), fission product inventory and transport of radionuclides from the point of their origin through all relevant pathways up to the dispersion in the environment and dose consequence at the site boundary. A separate MST code LTR will be submitted to NRC [24], which reviews the physical phenomena that leads to such releases, gives an overview of pertinent experimental data, and outlines an approach to V&V of the code for use in event-specific source term development and subsequent transient analyses. The approach to functional containment in the design of the Xe-100 will be discussed along with functional containment performance criteria to be assessed in future analyses.

4.1.2. Neutronics Code

Steady-state conditions during normal operation are calculated with a neutronics computer code, which includes comprehensive numerical simulation of the multi-scale, multi-physical neutronics/thermodynamic phenomena of high temperature gas-cooled reactors in a thermal neutron spectrum. It implies setting up the reactor and its fuel elements, processing of the relevant cross-sections, evaluating the neutron spectrum, and performing the multi-group neutron diffusion calculation in two or three dimensions, while fuel burnup, fuel shuffling, reactor control, thermal hydraulics and fuel cycle costs are continuously performed as per input definition. The thermal hydraulics part (steady state and time-dependent) is restricted to high-temperature reactors (HTRs) and in two dimensions. The code can simulate all phases of reactor operation from the initial towards the equilibrium phase.

The neutronics code system includes internal modules for defining the fuel to be employed, for plant geometry and performs online cross-section group collapsing for few-group neutron diffusion analysis. However, a temperature-dependent resonance integral evaluation for different resonance absorber concentrations related to the fuel considered is performed with an external module of the neutronics code system prior to the analysis.

4.1.3. Thermal-Hydraulics Analysis Code

A system level thermal-hydraulics analysis code will be used for modeling all components of the reactor including pumps, heat exchangers and pebble bed reactor core. The kinetics and temperature modeling in the code applicable to reactor physics are described here:

- The pebble bed core is modeled as a porous medium discretized into nodes, based on discretization of the core into axial zones and radial zones (including reactor, reactor cavity and reflector).

- Heat transport conditions [[ ]]P,E.
- Heat transfer from the fuel to the helium and to the graphite reflector is modeled.
- The temperature profile for the fuel pebbles [[ ]]P,E.
- Time variation is simulated with a point kinetics model [[ ]]P,E.
- The power and temperature calculations are coupled, an essential aspect of modeling a core with strong negative temperature reactivity feedback.

4.2. XE-100 ANALYSIS FLOW CHART

The flowchart in Figure 5 documents an example high-level structure for performing safety and transient analyses for the Xe-100 reactor. The figure shows the inputs to the various codes, the order in which the types of codes are run, how the codes feed information into one another, and the final outputs that are generated. The analysis is an iterative process given that the dose consequence information can be fed back into the RIPB NEI 18-04 methodology to refine the analysis.

Figure 5: [[ ]]P,E

4.3. PHENOMENA MODELED

4.3.1. Thermal-Hydraulics

Thermal-hydraulics and plant response for the Xe-100 is modeled using a code that solves the cross sectional averaged (1-D) conservation of mass, momentum and energy equations using the finite volume method, for steady state and transient calculations. An exception to the 1-D representation is the pebble bed reactor core. This code contains a materials property database defining temperature and pressure dependent properties for substances, including Helium and water. Additionally, it has numerical models for specific components of a reactor including pumps, heat exchangers, and the pebble bed reactor core. The purpose of the model is to capture the integrated simulation of the reactor together with the rest of the main power system.

The model is used to predict the overall system response of the Xe-100. This code is also used to provide thermal-hydraulic boundary conditions to other codes.

4.3.2. Neutronics/Core Kinetics

4.3.2.1. Core Neutronics

The design of reactor lifetime follow requires information about many different physical events which are mutually coupled to one another.

The coupled neutron physics and thermo-fluid dynamics design of the Xe-100 reactor includes the following special characteristics:

1. The combined effect of a strong negative temperature coefficient of reactivity over the entire range of operations in the Xe-100 [[
   ]]^{P,E} ensures the reactor will automatically shut itself down in any event that leads to a temperature excursion.

   This characteristic is typical of a graphite moderated reactor system with a uranium fuel cycle, whilst the core power density results from the matching choice of decay heat production and heat flux capability of the helium pressure boundary, especially close to the core cavity.

2. The RCSS rods are traveling in channels situated in the side reflector allowing them to freely move for purposes of reactivity control.

   This characteristic allows the core to shutdown solely with the RCSS, when inserted into the side reflector region in channels provided for this purpose. [[
   ]]^{P,E}

3. A functional design capable of removing residual heat solely by thermal conduction, thermal radiation and natural convection. The maximum fuel temperature anywhere in the core remains

below [[ ]]^{P,E} during normal operation. This limit is expected to be met via deliberate design decisions based on the coupled neutronics and thermo-fluid dynamics behavior within the bounds of the given geometry and suitable selection of materials.

[[

]]^{P,E}

4.3.2.2. Temperature Reactivity Coefficients [11]

The reactivity coefficients vary with operating condition (first core to equilibrium core, xenon inventory, and temperature level in core). However, the fuel and moderator temperature coefficients remain negative over all operating conditions, whereas the reflector temperature coefficient is positive and of smaller magnitude, so that the overall temperature reactivity coefficient is negative. Component coefficients are discussed below.

Fuel Temperature Reactivity Coefficient

The fuel temperature coefficient indicates the direct influence of a temperature change in the materials of fuel pebbles on the value of k-eff. An increase in movement of the atoms when heated is manifested in a steady state configuration by a lowering and broadening of the absorption resonances of the materials (so-called Doppler broadening of the resonances). For extremely diluted materials, a change in the rate of absorption would have little effect. If however, the materials are strongly concentrated and organized heterogeneously (as is the case in the kernels of coated particles and in the coated particles of fuel elements), then a change is observed in its self-shielding and the rate of absorption accordingly. This is called the Doppler coefficient. It provides a prompt response, i.e., without any time delay.

Due to its importance in most reactors, the Doppler coefficient is usually calculated for the resonances of U-238. Absorption resonances are particularly marked in the case of U-238. The energy region between 4 and 4000 eV contains 416 resonances. Out of a continuous neutron energy spectrum, the neutrons with energies near the resonances are selectively absorbed. Other nuclides also with resonances in the epithermal energy range (U-235, Pu-239, Pu-240, Pu-242, and others) have been incorporated in the nuclear libraries as extremely diluted resonance integrals (rather than explicitly calculating self-shielding). This treatment is considered to be acceptable when the concentration of the other nuclides is lower than that of U-238 by a factor of up to 3.

In the thermal energy range (< 2.05 eV) the resonances of Pu-239, Pu-240, and other nuclides cover multiple energy groups in the 96-group library. These are explicitly calculated, and a temperature dependent broadening cannot be directly simulated. However, the temperature-dependent shift of the thermal energy spectrum, as contained in the moderator temperature coefficient, outweighs the effect of resonance broadening by a substantial margin.

Moderator Temperature Reactivity Coefficient

In the thermal energy range (< 2.05 eV) the neutrons are in thermal equilibrium with the temperature movement of the moderator atoms. A change in the moderator temperature causes a change in the neutron energy distribution (neutron spectrum). Therefore, all nuclides with an energy dependent rate of absorption will be modified.

This effect is especially strong when the maximum value of the neutron flux is shifted into a specific energy range, where the $\sigma_a(E)$ of particular nuclides have strong gradients. This is true in particular for U-235, for certain Pu isotopes with their thermal resonances, and for Xe-135. In this instance, the fuel and moderator temperature coefficients are interdependent.

A change in the power production of the fuel primarily causes a change in the fuel temperature. This is propagated in the moderator with a short time delay and changes the neutron spectrum. Accordingly, the moderator temperature coefficient is somewhat delayed in time. However, if the initiating temperature change occurs within the moderator (e.g., due to a change in the thermal removal process) then the coefficient has a prompt response.

Furthermore, the graphite moderator has a very low coefficient of expansion, so heat-up of the graphite has little effect on neutron absorption (other than that due to the spectral shift in any $\frac{1}{v}$ absorber). By comparison, in water-cooled or water-moderated reactors, heat-up will lead to expansion of the water, which can be source of positive reactivity because neutron absorption in H_2O and in soluble poison (if present) is reduced, while neutron moderation may not be affected if the lattice is over-moderated.

Reflector Temperature Reactivity Coefficient

In the reflector this coupling also exists between the moderator temperature and the neutron spectrum. As the temperature increases:

- Neutron absorption is reduced in the graphite and control rod poison. Subsequently, more neutrons are flowing back into the core, thereby reducing the leakage.
- Graphite starts acting like a free-gas neutron scatterer, so graphite becomes a less effective thermalizing agent.
- The ratio of the capture-to-scattering decreases.

The net effect is a positive impact on the neutron balance (i.e., an increasing k-eff).

The reflector coefficient is strongly delayed in time since temperature changes in the core slowly propagate in the reflector.

Total Temperature Reactivity

For the reasons noted above, the fuel and moderator temperature reactivity coefficients are large, negative and prompt for all operating conditions, while the reflector temperature reactivity coefficient is positive but of smaller magnitude and delayed, relative to the first two components. This ensures a prompt negative temperature feedback that provides an inherent power-limiting mechanism in reaction to a temperature or power increase.

4.3.3. Heat Management [15]

The thermal conductivity of graphite as function of fast neutron dose and temperature is a significant factor in calculating the temperature distribution. The fuel element graphite data is taken from the work of Binkele [35] up to a fast dose of approximately 6.09 x 10^{21} cm^{-2} EDN (approximately 2.34 x 10^{21} cm^{-2} EDN is experienced during Xe-100 operation) and a temperature up to 1000°C (which is the range covered by the experimental work). [[

]]P,E

In the Xe-100 EM, [[

]]P,E

Similarly to the thermal conductivity in the thermal-hydraulics formulation, the functions of heat capacity are coupled to the neutronics part of the code.

4.3.3.1. Decay Heat

Per [15], after reactor shutdown, whether deliberate or initiated by control devices, the output of energy due to nuclear fission stops, but heat produced by the decay of fission products continues. On average over the first hour, this heat output amounts to about [[ ]]P,E of the preceding reactor power, which subsequently decays slowly. Integrated over the first 24 hours, it matches [[ ]]P,E of reactor full-power operation. This decay heat is transferred to the coolant and moderator that surround the fuel assemblies, thereby heating these materials. The decay heat must be removed to prevent overheating and failure of the fuel cladding.

In smaller reactors designed for modular construction, such as the Xe-100, decay power is typically removed by passive means (e.g., conduction in the graphite materials).

Per [11], the decay heat generated in the fuel during and following an accident is modeled based on the DIN standard [32] that was derived for pebble bed reactors with a Low Enriched Uranium cycle. The decay heat resulting from decay of fission products and actinides in each pebble is calculated by a convolution integral summing up the contributions of the in-core prior history of each pebble. The Xe-100 code tracks the irradiation and power history of each fuel pebble as part of the process of calculating decay heat.

“Non-local heating” refers to heating other than from fission energy release. The predominant source of non-local heating is energy transfer from neutron and photon interactions with materials in the reactor structures (e.g., control rods, graphite moderator and graphite reflectors, core barrel, pressure vessel). Assessment of these effects requires a tool that can model both the neutron and gamma fields. Including non-local heating effects leads to a temperature redistribution throughout the core and reactor structures. [[

]]P,E A quantitative assessment of non-local heating effects will be included in the next phase of the analysis.

4.3.4. Radionuclide Inventory [11]

The Xe-100 reactor has a functional containment given that the TRISO-X fuel pebbles provide an effective barrier to fission products release under normal and accident conditions. Mechanistic source term analyses are used to demonstrate the consequences of LBEs meet applicable dose or derived acceptance criteria.

In addition to potential releases from the fuel, the generation of graphite and metallic dust within the reactor and fuel handling systems can carry radionuclides through the reactor system. In the event of a primary cooling circuit or fuel handling system breach, a release of fission products is expected.

Predictions of the radiological source term under normal operating and accident conditions are modeled by the MST code. Specifically, there are modules for fuel failure and fission product release, dust production and fission product transport through the facility, and to calculate dose external to the reactor building.

[[

with the dust throughout the HPB.

]][P,E]

The most important mechanisms that are considered for the mass transfer of radionuclides between the helium heat transport fluid and component surfaces within the HPB of a pebble bed HTR are:[[

]][E]

The MST code models gas and fission product transport through the reactor building and stack. Although the conditions are different than in the primary circuit, the transport phenomena occurring in the reactor building are essentially the same, consisting of transport of radionuclide vapors and dust in a flowing gas. This methodology allows condensable radionuclides and dust to be deposited and re-entrained from compartment surfaces, though primarily only deposition phenomena are occurring in the reactor building.

The rates of transport, deposition and re-entrainment are functions of the gas mass flow rate in the compartment.

The code also calculates potential public dose by radionuclides transported via atmospheric dispersion through a model based on the equations and data in RG 1.109 [38]

4.3.5. Functional Containment [11]

The Xe-100 reactor has a functional containment, which is defined as a barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOOs, and accident conditions. There are five concentric functional containment barriers to radionuclide transport, each of which is passive:

- Fuel kernels
- Applied coatings, i.e., Inner Pyrolytic Carbon (IPyC), Silicon Carbide (SiC) and Outer Pyrolytic Carbon (OPyC) layers
- Fuel free zone and graphite matrix pebble in which the fuel grains are contained
- Heat transport system helium pressure boundary (helium piping and reactor pressure vessel)
- Reactor building

With regard to the fuel-related barriers (i.e., the first three barriers in the above), the TRISO fuel is robust and does not rely on a single component to achieve its safety function. For example, each of the layers of the applied coatings plays a role in preventing radionuclides from being released into the helium pressure boundary, and the layers themselves are physically independent. The SiC coating acts as the main load-bearing member of the multi-shelled mini-pressure vessel, withstanding the stresses from internal gas pressure buildup and irradiation-induced dimensional change in the PyC coatings, and provides the primary diffusion barrier to the release of gaseous and metallic fission products. The two PyC layers act as additional diffusion barriers to fission products, especially fission gases. [[

]]^{P,E} An operational limit for activity in the helium is expected to be established. Based on the current analysis results and the margins to the dose limits, the operating limit is expected to be much higher than the normal level of circulating activity.

4.3.6. Integrated System Response

To support the EMDAP documented in RG 1.203, the Xe-100 suite of codes will need to incorporate the integrated system response, which is continuing development as part of the transient and accident analysis methodology. As this information becomes available, X-energy will share with the NRC staff in future submittals.

5. QUALITY ASSURANCE FOR TRANSIENT AND SAFETY ANALYSES

X-energy developed and implemented a Quality Assurance Program (QAP) to support analysis activities. The following list includes all applicable X-energy procedures, as well a short description of the requirements contained in the procedure.

- QAP 3.1 [17], Control of Design & Development Procedure: The Systems Engineering Management Plan (SEMP) defines the overall plan to implement a phased systems engineering approach for Xe-100 design and development activities. This procedure establishes the process, responsibilities and requirements for performing and documenting the design of Structures, Systems, and Components within the context of the SEMP. This procedure, in conjunction with other referenced X-energy procedures, implements the requirements for design control in the X-energy Quality Assurance Program Description.
- QAP 3.2 [18], Technical Analysis Procedure: This procedure establishes the process, responsibilities and requirements for performing and documenting Technical Analyses, including both work planning and executing an analysis.
- QAP 3.6 [19], Software Procedure: The purpose of this procedure is to ensure that software used for design, analysis and supporting activities that impact safety for the Xe-100 program is developed and documented in a planned and systematic manner. This procedure also provides an overview of the software engineering process and provides direction to other X-energy procedures providing more detail on specific software requirements.
- QAP 3.9 [20], Computer Program Technical Evaluation & Acceptance Procedure: The purpose of this procedure is to ensure that Software used for design, analysis and supporting activities that impact nuclear safety for the Xe-100 program is qualified, acquired and used in a planned and systematic manner. Qualification ensures that software is in conformance with the requirements of NQA-16.7. The acquisition process ensures that third-party software is selected, acquired and installed following a systematic and verifiable series of steps. Use requirements ensure that the correct software and associated input data is used in target applications in a verifiable and repeatable manner.
- QAP 3.10 [21], Software V&V for Design & Safety Analysis Procedure: The purpose of this procedure is to ensure that the software used for design, analysis and supporting activities that impact safety for the Xe-100 program is verified and validated in a planned and systematic manner. Verification demonstrates that the software implements the required theoretical and mathematical basis in a correct and error free manner that is consistent with the Software Design Specification (SDS). Validation assesses the fitness-for-purpose of the software and quantifies the accuracy of key parameters calculated by the software.
- QAP 3.11 [22], Software Problem Reporting and Resolution Procedure: The purpose of this procedure is to define the process for software problem reporting and resolution for the Xe-100 program.
- QAP 3.14 [39], Software Configuration and Change Control Procedure: The purpose of this procedure is to ensure that software used for design, analysis and supporting activities that impact safety for the Xe-100 program is placed under a configuration and change control process that maintains an approved baseline version of the software. Configuration and change control

does not necessarily imply the use of any particular tool – it only requires that the software be managed in such a way as to meet configuration and change control requirements. This procedure has been written to address the configuration and change control requirements. This procedure specifies how each software configuration, comprised of the software and its associated components, shall be identified and maintained. The procedure also specifies the requirements for identification and documentation of changes to a configuration component to ensure access to all information necessary to understand the purpose and design of the software being changed.

6. CONCLUSIONS AND NRC REVIEW REQUEST

This report provides the Xe-100 high-level approach to establish a transient and safety analysis methodology and to perform and document safety analyses. The approach aligns with the requirements of RG 1.203 [2] and is informed by the RIPB approach described in NEI 18-04 [4] and X-energy's associated implementing topical report for that approach [10]. This framework will be used as the basis for formulating the Xe-100 transient and safety analysis methodology in preparation of submittal of a SAR.

It is understood that what is being submitted herein does not represent a full transient and safety analysis methodology LTR. Several sections throughout the document describe future X-energy actions, but do not provide the technical basis to support a full review and approval of the proposed methodology (e.g., code V&Vs, basedecks). X-energy will complete this work based on design and licensing program planning, and will provide the NRC staff with supplemental documents and maintain continued communication as appropriate as additional information is available relevant to the analysis methodology.

X-energy requests the NRC staff review and approve the proposed approach as acceptable for continued Xe-100 transient and safety analysis methodology development. This includes relevant assessment where the approach aligns with NRC guidance, such as RG 1.203, RG 1.70, and RG 1.232, as well as the application of a RIPB approach, using NEI 18-04, RG 1.233, and NEI 21-07 guidance.

X-energy will continue to coordinate with the NRC staff for the schedule to provide additional information discussed herein and conduct periodic engagement on this topic.

7. REFERENCES

The following documents are referenced within this topical report.

Document Title	
[1]	10 CFR 50.34, "Contents of applications; technical information."
[2]	U.S. NRC Regulatory Guide 1.203, "Transient and Accident Analysis Methods," December 2005.
[3]	X-energy, "Xe-100 Technical Report, Technology Description," XE00-P-G1ZZ-RDZZ-D-001118, Revision 1, April 2021.
[4]	NEI 18-04, Revision 1, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," August 2019.
[5]	Policy Statement on the Regulation of Advanced Reactors (72 FR 60612, 2008)
[6]	U.S. NRC Regulatory Guide 1.232, "Developing Principal Design Criteria for Non-Light Water Reactors," April 2018.
[7]	U.S. NRC Regulatory Guide 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors," June 2020.
[8]	U.S. NRC Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Revision 3, November 1978.
[9]	NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition."
[10]	X-energy, "Xe-100 Licensing Topical Report Risk-Informed Performance-Based Licensing Basis Development," XE00-R-R1ZZ-RDZZ-L-001522, Revision 1, June 2021.
[11]	X-energy, "Xe-100 Safety Analysis Summary Report," XE00-R-R1ZZ-RDZZ-X-000850, Revision 2, August 2021.
[12]	X-energy, "Xe-100 200MWth Steady-State Core Design Report," XE00-N-RZZ-NSZZ-D-000288, Revision 4, February 2021.
[13]	X-energy, "Xe-100 Mass & Heat Balance Report", XE00-P-DZZ-GLZZ-A-000311, Revision 9, May 2021.
[14]	U.S NRC NUREG-2246, DRAFT, "Fuel Qualification for Advanced Reactors," June 2021.
[15]	X-energy, "VSOP-A Theoretical Description Baseline Report," XE00-T-S3ZZ-SWZZ-N-000232, Revision A, February 2020.
[16]	X-energy, "XGAS: Gaseous Fission Product Transport Code Manual," XE-S2-GL-G0-N11-100432, Revision A, November 2017
[17]	QAP 3.1, Revision 2, "Control of Design & Development Procedure," December 2020.
[18]	QAP 3.2, Revision 3, "Technical Analysis Procedure," February 2021.
[19]	QAP 3.6, Revision 2, "Software Procedure," July 2021.

Document Title	
[20]	QAP 3.9, Revision 1, “Computer Program Technical Evaluation & Acceptance Procedure,” February 2021.
[21]	QAP 3.10, Revision 2, “Software V&V for Design and Safety Analysis Procedure,” July 2021.
[22]	QAP 3.11, Revision 2, “Software Problem Reporting and Resolution Procedure,” September 2021.
[23]	Kinetrics Xe-100 Phenomena Identification and Ranking Tables, April 2021.
[24]	X-energy, “Mechanistic Source Term Approach and XSTERM Code Verification and Validation,” XE00-R-R1ZZ-RDZZ-L-000632,” Revision 1, April 2021.
[25]	NRC Letter Frederick D. Brown to Stephen E. Kuczynski, “Licensing Modernization Project,” February 21, 2018 (ML18047A149).
[26]	ASME/ANS RA-S-1.4-2020 Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants (updated for NRC endorsement, Fall 2020).
[27]	SECY-15-0002 “Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications”
[28]	U.S. Nuclear Regulatory Commission, “Non-Light Water Review Strategy,” Staff White Paper (Draft), September 2019.
[29]	U.S. Congress, “Nuclear Energy Innovation and Modernization Act,” January 2019.
[30]	U.S. Nuclear Regulatory Commission, “Analysis of Applicability of NRC Regulations for Non-Light Water Reactors,” Staff White Paper (Draft), September 2020.
[31]	U.S. Nuclear Regulatory Commission, “Safety Review of Power Reactor Construction Permit Applications,” Staff White Paper (Draft), February 2021.
[32]	Deutsche Institut für Normung eV, “Berechnung der Nachzerfallsleistung der Kernbrennstoffe von Hochtemperaturreaktoren mit Kugelförmigen Brennelementen”, DIN 25485, Mai 1990.
[33]	U.S. Environmental Protection Agency, "Limiting Values of Radionclide Intake and Air Concentration and Dose Conversion Factors for Inahalation, Submersion and Ingestion", EPA Federal Guidance Report No. 11 (1988).
[34]	U.S. Environmental Protection Agency, "External Exposure to Radionuclides in Air, Water, and Soil", EPA Federal Guidance Report No. 12 (1993).
[35]	L. Binkele, “Thermal Conductivity of Neutron-irradiated graphites at temperatures between 50 and 1000 C,” 1972.
[36]	P. Zehner, “Experimentelle und theoretische Bestimmung der effektiven Warmaleitfahigkeit durchstromter Kugelschuttungen bei massigen und hohen Temperaturen,“ 1973.
[37]	K. Robold, “Warmestransport im Innerne und der Randzone von Kugelschattigen,“ July 1982.
[38]	U.S. NRC Regulatory Guide 1.109, Revision 1, “Calculation of Annual Doses to Man rom Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I,” October 1977.

	Document Title
[39]	QAP 3.14, Revision 3, "Software Configuration and Change Control Procedure," September 2021.
[40]	U.S. Congress, "Nuclear Energy Innovation and Modernization Act," January 2019.
[41]	NUREG/CR-6844, Volume 1, "TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables (PIRTs) for Fission Product Transport Due to Manufacturing," July 2004.
[42]	Thykier-Nielsen, S., Deme, S., and Lang, E., "Calculation Method for Gamma Dose Rates from Gaussian Puffs," Riso National Laboratory, Roskilde, Denmark, Riso-R-775(EN), June 1995.
[43]	NEI 21-07, Revision 0, "Technology Inclusive Guidance for Non-Light Water Reactors," August 2021.

8. APPENDICES

8.1. APPENDIX A: NORMAL OPERATIONS

8.1.1. Pre-Equilibrium Core and Initial Start-up Core

It is expected that the fuel temperature coefficient of reactivity will differ in a fresh core or pre-equilibrium core compared to normal operating conditions due to different isotopic makeup of the fuel. Comparable information will be generated for fresh fuel to assess the system response in Initiating Events where the negative fuel temperature feedback is a key element.

Prior to equilibrium operation, the Xe-100 must first be loaded with fuel, brought to criticality under cold conditions, i.e., room temperature, heated up by conventional means at first, then switching to nuclear heating. It will then be in a 'run-in' or pre-equilibrium phase of operation.

The design of this initial period prior to equilibrium is still underway. For illustrative purposes, Sections 8.1.1.1 to 8.1.1.3 summarize the salient features of one startup design proposal, and Section 8.1.1.4 outlines the results of an implementation of that proposal and discusses the calculated temperature reactivity coefficients over the startup and run-in period.

8.1.1.1. First Core Loading and Startup

Initially, the core is loaded with pure graphite pebbles (also called blanks). Cold commissioning tests of the fuel handling system (FHS) are performed first.

In taking the Xe-100 from a core filled with pure/unfueled graphite pebbles (blanks) through its various stages to the steady-state operational modes, the following criteria are observed:

- The total number of pebbles in the core is measured [[ ]]^{P,E}.
- Maximum allowable temperature in the fuel calculated [[ ]]^{P,E}.
- Maximum allowable power per pebble calculated [[ ]]^{P,E}.
- Maximum allowable power per TRISO particle calculated [[ ]]^{P,E}.
- k-eff to remain in the range of 1.000 to 1.001.
- Number of pebbles measured [[ ]]^{P,E}.
- The reserve shutdown system [[ ]]^{P,E} and the reactivity control systems [[ ]]^{P,E} will be employed to ensure that the reactor can be shut down safely at any time.

First criticality is achieved by removing blank spheres and loading fuel spheres [[ ]]^{P,E} with the RCS and RSS rods poised above the core. Once criticality is achieved and new [[ ]]^{P,E} pebbles are added, the RCS rods are used to control criticality while the RSS rods remain poised.

[[ ]]^{P,E}

The start-up/run-in simulations are performed based on the assumption that full power condition (200 MWth) is achieved as soon as neutronically achievable (for practical purposes this approach might deviate from the actual regime to be followed). The following four distinct phases are simulated:

- Phase 1 – Cold (room temperature) criticality[4] achieved [[ ]]P,E. Upon reaching criticality, the RCS and RSS control rods worth will be measured by rod insertion tests.
- Phase 2 – Hot criticality with starting and ending [[ ]]P,E. The RCS and RSS rods worth will be confirmed under hot conditions.
- Phase 3 – Running-in with starting and final [[ ]]P,E.
- Phase 4 – Equilibrium cycle.

An important capability of the system of multi-physics codes is demonstrated in Section 8.1.1.4, i.e., the explicit thermal cell definition to evaluate the thermal spectra related to the changing fuel composition in the core in association with the updated core composition throughout the transition and run-in phases of the reactor. A modeling approach was followed to simulate the calculated switchover of the temperature-dependent resonance integrals at each of these discrete positions to ensure a relatively smooth transition of k-eff.

8.1.1.2. Heating of Core

The next step is [[ ]]P,E

The following step is [[ ]]P,E

8.1.1.3. Run-In Period of Operation

After the onset of nuclear heating the next aim is [[ ]]E. The RSS and RCS rods are again inserted and withdrawn to demonstrate the shutdown margins.

After about [[ ]]P,E, switchover to full power mode is possible. For purposes of the start-up and run-in calculations, this is demonstrated and after the switchover to 200 MWth power operations, the RSS and RCS rods are inserted and withdrawn to demonstrate the shutdown margins.

Between [[ ]]P,E, the RCS can be inserted into positions to maintain criticality.

On about [[ ]]P,E, the reactivity requirement demands a fueling rate that would demand [[ ]]P,E.

By this time all graphite spheres are removed and the start to steady state conditions are initiated. Reactivity adjustments will continue for some time by the FHS manipulation.

As the run-in phase progresses, fission products are building in and all neutronics and feedback parameters are converging on their equilibrium steady-state values.

8.1.1.4. Sample Implementation of One Pre-Equilibrium Core Design Option

The results of the design options described in the three preceding subsections are described in this section. The core startup and run-in period were simulated with a neutronics code [[ ]]P,E and operating at 200 MWth power.

Figure 6 shows the gradual replacement of graphite pellets with fuel pebbles during the run-in period, and Figure 7 shows the ramping up in power and maximum fuel temperature over the same period.

[4] First criticality under cold conditions is actually safer than going first critical under hot conditions. The core cannot cool any further, so there is no excess reactivity due to the negative temperature coefficient.

Figure 6: [[ ]]P,E

Figure 7: [[ ]]P,E

8.1.1.4.1. Temperature Reactivity Coefficient in Pre-Equilibrium Core

The temperature reactivity coefficient is a key measure of intrinsic core safety characteristics for any particular state.

The temperature reactivity coefficient was computed with a neutronics code at 6 different core states, with progressively increasing reactor power and decreasing number of graphite pellets in the core.

The pre-equilibrium core coefficients are preliminary, and subject to change. Nonetheless, it is noted that:

- The temperature coefficients for the first four states are dominated by the much larger (negative) moderator coefficient and (positive) reflector coefficient, due to the significant amounts of graphite pebbles in the core.
- The temperature reactivity coefficient remains negative for all pre-equilibrium core states examined and is consistently more negative than for the equilibrium core.

These results are preliminary but indicate that the safety characteristics of a fresh core, as exemplified by the temperature coefficients, are better than those of an equilibrium core. Consequently, the results of a postulated accident occurring in an equilibrium core and that leads to fuel heatup are expected to bound the results of the same postulated accident occurring during the startup and pre-equilibrium phases.

8.2. APPENDIX B: DEPRESSURIZED LOSS OF FORCED CIRCULATION (DLOFC) ANALYSIS

A DLOFC is defined as a failure in the HPB that leads to depressurization of the heat transport system. The specific event analysis in this section is a 65mm break in the cold leg of the HPB. It is noted that there is no large piping connected to the HPB, which primarily comprises the reactor vessel, the cross vessel and the steam generator. While classification of initiating events will be finalized during the Preliminary Design phase, it is anticipated that this event will be classified as a BDBE due to the inherent safety features of the Xe-100 plant, and the PRA predicted frequency of the event.

8.2.1. Important Phenomena

As described in Section 3.2.1.1, a preliminary PIRT was previously compiled for Xe-100 [23]. An explanation of how the PIRT was created is contained in Section 3.2.1.1. The Xe-100 PIRT includes a ranking of phenomena for normal operating conditions, as well as for several safety analysis cases. DLOFC was included in the PIRT.

Table 2 provides the phenomena ranking for DLOFC.

8.2.2. Event Sequence Description

The loss of helium pressure and inventory leads to undercooling of the fuel, which causes fuel temperatures to increase. RPS action can initiate by several parameters, [[

]]P

The release of helium from the HPB causes pressurization of the reactor building. Figures 8 through 10 show the response within the reactor pressure vessel citadel and the steam generator citadel to the overpressure. Helium will preferentially fill the tops of the citadels and push air towards the bottom. [[

]]P

Heatup of the fuel leads to a reduction in reactor power due to the strong negative temperature coefficient of reactivity and the reactor is effectively shut down [[

]]P,E Fuel temperatures slowly decrease thereafter due to effective passive heat removal and the slow reduction in decay power.

The reactor continues to be sub-critical due to the strong negative temperature coefficient, but the reactivity margin slowly decreases once fuel temperatures start to decrease and due to the decay of neutron absorbing fission products such as xenon. If no recovery actions are taken, the reactor can become recritical after [[
]]P,E. Should recriticality occur, the associated increase in fuel temperatures limits reactor power to a small fraction of the decay power, and fuel temperatures remain well below the peak experienced during the transient. The reactor can remain in this quasi-steady state indefinitely with no adverse effects on fuel or radioactivity release.

[[

]]P,E, and the predicted dose to the public is well below the BDBE dose limit.

8.2.3. Methodology and Assumptions

The DLOFC event documented in this Appendix was analyzed with XSTERM as the MST suite of computer codes, and using the following thermal-hydraulic boundary conditions provided by the Flownex code: [[
]]P,E. See Figure 11 for a process flow diagram that

illustrates how the DLOFC event described herein was performed. As described in the body of this report, X-energy is still finalizing which codes will be used and how they will be used for the X-energy Transient and Safety Analysis methodology. This appendix provides an example of codes that may be used to perform the analyses for Xe-100.

The initial steady-state conditions are shown in Table 1.

[[

]]

The public dose consequences are calculated for the following case:

- Explicit depressurization modeling of a 65 mm break with radionuclides released [[

]]^{P,E}

[[

]]^{P,E}

[[

]]P,E.

8.2.4. Results

Results for this case for the depressurization period are shown in Figure 12. The figure shows the reduction in helium flow rate and inventory resulting from the HPB break. With the decrease in helium flow rate, there is a decrease in the helium inlet temperature. [[

]]P,E.

After the depressurization phase, the fuel will continue to heat up because there is no forced circulation to remove heat. Passive heat removal limits the temperature excursion as shown in Figure 13. [[

]]P,E

The calculated dose is 0.091 rem at the 400 meters EAB. This is about 0.36% of the DBA dose limit of 25 rem TEDE.

[[

]]P,E

The decay of xenon and other neutron absorbing fission products will lead to recriticality. [[

]]P,E

8.2.5. Conclusions

A postulated 65 mm HPB break has been analyzed. The results indicate that the negative temperature coefficient of reactivity is effective in shutting the reactor down. The results also indicate that passive heat removal is effective in limiting fuel temperature excursions such that radioactivity release is small and the dose at the 400 meters EAB is 0.091 rem, compared to the DBA dose limit of 25 rem TEDE. It is expected that accounting for radioactivity holdup, deposition and plateout in the RB will reduce the predicted dose.

Table 2. [[ ]]

[[				

]][P,E]

Figure 8: [[ ]]^{P,E}

Figure 9: [[ ]]^{P,E}

Figure 10: [[ ]]P,E

Figure 11: [[ ]]P,E

Figure 12: [[ ]]P,E

Figure 13: [[ ]]P,E

8.3. APPENDIX C: SPURIOUS WITHDRAWAL OF REACTIVITY CONTROL SYSTEM CONTROL RODS ANALYSIS

8.3.1. Important Phenomena

As described in Section 3.2.1.1, a preliminary PIRT was previously compiled for Xe-100 [23]. An explanation of how the PIRT was created is contained in Section 3.2.1.1. The Xe-100 PIRT includes a ranking of phenomena for normal operating conditions, as well as for several safety analysis cases. The spurious withdrawal of reactivity control system control rods analysis was included in the PIRT.

Table 3 provides the phenomena ranking for the spurious withdrawal of reactivity control system control rods analysis.

8.3.2. Event Sequence Description

In this event, the core is initially operating at 100% MCR and the [[

]]^{P,E}

The event begins with all [[ ]]^{P,E} control rods being withdrawn by an unspecified postulated RCS failure at an assumed maximum rate of [[

]]^{P,E} This positive reactivity results in a power increase and reactor outlet temperature increase. [[

]]^{P,E}

While classification of initiating events will be finalized during the Preliminary Design phase, it is anticipated that this event will be classified as a BDBE due to the inherent safety features of the Xe-100 plant, and the PRA predicted frequency of the event. [[

]]^{P,E}

The core inherent negative temperature coefficient provides a source of negative reactivity that counters the positive reactivity from control rod withdrawal, and causes the reactor to stabilize at a very low power level where the positive reactivity from control rod withdrawal balances the negative reactivity due to the temperature coefficient.

Over the longer timeframe of hours, xenon buildup and decay is a dominant effect. [[

]]^{P,E}

8.3.3. Methodology and Assumptions

The spurious withdrawal of reactivity control system control rods event documented in this appendix is analyzed with XSTERM as the MST suite of computer codes, and using the following thermal-hydraulic boundary conditions provided by the Flownex code: [[
]]^{P,E}. See Figure 14 for a process flow diagram that illustrates how the control rods withdrawal event described herein was performed. As described in the body of this report, X-energy is still finalizing which codes will be used and how they will be used for the X-energy transient and safety analysis methodology. This appendix provides an example of codes that may be used to perform the analyses for Xe-100.

The initial steady-state conditions are shown in Table 1.

XSTERM (XTDYN and XSIM modules) simulates the reactivity, power and temperature transients, as well as probability of fuel failures (XFP module) for different scenarios. The XSTERM kinetics model is applied to a two-dimensional r-z representation of the core.

[[

]]P,E

8.3.4. Results – Short-Term

Control rods start to withdraw [[

]]P,E

[[

]]P,E

[[

]]P,E

8.3.5. Results – Long-Term

[[

]]P,E

8.3.6. Conclusions

A spurious control rod withdrawal initiated from full power results in an increase in reactor outlet temperature, but the reactivity due to control rod withdrawal is largely balanced by negative temperature feedback reactivity, [[

]]P,E

[5] The bottom tips of the RCS rods are all [ ]] below the bottom of the top reflector.

[[

]][p] the inherent reactivity characteristics alone are sufficient to control the reactivity transient. The probability of fuel failures is negligible, with no release to the environment.

Table 3. [[ ]]

[[		

]]P,E

Figure 14: [[ ]]P,E

Figure 15: [[ ]]P,E

Figure 16: [[ ]]^{P,E}