Xe-100 Licensing Topical Report GOTHIC and Flownex Analysis Codes Qualification

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SYNOPSIS

This Licensing Topical Report (LTR) provides X-energy, LLC's (X-energy) approach to develop evaluation models (EMs) in Flownex and GOTHIC 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 EM Development and Assessment Process (EMDAP) described in U.S. Nuclear Regulatory Commission Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods," as applicable, to prepare the Xe-100 plant evaluation model. Regulatory Guide 1.203 includes the following four elements, which are addressed within the report:

1. Transient and Safety Analysis Codes (Flownex and GOTHIC) Methodology
2. Transient and Safety Analysis Codes (Flownex and GOTHIC) Evaluation Models
3. Transient and Safety Analysis Codes (Flownex and GOTHIC) Qualification
4. Transient and Safety Analysis Codes Quality Assurance

Additionally, the approach X-energy has provided for developing the transient and safety analysis codes methodology are in accordance with the regulatory requirements discussed in Section 2. As a starting point, this report provides the overall framework and approach taken by X-energy to develop specific code methodologies, code evaluation models, and code Verification and Validation (V&V).

X-energy is seeking U.S. Nuclear Regulatory Commission review and approval of the proposed Flownex and GOTHIC codes to perform 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 possibly a future licensing application under 10 CFR 53.

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Abbreviations/Acronyms

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Short Form	Phrase
AGSS	**Auxiliary Gas Services System**
AOO	**Anticipated 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**
BV	**Bypass Valve**
CB	**Core Barrell**
CDCW	**Condenser Circulating Water (CDCW)**
CFD	**Computational Fluid Dynamics**
CFR	**Code of Federal Regulations**
CI	**Conventional Island**
CICW	**Conventional Island Cooling Water**
CLS	**Core Loading System**
COL	**Combined Operating License**
CP	**Construction Permit**
CRW	**Control Rod Withdrawal**
CSA	**Canadian Standards Association**

CSAU	**Code Scaling, Applicability, and Uncertainty**
CUS	**Core Unloading System**
DA	**Deaerator**
DBA	**Design Basis Accident**
DBE	**Design Basis Event**
DC	**Design Certification**
DCS	**Distribution Control System**
DCL	**Deaerator Cooling Loop**
DCS	**Distributed Control System**
DHL	**Deaerator Heating Loop**
DID	**Defence in Depth**
DLOFC	**Depressurized Loss of Forced Cooling**
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**
FCV	**Flow Control Valve**
FHCS	**Fuel Handling Control System**
FHS	**Fuel Handling System**
FR	**Flow Restrictors**
FSAR	**Final Safety Analysis Report**

FOM	**Figure of Merit**
FSC	**Flownex Screening Criteria**
FSS	**Fuel Handling Support System**
FW	**Feedwater**
GDC	**General Design Criteria**
GUI	**Graphical User Interface**
GS	**Gland Steam**
HPB	**Helium Pressure Boundary**
HPFWP	**High Pressure Feedwater Pump**
HSS	**Helium Service System**
HTGR	**High Temperature Gas-Cooled Reactor**
HTR	**High Temperature Reactor**
HTR-10	**10 MW High Temperature Gas-Cooled Reactor Test Module**
HTS	**Heat Transport System**
HTTR	**High Temperature Engineering Test Reactor**
HVAC	**Heating, Ventilation and Air Conditioning**
HX	**Heat Exchanger**
IET	**Integral Effects Test**
IPS	**Investment Protection System**
IPT	**Intermediate Pressure Turbine**
IPyC	**Inner Pyrolytic Carbon**
ISF	**Intermediate Storage Facility**
IV	**Isolation Valve**
JAERI	**Japan Atomic Energy Research Institute**

LAR	**License Amendment Request**
LBE	**Licensing Basis Event**
LCV	**Level Control Valve**
LEU	**Low Enriched Uranium**
LMP	**Licensing Modernization Project**
LOCA	**Loss of Coolant Accident**
LOFC	**Loss of Forced Coolant**
LPFWHX	**Low Pressure Feedwater Heat Exchanger**
LPT	**Low Pressure Turbine**
LPZ	**Low-Population Zone**
LTR	**Licensing Topical Report**
LWA	**Limited Work Authorization**
LWR	**Light-Water Reactor**
MCR	**Maximum Continuous Rating**
MF	**Mass Fraction**
MHTGR	**Modular High Temperature Gas-Cooled Reactor**
ML	**Manufacturing License**
MSL	**Main Steam Line**
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**

NIAB	**Nuclear Island Auxiliary Building**
NIFW	**Nuclear Island Feed Water**
NILR	**Nuclear Island Liquid Radwaste**
NIPW	**Nuclear Island Process Water**
NNR	**National Nuclear Regulator**
NOC	**Normal Operating Condition**
NRC	**Nuclear Regulatory Commission**
OL	**Operating License**
OPyC	**Outer Pyrolytic Carbon**
PDC	**Principal Design Criteria**
P&ID	**Piping and Instrument Diagram**
PIRT	**Phenomena Identification and Ranking Table**
PLOFC	**Pressurized Loss of Forced Cooling**
PRA	**Probabilistic Risk Assessment**
PRV	**Pressure Relief Valve**
PSAR	**Preliminary Safety Analysis Report**
PSF	**PRA Safety Functions**
PyC	**Pyrolytic Carbon**
QA	**Quality Assurance**
QAP	**Quality Assurance Procedure**
QAPD	**Quality Assurance Program Description**
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**
SBS	**System Breakdown Structure**
SDA	**Standard Design Approval**
SDD	**System Design Description**
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**
SG	**Steam Generator**
SGDS	**Steam Generator Dump System**
SGDT	**Steam Generator Dump Tank**
SiC	**Silicon Carbide**
SP	**Standpipe**
SRNS	**Sphere Recirculation System**

SRP	**Standard Review Plan**
SRTS	**Sphere Replenishment System**
SSC	**Structure, System, and Component**
SSS	**Startup and Shutdown System**
TB	**Turbine Building**
TEDE	**Total Effective Dose Equivalent**
TICAP	**Technology-Inclusive Content of Application Project**
TI-RIPB	**Technology-Inclusive, Risk-Informed, and Performance-Based**
TRISO	**TRistructural ISOtropic**
TSA	**Transient and Safety Analysis**
UCO	**Uranium Oxy-Carbide**
V&V	**Verification and Validation**
X-energy	**X Energy, LLC**

1. Introduction

1.1 Purpose

This report documents the GOTHIC and Flownex Codes used by the safety analysis team for work supporting the submittal of a Preliminary Safety Analysis Report (PSAR) and the Final Safety Analysis Report (FSAR). It provides the theory associated with GOTHIC and Flownex base models, and the specific theory utilized for the Xe-100. In addition, this report documents the code manuals and qualifications for GOTHIC and Flownex; which includes validation.

1.2 Scope

The GOTHIC and Flownex code theories described herein apply to the transient and safety analysis performed to support the preparation of the Safety Analysis Report (SAR) as defined in 10 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 the code theory of how X-energy plans to approach performing production safety analyses. Reference [2] describes specific methodology elements described in U.S. Nuclear Regulatory Commission (NRC) RG 1.203, “Transient and Accident Analysis Methods” [3]. This report does not provide all the details of the specific codes used to perform safety analyses or how specific analyses have been categorized/performed.

The GOTHIC and Flownex theories for the transient and safety analysis EMs and analysis methods provided in this report are applicable to 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 if the code theories change as a means of providing the NRC staff the opportunity to perform reviews, audits, and inspections. However, no changes are expected to the code theories.

This report is intended to document the GOTHIC and Flownex code theories that will be used for safety analysis event transients. No event-specific analyses, results, or conclusions are presented within this document. The full mass and energy balance Flownex model for the Xe-100 was developed in Reference [4]. Components implementing the protection system controller, Reactor Protection System (RPS), Distribution Control System (DCS), and Investment Protection System (IPS) logic were added to that model as documented in Reference [5]. This report documents the changes and simplifications made to Reference [5] to support safety analysis work.

The conclusions presented in this report will support the Xe-100 Transient and Safety Analysis (TSA). Future uses of the model include (but are not limited to) evaluation of proposed Reactor Building (RB) design changes and evaluation of Reactor Cavity Cooling System (RCCS) design changes.

1.3 Interfacing Reference

This report incorporates insights from several sources as described in the Reference section (Section 8) and throughout the report. The related Flownex Safety Analysis Base Model Report [6], Preliminary GOTHIC Reactor System Model Report [7], verification and validation of the safety analysis code suite, application of Nuclear Energy Institute (NEI) 18-04 [8], and associated details about the validation of the codes are provided in separate LTRs. Reviewers are also advised to reference the Xe-100 Technology Description Technical Report [9] for details regarding the Xe-100 design and unit and plant operations.

1.4 Document Layout

This report documents the Analysis Codes used to perform the analysis for transient and accident analysis for the Xe-100 plant. This report presents the high-level approach that X-energy is applying to develop GOTHIC and Flownex evaluation models that perform the analysis for review by the NRC staff through the guidance of RG 1.203 [3]. It summarizes the relevant regulatory requirements and guidance, the theory for Code Methodologies and Code Models of the Xe-100 plant, V&V, and quality assurance performed to support the Xe-100 plant. The layout of the report follows the elements described in RG 1.203. Any discussion of methodology 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 [8].

1.5 Outcome Objectives

The Transient and Safety Analysis (TSA) Codes LTR is being submitted to the NRC for review and approval. It is being developed to support the TSA Methodology LTR, and ultimately the development of the PSAR and FSAR.

2. Overview of Regulatory Requirements and Guidance

The NRC provides rules for the design, licensing, construction, operation, and decommissioning of reactors 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 approaches 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 SECY Paper Precedents

2.1.1 SECY-15-0002

As described in SECY-15-0002 “Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications” [10], 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.

2.2 Code of Federal Regulations

2.2.1 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 Construction Permit (CP) and Operating License (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 PSAR and FSAR 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).

2.2.2 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(a) and 10 CFR 50.34(b) includes the following requirements for the PSAR and the FSAR, which are applicable to this report:

> **(a)** *Preliminary safety analysis report*. Each application for a construction permit shall include a

preliminary safety analysis report. The minimum information[5] to be included shall consist of the following:

(a)(1)(i) A description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design. Special attention should be directed to the site evaluation factors identified in part 100 of this chapter. The assessment must contain an analysis and evaluation of the major structures, systems and components of the facility which bear significantly on the acceptability of the site under the site evaluation factors identified in part 100 of this chapter, assuming that the facility will be operated at the ultimate power level which is contemplated by the applicant. With respect to operation at the projected initial power level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (a)(8) of this section, as well as the information required by this paragraph, in support of the application for a construction permit, or a design approval.

(a)(1)(ii) A description and safety assessment of the site and a safety assessment of the facility. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The following power reactor design characteristics and proposed operation will be taken into consideration by the Commission:

(a)(1)(ii)(E)(4) A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance and the need for high point vents following postulated loss-of-coolant accidents must be performed in accordance with the requirements of § 50.46 and § 50.46a of this part for facilities for which construction permits may be issued after December 28, 1974.

(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:

(b)(2) A description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, 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.

(b)(2)(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.

(b)(2)(i)(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.

2.3 Regulatory Guidance

2.3.1 Policy Statement on the Regulation of Advanced Reactors

The Advanced Reactor Policy Statement [11] 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.

2.3.2 NRC Regulatory Guide 1.203, Transient and Accident Analysis Methods

RG 1.203 [3] describes a process that the NRC staff considers acceptable for use in developing and assessing 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 [12] for Loss of Coolant Accident (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:

A. 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).
B. Specification of those portions of the analysis not included in the computer programs for which alternative approaches are used.
C. 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:

A. Determine requirements for the EM. 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 [82]. 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.

B. 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.
C. Develop the EM. 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.
D. Assess the adequacy of the EM. 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.
E. Follow an appropriate quality assurance protocol during the EMDAP. Quality assurance standards, as required in Appendix B to 10 CFR Part 50 [13], 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.
F. 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 of Reference [3].

2.3.3 RG 1.232, *Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors*

RG 1.232 [14] 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 to 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 [8].

2.3.4 RG 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*

RG 1.233 [15] provides the NRC staff's guidance on using a TI-RIPB methodology to inform the licensing basis and content of applications for non-LWRs, including, but not limited to, molten salt reactors, HTGRs, 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."

RG 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 [16]. NEI 18-04 outlines an approach for use by reactor developers to select LBEs, classify Structures, Systems, and Components (SSCs), determine special treatments and programmatic controls, and assess the adequacy of a design in terms of providing layers of Defense in Depth (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.

X-energy is implementing the NEI 18-04 approach and uses safety analysis methods as described in References [6] and [2] as a means of evaluating plant performance and response to various LBEs.

2.3.5 RG 1.70, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition*

RG 1.70 [17] 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 is currently being evaluated.

2.3.6 NUREG-0800, *NRC Standard Review Plan for LWRs*

NUREG-0800 [18] The standard review plan (SRP) provides guidance to the U.S. NRC staff in performing safety reviews of CP or OL applications (including request for amendments) under 10 CFR 50 and Early Site Permit (ESP), design Certification (DC) Combined Operating License (COL), Standard Design Approval (SDA) or Manufacturing License (ML) applications under 10 CFR Part 52 (including requests for amendments).

The principal purpose of the SRP is to assure the quality and uniformity of staff safety reviews. It is also the intent of this plan to make information about regulatory matters widely available and to improve communication between the NRC, interested members of the public, and the nuclear power industry, thereby increasing understanding of the NRC review process.

2.4 Additional NRC Guidance

2.4.1 NRC Non-Light Water Reactor Review Strategy (Draft)

The NRC staff issued this draft white paper [19] 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) [20].

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.

2.4.2 NRC Analysis of Applicability of NRC Regulations for Non-Light Water Reactors (Draft)

In September of 2019, and updated in September 2020 and July 2021, the NRC staff produced a draft white paper [21] providing their analysis and assessment of the applicability of regulations to non-LWR technologies [15]. 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.

2.4.3 TICAP/ARCAP

Since the issuance of NEI 18-04 [8] in 2019 and its associated endorsement by the NRC staff in RG 1.233 [15], 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 [22] 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.

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

The NRC staff issued this draft white paper [23] 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 July 2021 [21] by focusing on the necessary information required at the PSAR 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.

2.5 US HTGR Precedents

The X-energy team evaluated multiple reactor technology safety analysis reports and, when available, safety evaluation documents. The extensive experience gained through the General Atomics’ Modular HTGR review, as documented in NUREG -1338 [24], 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.

Exelon transmitted a general description of the South African version of the Pebble Bed Modular Reactor (PBMR) to the NRC as an introduction to the PBMR [25]. It was transmitted for information only and no formal review was requested. The body of the document stated, " This document presents an overview of the PBMR, the concept, the basic principles of design, safety analysis, and operation. It is not intended to be used (or maintained) as a design document."

3. Overview of Safety Analysis Codes Evaluation Models

3.1 Introduction

There are two codes utilized for the Xe-100 Safety Analysis integrated plant transients, namely Flownex and GOTHIC. Flownex provides the short-term transient response and GOTHIC provides the long-term transient response. The short-term transient is defined as the period at which SSCs are actively responding to an initiating event, forced cooling remains available or the primary system is actively depressurizing. The long-term transient is defined as the period at which passive heat transfer begins and no additional active plant responses to the initiating event are considered. The safety analysis team is primarily utilizing Flownex to capture the short-term transient response of the plant, focusing on the primary system conditions and key secondary side systems such as the Steam Generator (SG). The [4] model is built in a modular fashion, such that the boundaries between subsystems are well-defined.

Flownex uses 1D Finite Volume discretization in general and 2D axi-symmetric discretization for the reactor. Flownex also solves the partial differential equations for mass, momentum, and energy conservation to obtain the mass flow, pressure, and temperature distributions throughout Xe-100. Flownex is used to model the Xe-100 for use in evaluating transients for safety analysis.

GOTHIC is a hybrid code that bridges the gap between computational fluid dynamics (CFD) and standard thermal-hydraulic systems analysis codes. Lumped parameter and 3-D regions can be combined into a single model to allow for detailed analysis in regions of interest. As a result, assuming proper care is taken in development of the model, a relatively fast running simulation can be produced. These features allow for the development of models suitable for sensitivity studies and design change analyses.

With Flownex, the reactor cavity cooling system (RCCS) is a passive air-flow system that is modeled with constant flow rates and tuned to computational fluid dynamic (CFD) results to approximate its behavior, benchmarked to normal operating conditions. Future implementation of RCCS in Flownex is expected to be benchmarked to GOTHIC modeling of long-term DLOFC and PLOFC events.

In broad terms, model experience has shown that the fuel temperatures are not strongly affected by moderate deviations in the RCCS performance over short time frames. Vessel temperatures, on the other hand, will be strongly affected by it. Future implementation of RCCS in Flownex for PSAR submittal is expected to be based on GOTHIC modeling of long-term DLOFC and PLOFC events.

3.2 Flownex – Xe-100 Transient and Safety Analysis

The Flownex EM is used to calculate mass flows, temperatures, and pressures in the reactor core during expected operational modes and states, as well as under accident conditions. It is used for both steady state and transient simulations. Below is a brief description of the Xe-100 Flownex Theory. A brief discussion of the base code is found in Appendix A.

3.2.1 Flownex Theory

The Flownex SE program is an engineering simulation program that represents thermo-fluid systems as networks of one-dimensional components. It is scheduled to be used for Xe-100 safety analysis and is capable of modelling reactor kinetics and thermal-hydraulic phenomena. The complete Xe-100 thermal-hydraulic architecture, including the primary system, secondary system and power conversion cycle, is modelled with the Flownex SE code. Flownex is used to predict the overall system response of the reactor when integrated with the primary and secondary heat transport systems for the Xe-100 safety analyses. It is also used to provide thermal and flow boundary conditions to other codes. To support the use of Flownex for Xe-100 safety analyses, Flownex validation for steady-state fuel temperature calculation is performed by using the High Temperature Reactor (HTR)-10 benchmark.

Sections 3.1.3 through 3.1.8 briefly discusses the specific Xe-100 base Flownex Model changes. This is discussed in more detail in References [26], [27], [28], [29], [30], [6], [31], and [32].

3.2.2 Flownex Model – Xe-100 Transient and Safety Analysis

The Flownex Model is described in the Xe-100 Licensing Topical Report Transient and Safety Analysis Methodologies [2].

The Flownex Model contains the following systems/components and discusses the inputs, assumptions, geometry, boundary conditions, results, risks, and conclusions for each of these subsystem models. These are discussed in Reference [6].

- Deaerator
- Reactor Cavity Cooling System
- Startup/Shutdown System
- Steam Generator
- Turbine and Condenser
- Reactor and Helium Pressure Boundary
- Material Properties (including Fuel)
- Core Average Steam Concentration

The following sections describe how the system was modeled in Flownex for the TSA.

3.2.3 Deaerator

The Deaerator (DA) was removed from the Flownex model. The DA tank, and all upstream piping between the SSS and Turbine Bypass have been removed from the model. The removal of the DA tank component assumes that all the steam through the turbine bypass and turbine goes to the DA tank. This is discussed in more detail in References [26] and [6].

3.2.4 Reactor Cavity Cooling System

The RCCS is designed as an air-cooled passive system with a dual inlet/outlet plenum based on [33]. It is a safety related system that is expected to operate during all transients. The RCCS Structures, Systems, and Components (SSC) shall be capable of controlling heat removal for all LBEs. Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure the system safety function can be accomplished, assuming a single failure. The RCCS shall be designed to limit the maximum concrete temperature to maintain its integrity. This is discussed in more detail in References [6] and [31]. With Flownex, the reactor cavity cooling system (RCCS) is a passive air-flow system that is modeled with constant flow rates and tuned to computational fluid dynamic (CFD) results to approximate its behavior, benchmarked to normal operating conditions.

3.2.5 Startup/Shutdown System

The safety analysis base model includes a simplified representation of the SSS to simulate the effects of decay heat removal and sensible heat removal from the Xe-100 reactor. The purpose of the simulated SSS is to provide secondary heat removal when main loop cooling capabilities are unavailable. This is discussed in more detail in References [6] and [27].

3.2.6 Steam Generator

In the SG, the heat transfer is assumed to be uniform in the radial direction in the tube coil region, so only the axial subdivisions were defined regarding the number of axial nodes. In addition, the water/steam side of the SG tube coils are modeled. Similar to the helium volume, the heat transfer is assumed to be uniform in the radial direction in tube coil region. Axially, this volume was divided into axial nodes to match the axial regions for the helium side of the tube coils.

A negligible amount of volume (~1.0 L) was added to the nodes in the relief valve lines to improve numerical stability in the low flow lines between the steam lines and the relief valve discharge nodes. The discharge boundary conditions were set with a superheated temperature of 130°C to prevent two-phase conditions out of the relief valves, which have been observed to lead to model errors. The Steam Generator Dump System is not included in the model explicitly but is represented by a boundary condition. This is discussed in more detail in References [6], [27], and [29].

3.2.7 Material Properties

All fluid and solid material properties in the model are inherited from Reference [31], excluding the exceptions detailed in Section 2.4 of Reference [6]. The exceptions are related to the modeling of irradiation-induced aging effects of graphite within the Xe-100 reactor. Certain graphite properties are strongly dependent on the amount of irradiation damage they incur during their operational lifetimes. Updates have been made to the model to allow for a detailed accounting of these effects and their impacts on plant response and heat transfer. This is discussed in more detail in References [6] and [34].

3.2.8 Reactor and Helium Pressure Boundary

The Flownex reactor model that is used from [31] models the neutronic response of the Xe-100 reactor using a point kinetics script that is delivered as part of the Flownex software package. It uses inputs derived from other neutronics simulation software (primarily VSOP) to calculate the reactor fission and decay power. This is discussed in more detail in References [6] and [31].

3.2.9 Flow Paths

It is assumed that the model will be based on a two-dimensional axially symmetric coordinate system rather than a full three-dimensional cylindrical coordinate system. This implies that all variations in geometry or material properties around the perimeter of the reactor will be spread evenly around the circumference at that radius to form a material with constant properties at a given height and radius. [38].

3.2.10 Heat Transfer

In Flownex, heat transfer through graphite is not significantly affected by density changes and is not accounted for because irradiation damage does not affect the mass of the graphite just the volume a given mass occupies. [6]

Reactor Pressure Vessel (RPV) temperatures are reported by aggregating the nodal results from the appropriate components. The heat transfer coefficients for convection, radiation, and conduction are unchanged from the upstream model. More detailed analysis will be conducted outside of event transient analysis to determine the validity of the RPV temperatures, but analysts may report this as best-available data. [6]

In the absence of forced helium flow, heat transfer within the pebble bed is dominated by radiative and conductive mechanisms. As described in Reference [73], the complexities of these phenomena may be represented in a simplified manner using an effective thermal conductivity between pebble nodes that varies as a function of fuel pebble surface temperature. Additional information for pebble to pebble heat transfer is found in Reference [6].

3.2.11 Boundary Conditions

All boundary condition components in the model are described in Table 17 and Table 18 of Reference [6], along with their values for select circumstances. The analyst may change them for their specific event as appropriate. Table 17 of Reference [6] lists the boundary conditions used in the simplified model for transient analyses. The listed values correspond to the parameters used to establish the 100% power controlled operating conditions. They are derived from the upstream [84] model conditions at 100% power and the various sub-models. Most of these boundary conditions will remain unchanged during an event unless that event specifically requires a change.

The boundary conditions in Table 18 of Reference [6] and component inputs must be set when performing a Steady State solution, in addition to those listed in Table 17 of Reference [6]. The values shown in Table 18 of Reference [6] are for 100% power conditions and match those used in [84]. They are the stored values for the *SS-100pct* snap. If a user requires a new snap at a different power level, it is suggested to use values to match the appropriate snap in Reference [84].

3.2.12 Nodes

Flownex uses nodes and elements to represent a thermal-fluid network graphically. Elements are components such as pipes, pumps, valves, compressors or heat exchangers, while nodes are the end points of elements. Elements can be connected in any arbitrary way at common nodes to form a network.

Flownex solves the momentum equation in each element and the continuity and energy equation at each node. Although components may be represented on the systems level as a single entity, they may in actual fact be complex sub-networks. The main network with embedded sub-networks is treated as one large network in the solution algorithm. [38]

3.2.13 GOTHIC – Xe-100 Transient and Safety Analysis

GOTHIC was developed and is maintained under Numerical Advisory Solutions, LLC QA Program that conforms to the requirements of 10CFR50 Appendix B and 10CFR50 Part 21. GOTHIC is a hybrid code that bridges the gap between computational fluid dynamics (CFD) and standard thermal-hydraulic systems analysis codes. Lumped parameter and 3-D regions can be combined into a single model to allow for detailed analysis in regions of interest. As a result, assuming proper care is taken in development of the model, a relatively fast running simulation can be produced. These features allow for the development of models suitable for sensitivity studies and design change analyses. The purpose of this model is to provide a relatively fast running tool to provide insight into dose analysis, RCCS design changes, and reactor building design changes. [7]

Thermal-hydraulic analysis is needed for a variety of reasons; to address certain safety issues such as containment response following a loss of coolant accident (LOCA); to predict the containment response to a change in operating conditions or equipment; for designing containment systems; to predict performance of special purpose systems and components; to model the behavior of plant buildings under normal operating conditions or in response to a loss of ventilation or cooling; or for a wide variety of situations that involve the response of a system to applied mass and energy sources. In these analyses, the focus may be on the maximum pressure or temperature, pressure loadings on internal structures due to jet impingement or pressure gradients, local temperature variation, equipment temperatures, buildup of volatile gases or a variety of other conditions of interest.

The information below discusses the application of the GOTHIC theory specifically required for the GOTHIC Reactor System Mode used in transient and safety analysis. Appendix B discusses the base model theory. Specifics on the Xe-100 GOTHIC Methodologies and Model are found in the Xe-100 Licensing Topical Report Transient and Safety Analysis Methodologies [2].

3.2.14 Volumes

The volume of the pebble bed region of the core is modeled. The fuel pebbles are assumed to occupy a defined region (cylindrical), because the cone region is not explicitly modeled. However, the pebble bed is modeled radially with the dimensions equal to that used in the VSOP model [36]. The cavity and RCCS volumes for the GOTHIC model correspond to the values in Reference [35]. Volume summaries used in the GOTHIC model are described in Table 3 of Reference [7].

3.2.15 Steam Generator

GOTHIC models the helium space. In the SG, the heat transfer is assumed to be uniform in the radial direction in the tube coil region, so only the axial subdivisions were defined regarding the number of axial nodes. Sensitivity studies were performed to determine the number of nodes required to achieve acceptable heat transfer performance. In addition, the water/steam side of the SG tube coils are modeled. Similar to the helium volume, the heat transfer is assumed to be uniform in the radial direction in tube coil region. Axially, this volume was divided into 30 axial nodes to match the axial regions for the helium side of the tube coils.

3.2.16 Reactor Cavity Cooling System

The RCCS model was developed and run using GOTHIC computer code. The RCCS model shown in Figure 2 of Reference [35], includes both trains and a representative volume for the reactor cavity and control rod drive mechanism (CRDM) regions of the reactor building. The RCCS layout is not complete and thus the equipment locations are approximated. The elevation and plan views of the layout used for the GOTHIC model development are shown in Figure 3 and Figure 4 of Reference [35], respectively. All pipe segments are either vertical or horizontal, changing directions at tees and elbows. The trains are laid out symmetrically but shown in mirror on the GOTHIC model diagram.

3.2.17 Reactor Building

A detailed 3-D model of the Reactor Building is developed using the GOTHIC computer code for the purpose of evaluating pressure and temperature response as well as helium, fission product and dust transport. The model demonstrates the diving bell concept of the helium pushing the air down to the basement of the reactor citadel, and out to atmosphere via the vertical shaft. Additionally, a (lumped volume) O-D model of the Reactor Building (RB) is developed for faster runs of sensitivity studies, which is adequate for building response. This representation could be included with a GOTHIC model of the Helium Pressure Boundary (HPB) to provide an integrated system response of the HPB and RB.

3.2.18 Flow Paths

Reactor cavity and RCCS Flow Paths correspond to those for the stand-alone RCCS GOTHIC model documented in [35]. Development of the remaining Flow Paths (i.e., steam/water side of SG) are provided in the Excel spreadsheets described in Section 4.1 of Reference [7] and listed in Table 4 of Reference [7].

3.2.19 3-D Connectors

3-D connectors that connect ends of RCCS outlet and inlet ring header ‘A', ends of RCCS outlet and inlet ring header ‘B', connection for CRDM regions, and connection for reactor cavity regions are described in the GOTHIC RCCS model documented in [35]. 3-D connectors that connect fuel channels 1 & 2, 2 & 3, 3 & 4, and 4 & 5 were defined for the GOTHIC reactor system model to connect the five radial fuel regions defined for the fuel pebble bed.

3.2.20 Conductors

A. **Conductors**

Table 6 of Reference [7] provide a summary of the conductors defined for the GOTHIC reactor system model. Conductor numbers with a “s” appended refer to subdivided conductors. The RCCS Conductors (Conductors 116s through 193s), Reactivity Cavity floor concrete conductor (194s), and the ring surrounding vessel head separating CRDM region from rest of reactor cavity conductor (195s) are documented in the GOTHIC RCCS model report [35]. The remaining conductors are documented in calculations described in Section 4.1 of Reference [7].

1. Pebble Bed Fuel Conductors

The geometry of conductors in GOTHIC is limited to slabs, cylinders, cylindrical shells and spherical. The process is described in Reference [7].

2. Side Reflector Conductors

The Helium gap between the outer graphite surface of the side reflector and the inner surface of the core barrel is not explicitly modelled as a separate volume. This is due to uncertainties concerning graphite expansion and fissuring due to radiation exposure along with uncertainties concerning Helium bypass around the graphite blocks into the Helium gap. Therefore, a composite conductor consisting of the side reflector graphite inner blocks, graphite outer blocks, Helium gap, and core barrel was defined. The side reflector support rings and wedges are not explicitly modelled. Therefore, the thermal conductivity of the helium gap in the side reflector conductor was adjusted upwards.

B. **Surface Options**

GOTHIC surface options define the heat transfer coefficients along with various heat transfer options and defined for both sides of the conductor. The 18 surface options are listed below and defined in Section 4.5.2 of Reference [7].

1. Surface Option 1 – Zero Flux
2. Surface Option 2 – Convective Reflector Inner
3. Surface Option 3 – Convective Reflector Outer
4. Surface Option 4 – Convective He Out Reflector
5. Surface Option 5 – Face Down
6. Surface Option 6 – Face Up
7. Surface Option 7 – Pebble Bed
8. Surface Option 8 – Adiabatic
9. Surface Option 9-13 – This surface options are defined in the RCCS model documented in [35]
10. Surface Option 14 – Rx Cavity Floor HTC
11. Surface Option 15 – Rx Cavity Floor
12. Surface Option 16 – SG Tube Interior
13. Surface Option 17 – SG Tube Exterior
14. Surface Option 18 – RPV Vertical Wall

3.2.21 Boundary Conditions

The GOTHIC boundary conditions are listed below with the Gothic Model Designator (e.g., 1F, 2F, 3P…). A brief description is provided in Section 4.6 of Reference [7].

- 1F and 2F – Supply 'A' & 'B' (RCCS SPs}
- 3P and 4P – Out 'A' and 'B' (RCCS SPs)
- 5P – Blowdown
- 6F – FW In
- 7P – Steam Out
- 8P – Press Control
- 9P – PRV Discharge
- 10P and 11P – Cavity Pressure and CRDM Pressure

3.2.22 Components

Information on the following valve and pumps modeled in GOTHIC are listed below with the Gothic Model Designator (e.g., 1V, 2V, 3V,…). A brief discussion is provided in Section 4.7 of Reference [7].

- Valves
    - 1V & 2V Defuel In & Out Isolation
    - 3V Break
    - 4V Press Control
    - 5V PRV
    - 6V Isolate Steam
    - 7V & 8V SG Check1 & SG Check2
- Pumps
    - Helium Circulators

4. Flownex Code Manuals and Qualifications

4.1 Flownex Code User Manuals

4.1.1 Flownex Simulation Environment General User Manual

The general user manual, which describes the functionality of the Flownex Simulation Environment this includes [37]:

- Descriptions of all the graphical user interface components.
- Methods to create a Flownex SE project.
- Chart and lookup table functionality.
- Methods to create dynamic simulations.
- Descriptions and examples of Flownex Simulation Environment utilities, this includes the Designer, Optimizer, Scheduler, Sensitivity Analysis, Excel Network Importer, Aspen Fluid Generator, Excel Reporting and Alarms.
- Procedures to create graphs, set-up compound components, log results.
- Methods to link third party software.

4.1.2 Flownex Library Manual

The Flownex library manual describes the functionality of Flownex library components this includes [38]:

- Descriptions of all components within the Flownex library.
- Component theory.
- Component input properties.
- Component results.
- Flownex charts and lookup tables.

4.1.3 Flownex Theory Manual

The Flownex theory manual describes the theory used by the Flownex solver this includes [39]:

- Governing equations theory
- Two phase flow theory
- Combustion modeling theory
- Dynamic modeling theory
- Slurry modeling theory

4.2 Verification and Validation

4.2.1 V&V Scope

It is important to distinguish between "Software V&V" and "Model and Data V&V". Software V&V is done to ensure the integrity and consistency of the simulation software. However, the accuracy of simulation results is not only dependent on the integrity of the software itself, but also on the way the software is used to setup a simulation model, as well as the component model input values for the specific simulation model. This area is covered by Calculation Model and Data V&V.

Calculation Model and Data V&V is unique to each simulation model and is the responsibility of the software user. Calculation Model and Data V&V includes verification that the calculation model was set up in such a way that all the important phenomena for the specific case are appropriately accounted for and ensuring that the input values for each component are accurate.

4.2.2 Vendor-Based V&V Efforts

Flownex is developed in an implemented NQA-1 quality management system at MTech Industrial (software design authority) which include:

- test plans and procedures,
- code reviews, user testing,
- automated testing and regression testing.

More than 1000 networks are tested between each release of Flownex to ensure that the new version of Flownex gives the same, or more accurate, results as compared to the previous version.

Verification and Validation (V&V) is a crucial part in the development of Flownex.

More than 40-man years of work was done on the V&V of Flownex. In the V&V process of Flownex, "Verification" is the process of ensuring that the controlling physical equations have been correctly translated into computer code or in the case of hand calculations, correctly incorporated into the calculation procedure. Validation is defined as the evidence that demonstrates that the code or calculation method is fit for purpose. This includes confirmation that the results from the verified model agrees with the benchmarks.

To ensure that all phenomena for each component in Flownex are validated for the various extremities is a comprehensive exercise. Furthermore, V&V of the individual components as well as integrated systems of components, for both steady-state and dynamic analysis are required.

V&V forms part of the overall Flownex development process and includes the verification activities that form part of the software engineering process, as well as all related verification that is done as part of the derivation and implementation of the theory for component models or model enhancements.

Validation of Flownex is performed by comparing the results of the implemented theoretical models in Flownex with benchmark data obtained from appropriate methods or sources such as analytical data, experimental data, plant data and data obtained from other codes such as Spectra, XNet and Star CD. To further enhance the Flownex V&V effort, and to ensure that the Flownex capabilities comply with the latest requirements on HTGR analysis, M-Tech is an active participant in international standard problems and conferences, like CRP-5, ICAPP and HTR-TN.

4.3 Existing Code Verification and Validation

4.3.1 XE00-T-S3ZZ-GLZZ-E-002102, Revision 2, 15-May-2022, X-energy Flownex Validation Plan – Service Receipt Inspection Report [40]

This report lays out the validation plan for Flownex [40]. The postulated accident cases for validation of Flownex are as follows:

G. Loss of Feedwater
H. Circulator Trip
I. Primary Side Depressurization Events
J. Steam Generator Tube Rupture (SGTR)
K. Seismic Events

For all the accident scenarios, the primary safety concern is dose. Since dose is not calculated by Flownex, parameters predicted by Flownex which have a significant impact on subsequent dose assessments should therefore be considered as candidates for validation. Such parameters are those which measure conditions under which radionuclide release becomes more probable, which could challenge a boundary to radio nuclide release, or which could transport radionuclides to the environment. With these considerations, candidates for figures of merit in validation are:

A. Fuel Temperature
B. Primary Coolant Temperature
C. Primary Coolant Flow
D. Primary Side Coolant Pressure
E. Reactor Power
F. Break Discharge (Primary and Secondary)
G. Reactor Pressure Vessel Wall Temperature
H. Reactor Cavity Temperature

4.3.2 XE00-N-RX-CORE-GL-GL-N-004842, May 20, 2022, Flownex Fuel Temperature Validation Exercise Report [41]

The Phenomena Identification and Ranking Table (PIRT) for the Xe-100 safety analysis is documented in Table 17 of Reference [42]. The phenomena modelled by Flownex have been extracted from the overall PIRT documented in an earlier version of Reference [42] and are presented in Table 1 of Reference [42]. There are no differences in the phenomena relevant to this validation exercise and Table 1 of Reference [41]

The validation method used compares the Flownex pebble temperature predictions against a mathematical solution for a stylized problem. Specifically, the computer program predictions were compared to the relevant solutions to standard or benchmark problems.

The stylized problem uses pebble dimensions and material properties for the Pebble Bed Modular Reactor (PBMR) design, which uses 60 mm pebbles, consisting of a 5 mm thick graphite shell and an inner fuel matrix containing TRISO particles. The pebble dimensions are the same as the pebble dimensions for the Xe-100.

The pebbles in the stylized problem differ from the Xe-100 pebbles with respect to the number of TRISO particles. The stylized problem assumes 15,000 TRISO particles per pebble, versus 19,000 TRISO particles in the Xe-100 pebble. The difference is dispositioned in Section 6.2.1 of Reference [41].

The stylized problem in Reference [44] uses boundary conditions which result in a change in pebble temperature from 0°C to approximately 160°C. To extend the range over which the validation is performed, the calculations applied to the stylized problem in Reference [44] have been extended to cover pebble temperature of up to approximately 1200°C. The extended calculations are described further in Section 2.0 of Reference [41].

Use of the stylized problem for validation of the Flownex internal pebble temperatures demonstrates that given the correct thermal properties of the pebble, and the correct boundary conditions to the pebble, an accurate prediction of the internal pebble temperature is obtained.

The validation exercise for the Flownex predictions of fuel temperature demonstrate that discretization of the pebble model strongly influences the bias in the code predictions. For the 6-node model applied in Xe-100 safety analysis, the accuracy in the code predictions is summarized in Table 5 and Table 6 of Reference [41]. The prediction bias is a strong function of fuel power and therefore at low powers that would occur following a reactor trip or fuel temperature induced power reduction, the prediction bias is not as significant. [46]

4.3.3 XE00-N-RX-CORE-GL-GL-N-004843 Revision 1, 05-August-2022, Flownex Reflector Structure Temperature Validation Exercise Report Using HTR-10 Benchmark at Steady State Conditions [45]

Comparison of the Flownex temperature predictions against the temperature measurements is an acceptable method of validation. This approach is consistent United States Standard ASME NQA-1.

As discussed in Section 1.1 of Reference [45], the phenomena to be validated are thermal conduction (HT1), convection (HT2) and radiation (HT3), and the figure of merit is temperature at selected locations in the HTR-10 core.

There are significant similarities in the HTR-10 and Xe-100 reactor with some differences, as summarized in Table 3 of Reference [45]. They are both a graphite moderated and helium-cooled pebble bed reactor, employing the similar fuel type of 60-mm diameter pebbles, consisting of a 5 mm graphite shell and an inner fuel matrix containing TRISO particles with a pebble packing fraction of 61%. The overall reactor structure is very similar such that the reactor core is a cylindrical shape with the active core in the center, surrounded by the top/bottom/side reflectors inside the Core Barrell (CB) and Reactor Pressure Vessel (RPV) that are air-cooled. The reactor and SG are housed in two separate steel pressure vessels, which are arranged side by side and connected to each other by a horizontal hot gas duct pressure vessel. The main differences are largely due to the reactor power: 10 MWt for the HTR-10 and 200 MWt for the Xe-100. So, the Xe-100 reactor has higher power, larger core size, and higher reactor power density (2.0 MW/m3 for HTR-10 versus 4.8 MW/m3 for Xe-100). Therefore, the Xe-100 operates at higher coolant flow rate and higher pressure in the primary system: 4.32 kg/s at 3.0 MPa(a) for HTR-10 versus 78.91 kg/s at 6.0 MPa(a) for Xe-100. However, the operational reactor inlet and outlet temperatures in the primary side are similar in both reactors: e.g., 250/700°C for HTR-10 versus 260/750°C for Xe-100. Therefore, References [43] and [47] determined that the HTR-10 benchmark data [48] should be relevant for the heat transfer V&V of the Flownex code based on the similarities in the reactor type, the operational range and geometrical features.

The HTR-10 and the associated experimental data were gathered at actual operating conditions, with a steady state temperature distribution within the pebble bed. The measured data included solid material temperatures adjacent to the pebble bed for a full power initial core. The HTR-10 operational data provides integral test results for V&V of the Flownex code since the data represents the entire pebble bed core with power and flow at nominal conditions or at a percent fraction of nominal conditions. The validation focused on the total heat transfer including conduction, convection, and radiation, providing axial and radial temperature profiles in the core and in the graphite.

The reference heat conduction calculation model used by the Chinese Institute of Nuclear and New Technology (INET) is shown in Figure 3 of Reference [38]: note INET is the host organization of the HTR-10 benchmark problem. It includes the fuel zone and non-fuel zone of the core, reflectors, carbon bricks, cavity, thermal shield, CB, RPV, RCCS, and coolant flow paths, etc. The 2D axi-symmetric heat conduction model in the R-Z geometry for the HTR-10 consists of 33 radial and 57 axial mesh points, and there are 44 different calculating regions. The pebble bed, reflectors and gas cavity are treated to be homogeneous media whose heat capacities can be determined according to the void fraction in these regions. The heat transfer of conduction, radiation and natural convection is considered in this model. This model is used as the basis to construct a Flownex model for HTR-10 validation exercise.

The validation exercises for the HTR-10 benchmark problem demonstrate that Flownex can predict the temperature distribution in the HTR-10 reactor core reasonably well within ±15% prediction error (224°C – 882°C) for steady-state operation at full power of 10 MWt. Generally, a coarse node model is employed for the HTR-10 reactor core model, and the discretization error is estimated to be approximately 9°C for predicting fuel pebble centerline temperature (coarse grid underprediction of temperature). The code bias for the side reflector temperatures was determined to be 29.8°C (overprediction), while the bias is -37.6°C (underprediction) for the top and bottom reflectors. The Euclidian difference between the experiment and simulation is 12.8%.

The Flownex HTR-10 benchmark study is compared against the previous two benchmark study results (INET and CEA): INET employed the modular code THERMIX, while CEA completed the benchmark problem using the CFD code ARCTURUS. Overall, the predicted temperatures by all three codes agree well with those measured temperatures within ±15% prediction errors (224°C – 882°C). The code bias and its variation in all three benchmark studies are similar.

The sensitivity studies on two key parameters (i.e., core bypass flow and natural convection cooling in air cavities) show that the extent of core bypass flow can have a relatively significant impact on predicting the maximum pebble centerline and surface temperatures in the reactor core during steady-state operation. Selection of a thermal conductivity model (e.g., correlation for the side reflector versus correlation for the top/bottom reflectors) does not have appreciable impact on the reactor core temperature prediction by Flownex during the steady-state operation since the steady-state condition is dominated by convection heat transfer, which transports orders of magnitude more heat than conduction, i.e. the heat transfer-based Peclet number is very high (= convective heat transport / diffusive(conductive) heat transport).

4.3.4 XE00-N-RX-CORE-GL-GL-N-006477 Revision 1, January 05, 2023, Initial Flownex Validation Exercise Report Using HTR-10 Benchmark – Prediction and Quantification of Code Accuracy for Reactor Power Transients [48]

As discussed in [43], a test at the 10 MWt HTGR Test Module (HTR-10) was identified as appropriate for Flownex validation to support Xe-100 safety analyses. The HTR-10 tests were published in the formal IAEA benchmark report [49]. The benchmark study included steady-state operation at full power (10 MWt) and transients following a LOFC without scram and single CRW without scram at partial load of 30% of full power (3 MWt). The transient tests for Loss of Forced Coolant (LOFC) without scram and single Control Rod Withdrawal (CRW) without scram were performed for which the measured relative reactor power is available. All-important phenomena (see Section 1.1 of Reference [50]) relevant to the LOFC and single CRW safety analyses were active in the transient tests. Although not measured, a key parameter in these tests is the fuel temperature since the reactivity feedback induced by the change in fuel temperature causes the reactor to shut down during the test. The tests are therefore suitable for indirectly assessing the fuel temperature reactivity feedback effect on the reactor power prediction.

The validation exercise to assess the Flownex predictions of the temperature in the graphite structures within the reactor vessel for steady-state operation at full power operation was documented in [83]. The report documents the validation exercise to assess the Flownex predictions of the reactor power transients following the LOFC without scram and single CRW without scram by using Flownex code version 8.14.0.4675 [51]. The work documented in this report follows the Flownex validation plan [43] with the exception that the neutronic input parameters from the M-Tech and AGREE-XE analysis were used.

The Flownex version applied in the validation exercise is version 8.14.0.4675 [51]. The key solver settings for the Flownex transient simulations are as follow:

- The maximum iteration number is set to 300.
- The convergence criterion is set to 1E-05 for steady state and 1E-03 for transient.
- The simulation timestep is set to 250 milliseconds (ms).

It is important to note that these solver settings are consistent with the Xe-100 safety analysis Flownex model [31].

Overall, the point kinetics model in Flownex can qualitatively predict the reactor power transient trends during reactor shutdown by inherent reactivity feedback. The overall code bias is less than 4% SSP (overprediction) with a variation in the bias of 5.5%. Also, Flownex can qualitatively predict the periodic power oscillations but overpredicts timing of the reactor power peaks after core re-criticality by about 620s, with higher magnitude by about 17% SSP.

For temporal convergence assessment, the simulations with timesteps of 500ms or less underpredict relative reactor powers by less than 1.2% and timing of their occurrence by less than 0.26%, but overpredict fuel average temperatures by less than 0.39% (3°C). With 250ms timestep, the underprediction is reduced to 0.6% for magnitude of the reactor power peak and 0.06% for timing of its occurrence, and the overprediction to 0.22% (2°C) for average fuel temperature.

Also, the current analysis results show the relatively large differences between the measured and predicted magnitude of the reactor power peaks and timing of their occurrences after re-criticality.

Possible causes may be uncertainties in the Flownex input parameters such as neutronic input parameters (e.g., reactivity feedback coefficients for fuel, moderator, reflector and xenon), material input parameters (e.g., thermal conductivity) and the boundary conditions (e.g., nominal reactor power for both LOFC and CRW test as discussed in Section 4.2i of Reference [50]. A sensitivity study varying these parameters should be performed with formal point kinetics parameters from VSOP when available.

4.3.5 Pebble Bed Temperature Prediction Validation with SANA

Flownex is supported by a large V&V program directed by MTech. Much of the activity is focused on exercises which align with verification, as opposed to validation. However, the program also contains a significant number of validation exercises against experimental test data. The validation exercises that are based on experimental data, or independent analytical solutions, are identified in Reference [40]. The phenomena active in the validation exercises are noted and the associated FOM relevant to the validation is also identified.

Validation supporting the accuracy of graphite pebble temperature predictions in a pebble bed in the absence of forced flow is available in References [52] and [53]. The validation is based on the SANA experimental test data and is summarized in the code applicability section of Reference [54]. The SANA test rig is illustrated in Figure 1 of Reference [40]. The validation spanned pebble temperatures from 60°C to 1200°C. Both steady state and transient predictions of pebble temperature are compared at various axial and radial locations in the bed. The validation addressed conductive and radiative heat transfer phenomena in the pebble bed using the Zehner-Schlünder correlation. The convective heat transfer phenomenon under natural circulation conditions, such as would occur in a loss of flow event, was present in the test and was captured in the validation using the Kugeler and Schulten correlation. The fluid resistance phenomenon was modelled using the KTA-Ergun equation (also known as the PBR equation), although only under natural circulation conditions. The temperature measurements were compared to the Flownex predictions at the pebble surface. Examples of plots for steady state and transient validation cases are shown in Figure 2 and Figure 3 of Reference [40]. The accuracy of the pebble temperature predictions was reported in terms of the maximum normalized point difference and Euclidian difference.

The range of pebble surface temperatures in the SANA experiments exceeds the range of maximum fuel average temperatures predicted in the depressurized loss of flow analysis reported in Reference [54]. The SANA experiments are therefore sufficient to fully cover the required range of applicability for the Depressurized Loss of Forced Coolant (DLOFC) with respect to validation of the heat transfer models in the pebble bed.

4.3.6 Blowdown Pressure Prediction Validation

Validation supporting the accuracy of Flownex pressure predictions in the Flownex nodes under pressurization and depressurization transients is available in Reference [55]. The accuracy of the pressure predictions in this validation exercise also illustrates that accuracy of the break discharge modelling.

The volume blowdown test rig is illustrated in Figure 4 of Reference [40]. The piping connecting the three tanks in the test rig contain sharp edged orifices and isolation valves. The tanks are initially at different pressures as shown in the test matrix reproduced in Table 3 of Reference [40]. As noted in the table end notes, the test pressures ranged between approximately zero (vacuum) and 1 MPa(a). The test matrix shown in the table contains cases in which the tanks are initially filled with different gases, allowing the Flownex models for the mixing phenomenon to be assessed. Each test case is initiated by simultaneously and quickly opening the valves between the tanks. The tests were simulated with Flownex using ideal gas models, including modelling of discharge through the orifices. Using ideal gas modelling is consistent with the approach applied in the safety analysis documented in Reference [54]. Discharge through an orifice is also used for break modelling in the safety analysis [54].

An example of a plot from Reference [55] comparing Flownex predictions to the test results are reproduced in Figure 5 of Reference [40]. The accuracy of the gas pressure predictions was reported in terms of the maximum point difference and Euclidian difference.

The test is appropriate for assessing break discharge modelling for breaks in which the gas behaves as an ideal gas. Gases approximate ideal gas behavior for low pressures or high temperature, relative to the critical pressure and temperature of the gas. The range of conditions covered in the validation and in the safety analysis is discussed further below.

The blowdown tests were performed at low, ambient temperatures. At the higher temperatures predicted in the safety analysis, the gases will continue to approximate an ideal gas. The maximum primary pressure in the blowdown tests was 1 MPa(a). For comparison, the maximum pressure in the SGTR case reported in Reference [54] was approximately 8.4 MPa(a). Although the range of pressures in the safety analysis exceeds the pressures in the blowdown test, the gas temperatures in the safety analysis are much higher than in the ambient temperatures applied in the test. At the high temperatures predicted in the safety analysis, the gases continue to approximate ideal gas behavior. To illustrate this, at 8.4 MPa(a) and 800°C the volume occupied by one gmol of steam was calculated based on steam tables. The volume was then used to calculate the pressure of one gmol of steam at 800°C using the ideal gas law. The pressure predicted by the ideal gas law was 2% higher than that from the steam tables. The small overprediction is conservative in the context of assessment of pressure boundary integrity. The deviation from the ideal gas law for the primary coolant, helium, will be smaller than that for the larger steam molecules.

4.3.7 Integrated System Validation with PBMM

Validation supporting the accuracy of temperature and pressure predictions in an integrated system is available in Reference [56]. The validation was executed using data from the Pebble Bed Micro Model (PBMM), which is illustrated in Figure 8 of Reference [40]. The PBMM is based on the Brayton power cycle. The nitrogen is compressed, heated, and then expanded in a series of turbines. It is then cooled to complete the cycle. The nitrogen temperatures ranged from approximately 25°C to 600°C. The nitrogen pressures ranged from approximately atmospheric up to approximately 300 kPa(a). Further details are provided in Reference [56]. Examples of plots from Reference [56] comparing Flownex predictions to the test results are reproduced in Figure 9 and Figure 10 of Reference [40]. The accuracy of the gas temperature and pressure predictions after each step of the cycle were reported in terms of the maximum point difference and Euclidean point difference [57].

4.3.8 Integrated System Startup Validation with PBMM

Validation supporting the accuracy of temperature and pressure predictions in an integrated system is available in Reference [58]. Reference [58] provides a comparison of Flownet predictions with measurements from the PBMM during startup. Note that Flownet was the original name for Flownex. The code was used to determine the point at which the PBMM was "bootstrapped" during startup. The point at which the system is bootstrapped is defined as the point during startup at which the circulation in the system becomes self-sustaining without the need for the startup blower.

The manner in which bootstrapping is achieved is described in Reference [58] and is reproduced here. Refer to Figure 11 of Reference [40] for the location of the components discussed in the following quote from Reference [58]: "During startup the inline valve (IV) is closed and the startup blower system (SBS) is used to circulate the gas through the cycle. The SBS is a positive displacement device, thus the flowrate remains essentially constant. Heat is then added to the gas in the heater. This energy is converted in the turbines into shaft work to power the compressors. For startup, the power to the heater is kept constant at about 180kW. As the system heats up, the outlet temperature of the heater rises as the inlet temperature rises and the pressure increase across the IV drops. The cycle spirals towards selfsustained circulation and the SBS is disengaged when the pressure drop over the valve is 0kPa. At this condition the cycle is said to have bootstrapped."

Reference [58] proposes that the exit temperature of the heater is the most important parameter with respect to defining of the bootstrap point because the energy that the turbines can deliver depends on the inlet gas temperature.

An estimation of the bootstrap temperature was made by calculating with Flownet the pressure increases over the IV as a function of heater outlet temperature. The results of the comparison are shown in Figure 12 of Reference [40].

4.3.9 Integrated System Nitrogen Injection Transient with PBMM

Validation supporting the accuracy of pressure predictions in an integrated system is available in Reference [59]. Reference [59] presents a comparison of Flownex with PBMM test results for a case in which nitrogen is injected transiently into the circuit. Nitrogen is injected into the cycle just upstream of the Pre-Cooler (see Figure 11 of Reference [40]) in order to increase the inventory of nitrogen in the cycle. As the mass of nitrogen in the cycle increases, the power output of the power turbine also increases. Before injection commences, the plant is run at steady state. The steady state conditions prior to the two transient tests that were performed are reproduced in Table 4 of Reference [40]. For the transient, nitrogen was injected into the cycle at a rate of 0.0227 kg/s for about a minute.

The transient results from Reference [59] are shown in Figure 13, Figure 14 and Figure 15 of Reference [40] for the case with an initial low pressure compressor suction pressure of 94 kPa(a). Reference [59] indicated that a probable cause of the difference between measured and predicted values in the transient was that the connecting pipe work between the turbo machines is very short. The flow profile of the gas entering the turbine is therefore not properly developed and the actual characteristics will be different from the characteristics as determined by the supplier and implemented in the model.

4.3.10 Pressure in Branched Piping Network Validation

Validation supporting the accuracy of the pressure predictions in a branched network of piping is available in References [60] and [61]. The test rig used in the experiment is part of the test rig in Section 3.2 of Reference [40]. The portion of the test rig used to collect the piping pressure data is the piping which exhausts Tank C, illustrated in Figure 6 of Reference [40]. Valve VAL-111 is quickly opened, creating a pressure wave in the downstream piping which is recorded at the location of the pressure taps shown in Figure 6. The pressures encountered during the tests ranged from a maximum of approximately 387 kPa(a) to atmospheric. Examples of plots comparing Flownex predictions to the test results are reproduced in Figure 7 of Reference [40]. The accuracy of the gas pressure predictions was reported in terms of the maximum point difference and Euclidian difference.

4.3.11 Mass Flow for Compressible Gas in a Piping Network Validation

A validation exercise for mass flow is summarized in Reference [62]. The network balancing experiment was to validate the Flownex node component for the mass flow balancing behavior of a compressible gas in a complex pipe network. A schematic of the test rig is shown in Figure 16 of Reference [40]. A fan supplying 0.045 kg/s of air at a total pressure of 102.36 kPa(a) is positioned at the inlet of plenum A. The flow network configuration was altered by opening and closing valves in each of the fifteen pipes resulting in different network flow configurations. The results for two flow configurations were reported in Reference [62]. Examples of plots comparing Flownex predictions to experimental measurements are reproduced in Figure 17 of Reference [40]. The accuracy of the gas pressure predictions was calculated in terms of the maximum point difference and Euclidian difference but only the maximum point difference was reported in Reference [62].

The tests were performed at low, ambient temperatures. At the higher temperatures predicted in the safety analysis, the gas will continue to approximate an ideal gas. This provides confidence in the mass flow predictions in a high temperature gas reactor.

4.3.12 Transient Temperature Predictions in Heat Exchanger Validation

A benchmarking exercise for temperature predictions in a pipe-in-pipe counter flow heat exchanger is summarized in Reference [62]. A schematic of the experimental setup is shown in Figure 18 of Reference [40]. The fluid in the experiment was water. The test was performed with both fluid streams at 26°C. The temperature of one stream was then increased from 26°C to 60°C. Additional test conditions are specified in Reference [62]. A plot comparing Flownex predictions to experimental measurements is shown in Figure 19 of Reference [40]. The accuracy of the water Temperature predictions for both streams exiting the heat exchanger was reported in terms of the maximum point difference and Euclidian difference in Reference [62]. The test exercises the convective and conductive heat transfer phenomena with respect to the prediction of fluid temperature.

4.3.13 Existing Validation Against Analytical Solutions

The 2021 Validation and Verification documentation [63] purchased by X-energy includes multiple cases in which the equations from Flownex are solved external to the code using the Engineering Equation Solver (EES). EES is an engineering analysis software developed by F-Chart Software for educational and commercial applications. It is a general equation solver that can solve thousands of coupled non-linear algebraic and differential equation and includes a library of thermodynamic and transport properties for hundreds of fluids and solid materials. In the majority of applications of EES, the exercises would be better categorized as verification since they confirm the correct implementation of the equations in the code by demonstrating that the code produces the same result as obtained when EES is used to solve the Flownex equations external to the code. Although these verification exercises have significant value since they demonstrate the correct function of the code, they are not discussed here. Rather, the exercises noted below are those which compare the code predictions to analytical solutions.

The following are Examples of existing validation against analytical solutions. Validation against analytical solutions is also reported in Reference [64] for assessment of the choked flow phenomenon and phenomena related to compressible flow in the pipe with heat transfer to the pipe.

- Step change in inlet temperature to a fixed adiabatic volume with constant inlet and outlet Helium mass flowrates [66]. The benchmark solution was not accessible for review but was stated to be documented in Reference [66]. The high-level description of the approach to the benchmark solutions provided in Reference [66] provides some confidence in the independence of the analytical solutions from Flownex. Predicted node temperature change and pressure change were the FOMs.
- Step change in heat transfer to a fixed volume with constant inlet and outlet Helium mass flowrates [65]. The benchmark solution was not accessible for review but was stated to be documented in Reference [66]. The high-level description of the approach to the benchmark solutions provided in Reference [66] provides some confidence in the independence of the analytical solutions from Flownex. Predicted node temperature change and pressure change were the FOMs.
- Fixed volume with constant helium mass inlet flowrate and no outlet flow [65]. The benchmark solution was not accessible for review but was stated to be documented in Reference [66]. The high-level description of the approach to the benchmark solutions provided in Reference [66] provides some confidence in the independence of the analytical solutions from Flownex. Predicted node temperature change and pressure change were the FOMs.
- Sudden mass injection into a tank [70]. Predicted node temperature change and pressure change were the FOMs.

4.4 Planned Flownex Code Verification and Validation

4.4.1 Flownex Simulations

The Flownex simulation will be performed with pebble properties and boundary conditions which are identical to those applied in the analytical solution. Flownex calculates a fuel temperature for each fuel node in the Flownex model. The fuel center node temperatures of both the fuel matrix and the fuel kernel are therefore available for direct comparison with the benchmark calculations. The average fuel temperature, used for determining the fuel temperature reactivity feedback, is the temperature of either the fuel matrix node temperature or the fuel kernel node temperature (as selected by the user) and is the volume-weighted-average over the heated region (see Section 15.2.3 of Reference [27]). These average temperatures are available for direct comparison with the weighted average benchmark temperatures. The Flownex model for the validation exercise has identified the fuel kernel nodes in the TRISO particles as the group of nodes for determining the fuel average temperature. [43]

4.4.2 Calculation of Bias and Variation in the Bias

The mean bias and the variation in the bias will be calculated based on the residual differences between the Flownex predictions and analytical values. The maximum normalized point difference and the Euclidian difference, as defined in Reference [71], will also be calculated for consistency with existing Flownex validation. Calculation of the bias is discussed in Reference [41].

5. GOTHIC Code Manuals and Qualifications

5.1 GOTHIC Code Manuals

5.1.1 GOTHIC Thermal Hydraulic Analysis Package User Manual

The user manual describes the functionality of GOTHIC Program and how to use it [67]:

- Describes the GOTHIC Environment
- Instructions for the preprocessor
- Instructions for the thermal-hydraulics program
- Instructions for the graphics program to model nuclear reactor containment and auxiliary buildings
- Procedure for setting up and input model, running the solver, and accessing the results

5.1.2 GOTHIC Thermal Hydraulic Analysis Package Technical Manual

The GOTHIC technical manual describes the equations and models in GOTHIC S and the numerical methods used to solve them [68]:

- Modeling Overview
- Governing equations theory
- Models for Engineered Safety Equipment
- Models for Thermal Conductors
- Models for Interfacial Mass, Energy and Momentum Transfer, and Wall Heat Transfer and Drag
- Mechanics and Assumptions for the Multiple Drop Fields
- Models for Viscous and Turbulence Stress and Diffusion
- Models for Hydrogen Burn
- Model Theory for Neutron Point Kinetics
- Description of Various Fluid Property Functions
- Description of Control Models
- Description Finite Volume Formulations
- Description of the Numerical Methods Used to Solve the Equations

5.2 Verification and Validation

5.2.1 V&V Scope

It is important to distinguish between “Software V&V” and “Model and Data V&V”. Software V&V is done to ensure the integrity and consistency of the simulation software. However, the accuracy of simulation results is not only dependent on the integrity of the software itself, but also on the way the software is used to setup a simulation model, as well as the component model input values for the specific simulation model. This area is covered by Calculation Model and Data V&V.

Calculation Model and Data V&V is unique to each simulation model and is the responsibility of the software user. Calculation Model and Data V&V includes verification that the calculation model was set up in such a way that all the important phenomena for the specific case are appropriately accounted for and ensuring that the input values for each component are accurate.

The Gothic Qualification Report [74] includes benchmarks for a diverse set of physical phenomena to experimental data. The end user to determines which experiments are applicable and will supplement the information with additional benchmarking as needed to cover any gaps identified for the intended application. [69]

5.3 Existing GOTHIC Code Verification and Validation

For more details on the GOTHIC Verification and Validation Plan refer to Reference [69].

5.3.1 Reactor Cavity Cooling System Validation

The Argonne National Laboratory (ANL) Natural convection Shutdown heat removal Test Facility (NSTF) has initiated testing of a RCCS which is driven entirely by natural circulation. The test rig is air cooled, similar to the current RCCS design for the Xe-100. The test rig is focused only on the RCCS operation and is not suitable for validating prediction of heat losses from the reactor pressure vessel to the RCCS. [69]

5.3.2 Reactor Building Validation

Some examples of tests included in the current Qualification Report at Battelle-Frankfurt Model Containment (BFMC) Test, hydrogen mixing tests at the Hanford Engineering Development Laboratory (HEDL) Containment Systems Test Facility, Nuclear Power Engineering Corporation (NUPEC) Test, Heissdampfreaktor (HDR) full-scale multicompartment containment experiments, and International Standard Problem-47, which includes the ThAI, MISTRA, and TOSQUAN tests. These provide a comprehensive basis for GOTHIC’s accuracy and applicability for modelling natural convection, mixing, and stratification. [69]

5.4 Planned GOTHIC Code Verification and Validation

5.4.1 Helium Pressure Boundary

A. Pebble Bed Temperature Prediction Validation with SANA

The Gothic Validation for the Pebble Bed Temperature Prediction will be performed consistent with Flownex [39]; which, was based on the SANA experimental test data. Please refer Section 4.3.5 for more details.

Validation supporting the accuracy of graphite pebble temperature predictions in a pebble bed in the absence of forced flow can be based on the (Selbsttätige Abfuhr der Nachwäre) SANA experimental test data [40]. The experimental data covers pebble surface temperatures from 60°C to 1200°C. Both steady state and transient predictions of pebble temperature are compared at various axial and radial locations in the bed. The SANA experiments would provide validation of the heat transfer models in the pebble bed for both normal operating conditions and loss of forced circulation (LOFC) transients. The SANA experimental program has been widely used as a benchmark to support code developments intended for thermodynamic analysis of pebble bed HTGRs.

The main component of the test facility is a cylindrical steel vessel containing a pebble bed 1.5 m in diameter and 1.0 m high. The SANA experiments selected for validation used 60 mm graphite pebbles, corresponding to the pebble size in the Xe-100, with either helium or nitrogen as the coolant. For the cases used in this validation, the vessel was heated by an electrical heating element located along the central axis of the pebble bed. The heat could be applied in different locations (i.e., top half, bottom half, or full length) between tests. The figure of merit for the comparison would be the pebble temperature, measured radially at elevations close to the bottom, center and close of the top of the pebble bed.

There is also a series of tests with the same SANA experimental setup as discussed above, but with a step change increase or decrease in heating element power, allowing the transient response of the pebble bed to be assessed. Two validation cases were performed with nitrogen and two with helium. Pebble temperatures at the center height of the pebble bed and at seven radius points between the inner radius and the outer radius of the bed were compared over the sixty-hour transients.

A GOTHIC model of the SANA facility will be constructed, and benchmarks will be performed to the tests that used 60 mm graphite pebbles with helium coolant. [69]

B. Blowdown Pressure Prediction Validation

The Gothic Validation for the Blowdown Pressure Prediction will be performed consistent with Flownex [40]; which, was based supporting the accuracy of pressure predictions for both pressurized and depressurized transients. Please refer Section 4.3.6 for more details.

Validation supporting the accuracy of pressure predictions for both pressurized and depressurized transients is available in Reference [55]. The accuracy of the pressure predictions in this validation exercise also illustrates that accuracy of the break discharge modelling.

The piping connecting the three tanks in the test rig contain sharp edged orifices and isolation valves. The tanks are initially at different pressures ranging between approximately zero (vacuum) and 1 MPa(a). The tanks are initially filled with different gases, allowing for the mixing phenomenon to be assessed. Each test case is initiated by simultaneously and quickly opening the valves between the tanks.

The test is appropriate for assessing break discharge modelling for breaks in which the gas behaves as an ideal gas. Gases approximate ideal gas behavior for low pressures or high temperature, relative to the critical pressure and temperature of the gas.

A GOTHIC model of the facility will be built for benchmarking transient pressure response. [69]

C. Reactor Temperature and Power Validation using HTR-10

The 10 MW High Temperature Gas-Cooled Reactor Test Module (HTR-10) is an integrated effects test appropriate for validating GOTHIC to support the Xe-100 safety analysis. HTR-10 is a graphite moderated and helium-cooled pebble bed reactor. The reactor and steam generator are housed in two separate steel pressure vessels, which are arranged side by side and connected to each other by a horizontal hot gas duct pressure vessel. These three vessels make up the primary pressure boundary of the HTR-10.

The HTR-10 test was published in a formal IAEA benchmark report [49]. Steady state temperature measurements are available at various locations in the graphite structures within the reactor vessel. The active phenomena which directly influence the measurements include those related to conductive, convective and radiative heat transfer. The measurements are indirectly related to the fuel temperature and primary coolant temperature figures of merit. Accurate prediction of the temperatures of the internal reactor structures provides confidence in the accuracy of the predicted fuel and coolant temperatures within the reactor.

A transient loss of flow test without scram was performed for which the measured reactor power is available. All-important phenomena relevant to the loss of flow safety analysis is active in the loss of flow transient test. Although not measured, a key parameter in the test is the fuel temperature since the increase in fuel temperature causes the reactor to shutdown during the test. The test is therefore suitable for assessing the fuel temperature reactivity feedback effect on the reactor power prediction.

A transient test was also performed for a CRW without scram. All-important phenomena relevant to the CRW safety analysis are active in the CRW transient test. As noted above, although not directly measured, a key parameter in the test is the fuel temperature since the increase in fuel temperature causes the reactor to shutdown during the test. The test is therefore suitable for assessing the fuel temperature reactivity feedback effect on the reactor power prediction.

VSOP has already been used to derive the appropriate input data for the point kinetics model to support a comparable Flownex validation case. Although GOTHIC's point kinetics model is not currently used in any of the Xe-100 models, it could be applied as part of the HTR-10 benchmark or the power vs. time profile from Flownex or VSOP could be provides as an input to the GOTHIC model.

A GOTHIC model will be built of the HTR-10 reactor vessel based on the description of the HTR-10 reactor vessel provided in Chapter 2 of Reference [49] and Chapter 4 of Reference [72]. The GOTHIC model will incorporate correlations and modelling approaches that are consistent with those applied in the Xe-100 analysis. A steady state will be executed using the boundary conditions supplied in Reference [49] that are applicable to the steady state test. The temperature predictions at the test measurement points will be documented for comparison with the test data. The initial steady states for the transient tests will be established based on the conditions prior to the transient test reported in Reference [49]. The Transient Loss of Flow without Scram Test and Transient CR withdrawal without Scram Test will be simulated with GOTHIC. [69]

5.4.2 Reactor Cavity Cooling System

A. Reactor Cavity Temperature validation (HTTR-RCCS Mock-up)

The High Temperature Engineering Test Reactor (HTTR) of the Japan Atomic Energy Research Institute (JAERI) is a graphite-moderated and helium-gas-cooled reactor. The RCCS for the HTTR was tested via a mock-up. The test rig and the test results for the HTTR-RCCS mock-up are described in Section 4.1.1 of Reference [73].

The HTTR-RCCS mock-up tests are relevant to the Xe-100 because they provide a means of validating post-accident heat transfer from the reactor pressure vessel wall, through the reactor cavity, to the RCCS piping. In accident analysis in which forced circulation is lost, long term heat transfer from the reactor pressure vessel is only through the reactor cavity by means of radiative and convective heat transfer phenomena. The HTTR-RCCS mock-up tests provide temperature data for the reactor pressure vessel walls and the RCCS cooling panels for various heat loads. While HTTR uses a water-cooled RCCS (with forced circulation via pumps) the Xe-100 uses an air-cooled RCCS. This has no bearing on this validation model because the heat transfer across the reactor cavity to the RCCS is being validated. [69]

6. Quality Assurance

X-energy developed and implemented a Quality Assurance Program 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.

6.1 QAP 3.1, Control of Design & Development Procedure

Quality Assurance Procedure (QAP) 3.1 [75] in conjunction with other referenced X-energy procedures, implements the requirements for design control in the X-energy Quality Assurance Program Description. 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.

6.2 QAP 3.2, Technical Analysis Procedure

QAP 3.2 [76] establishes the process, responsibilities, and requirements for performing and documenting Technical Analyses, including both work planning and executing an analysis.

6.3 QAP 3.6, Software Procedure

The purpose of QAP 3.6 [77] 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.

6.4 QAP 3.9, Computer Program Technical Evaluation & Acceptance Procedure

The purpose of QAP 3.9 [78] is to ensure that Software used for design, analysis and supporting activities that impact 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-1. 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.

6.5 QAP 3.10, Software V&V for Design & Safety Analysis Procedure

QAP 3.10 [79], 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.

6.6 QAP 3.11, Software Problem Reporting and Resolution Procedure

The purpose of QAP 3.11 [80] is to define the process for software problem reporting and resolution for the Xe-100 program.

6.7 QAP 3.14, Software Configuration and Change Control Procedure

The purpose of QAP 3.14 [81] 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 do 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.

7. Conclusions

7.1 Flownex Codes

The analysis of thermal-fluid networks is based on the numerical solution of the governing equations of fluid dynamics and heat transfer. Flownex solves the partial differential equations for mass, momentum, and energy conservation to obtain the mass flow, pressure, and temperature distributions throughout Xe-100. Flownex is used to model the Xe-100 for use in evaluating transients for safety analysis. Based on the assessment of Flownex conducted to date, it is concluded that the code is applicable for its intended purpose.

7.2 GOTHIC Codes

The purpose of the GOTHIC model is to capture the long term response (30 days) of the HPB and provide assessment of the fuel and solid material temperatures under different accident scenarios, including PLOFC, DLOFC, CRW and SGTR. Based on the assessment of GOTHIC conducted to date, it is concluded that the code is applicable for its intended purpose.

8. Cross References and References

8.1 Cross References and References

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Appendix A Flownex Base Model Theory Overview

Flownex base model theory specific detail is located in Reference [6].

A.1. Introduction

Below is a brief history of the development of Flownex in South Africa. In the USA, Flownex is developed within ASME NQA1-2015.

Flownex is developed within an ISO 9001 and ASME NQA1 accredited quality system. Flownex is also validated and verified within this quality system. As can be seen below, Flownex has been audited based on mostly nuclear standards and quality procedures, which are some of the most stringent regulations.

- In 2007 the National Nuclear Regulator (NNR) of South Africa reviewed the Flownex Software V&V status and found it to be acceptable to be used to support the design and safety case for the PBMR.
- NQA1:2008 accredited by Westinghouse USA.
- 2009: ISO 9001 audited and approved.

A.2. GOVERNING EQUATIONS

Three conservation laws govern the transport processes occurring in thermal-fluid networks and transport processes in general. These are:

- Conservation of mass.
- Conservation of momentum.
- Conservation of energy.

These are fundamental laws of nature that are universally applicable in natural and man-made systems. The conservation laws are described in the language of mathematics by partial differential equations. The differential equations form a system of coupled equations that must be solved using a suitable solution algorithm. An essential part of the solution process is the specification of realistic boundary conditions for the network and in the case of dynamic simulations, realistic initial conditions. Additional equations are introduced to complete the system of governing equations mathematically, i.e., there must be as many independent equations as there are unknown variables. The variables that Flownex solves are usually referred to as the dependent variables and are variables of particular interest in the design and analysis of thermal-fluid networks. The variables are, amongst others, flow velocity, pressure, and temperature. Flownex presents the calculated distributions of these variables and many additional variables in a formatted result file and in additional graphical format in the event of a dynamic (transient) simulation. In addition to the basic dependent variables, Flownex also provides results for the heat transfer rates for certain elements, which are calculated from the solution of the dependent variables. Additional equations are solved for the advanced functionalities of Flownex, for example the shaft dynamics of turbomachinery and the functioning of controllers.

When dealing with the analysis of transport processes, a reference frame is required to use as a basis from which the governing equations can be constructed and described. Two popular reference frames are commonly used in fluid dynamics namely the Lagrangian reference frame and the Eulerian reference frame.

The conservation equations formulated in the Lagrangian reference frame are not particularly suitable for modelling thermal-fluid systems. However, they do explain the physical meaning of the equations in easy-to-understand mathematical terms. The equations are therefore converted to the Eulerian reference frame, which is more suitable for numerical analysis.

The specific General and Flownex Governing Equations can be found in Reference [39].

A.3. Solution Procedure

Flownex employs a state-of-the-art implicit pressure correction solution algorithm that results in fast and accurate simulations. The various steps involved in the implicit pressure correction algorithm [11] are listed below in the order of execution.

1. Guess initial node pressures
2. Calculate mass flows using relationships
3. Test for continuity at all nodes
4. Adjust pressures to ensure continuity at all nodes
5. Update mass flows using new updated pressures
6. Repeat 1 to 5 until convergence
7. The energy equation
8. Repeat 1 to 7 until convergence
9. Move to next time step and repeat 1 to 8 (Only for transient simulations)

Flownex employs a segregated solution algorithm in which the different governing equations and additional closure equations are solved sequentially. This allows the user to control various aspects of the solution procedure through relaxation parameters, adjusting the number of iterations and the convergence criteria for the solution.

Discussion of two-phase flow can be found in Chapter 3 of Reference [6] and Discussion of Two Phase Flow with Incondensable can be found in Chapter 4 of Reference [6].

Two-phase flow is a phenomenon that occurs mainly as a result of one of the following processes:

- Flashing
- Boiling
- Condensation

The three processes mentioned are typically present in a one-fluid two-phase flow system. A pressure-enthalpy diagram showing the different processes is depicted in Figure 3.1 of Reference [6].

Flashing takes place when the pressure of a fluid is reduced (adiabatic process) below the saturation pressure for the given temperature. Boiling occurs when heat is added to the system and the fluid reaches the liquid saturation temperature for the given pressure (illustrated for isobaric process). Condensation takes place when the vapor is cooled down and the vapor saturation temperature is reached (illustrated for isobaric process). Two-phase flows are governed by mass, momentum, and energy conservation equations. A number of constitutive equations are also used to calculate the two-phase friction factor and heat transfer coefficient. The final process mentioned takes place when two immiscible fluids are mixed. Such two-phase flow systems are encountered in the oil industry or when air from the atmosphere enters into a liquid filled system from a leaking pipe or open valve.

Incondensable gas mixtures are defined as mixtures that contain both an incondensable gas and a liquid-vapor mixture. A typical example is the situation found in condensers/evaporators with air entrainment. In this case, the air would be the incondensable gas with the saturated steam present in liquid and vapor phases.

- Thermo-dynamic conditions determine in which of three possible states the mixture exists:
- Liquid-vapor mixture with incondensable gas
- Super-heated vapor containing only vapor and the incondensable gas

This document describes the equations used to calculate various thermo-dynamic properties of such a mixture. The properties that can be calculated are the partial pressures, enthalpy, mixture density, temperature, vapor mass fraction and entropy.

The remaining Chapters of Reference [6] discuss:

- Chapter 5 – Combustion Modeling
- Chapter 6 – Dynamic Modeling
- Chapter 7 – Forces
- Chapter 8 – Slurry and Non-Newtonian Flow
- Chapter 9 – Exergy

Appendix B GOTHIC Base Model Theory

GOTHIC base model theory specific detail is in References [67] and [68].

B.1. Introduction

GOTHIC is an integrated, general purpose thermal-hydraulics software package for design, licensing, safety and operating analysis of nuclear power plant containments, confinement buildings and system components. Applications of GOTHIC include evaluation of containment and containment sub-compartment response to the full spectrum of high energy line breaks within the design basis envelope and a wide variety of systems evaluations involving multiphase flow and heat transfer, gas mixing and other thermal hydraulic behavior. Applications may include, but are by no means limited to, pressure and temperature determination, equipment qualification profiles and inadvertent system initiation, and degradation or failure of engineered safety features. As a general purpose tool, GOTHIC can be used for a wide variety of plant operations support issues involving single and multiphase heat transfer and fluid flow provided that the application is consistent with the underlying physical basis and assumptions and the- code validation basis.

B.2. GOTHIC BASE MODEL THEORY

B.2.1. Governing Equations

The conservation equations solved by GOTHIC are written in integral form. This form is closely related to the finite volume numerical method used to solve the equations. The equations are written for a fixed volume bounded by area. The volume may be the entire region of interest or a portion of the total volume, although, in practice, the volume corresponds to one of the finite volumes of the computational grid. The specific equations for the following parameters can be found in the EPRI document titled "GOTHIC Thermal Hydraulic Analysis Technical Manual" [68] dated May 2022. The equations are for multidimensional analysis with simplifications made for lumped parameter analysis and junction flows.

- Mass Conservation
- Energy Conservation
- Momentum Conservation
- Drop Number Concentration and Surface Area
- Reduced Equations for Lumped Parameter Volumes
- Liquid Components
- Equations of State
- Source Terms
- Interphase Source Terms
- Laminar Leakage
- Turbulent Leakage

- Drop Leakage
- Tracer Conservation

B.2.2. Flow Paths

Flow paths, also referred to as junctions, are used to hydraulically connect any two computational cells. The flow paths may represent doorways, pipe or instrument penetrations, pipes, duct work, and so forth. These hydraulic connections are in addition to the vertical and lateral flow connections for 1-, 2- and 3-dimensional meshes that give the sub volume face velocities. Momentum equations for the vapor, droplets and liquid are solved for each junction. The junction momentum equations are consistent with those used for the sub volume face velocities and 3D Connectors except that the effects of viscous and turbulent shear are not included. The following parameters are calculated to adequately represent the movement within the flow path (between each computational cell). The equations and information for these parameters are found in Reference [68].

- Momentum Conservation
- Pool Height and Vapor Fraction
- Junction Volume Fractions
- Flow Path Gravity Heads
- Momentum Transport
- Intrinsic Pressure Loss
- Critical Flow Model
- 3D Flow Connectors

B.2.3. Engineered Safety Equipment

The base model of GOTHIC contains a variety of equipment that may exist within containment and is part of the operation for safety and emergency response. These are referred to components and include the list equipment listed below. Each component group is treated as a single component (e.g., pumps and fans). For this example, if a pump is referred to, it would also include the fan; however, only one component class can be located on a flow path. A fan, recombiner, and valve can be assigned to a flow path.

- Pumps and fans
- Valves and doors and vacuum breakers
- Pressure Relief Valves
- Heat exchangers
- Spray nozzles
- Coolers and heaters
- Volumetric fans
- Filter Systems

- Dryer/Demisters
- Filters
- Charcoal Filters
- Recombiners
- Ignitors

Reference [68] Sections 6.1 – 6.13, contains technical descriptions and mathematical models for each component listed. The mathematical models define the mass, momentum and energy source terms associated with the component.

B.2.4. Conductors

Conductors are used to model heat sinks such as concrete walls and floors and structural steel in containment buildings. Conductor geometries that can be modeled in GOTHIC are flat plates, hollow tubes, and solid cylinders with conduction in one or two dimensions with an optional ice layer on the surface. Thermal radiation exchange among conductor surfaces can also be modeled. The applicable portions of the base model for the Xe-100 are derived from the conduction equations in Section 7.1 of Reference [68] and utilizing heat transfer coefficients for the wall surface heat flux equations in Section 7.2 of the same reference.

B.2.5. Interface Source Terms

The interface source terms are calculated by performing mass, momentum, and energy balances for the interfaces. It is assumed that there is no storage of any of these quantities at the interfaces. Seven interface combinations are considered; liquid/vapor, drops/vapor, ice/vapor, ice/liquid, drops/liquid, mist/vapor, and mist/(drop or liquid). Interchange at these interface combinations is due to heat transfer with a corresponding phase change and mechanical interaction resulting in interfacial mass and momentum transfer.

The interface models described in this section are applicable, as needed, to subdivided volumes, lumped parameter volumes and junctions. For purposes of interface heat, mass and momentum transfer, each of the drop fields is treated as a separate phase. The interface models that may be utilized are listed below. Details of the equations used for each of these interface models are found in Section 8.1 of Reference [68].

- Flow Regimes
- Interfacial Area
- Interfacial Drag
- Liquid/Vapor and Drop/Vapor Heat and Mass Transfer
- Mist/Vapor Heat and Mass Transfer
- Liquid Component Source Terms
- Ice

- Drop/Liquid Phase Transformation
- Combined Interface Source Terms

B.2.6. F. Wall Source Terms

Wall source terms include convection and radiation heat transfer, condensation and boiling at the wall and friction and orifice drag. The equations used are discussed in Section 9 of Reference [68] along with the correlations (see below) derived from the equations used to calculate these source terms.

1. Mass and Energy Source Terms

The available options for calculation of the energy source for each conductor available for selection by the user are:

b. Convection
    - Natural Convection
    - Forced Convection
    - Specified Convection

c. Radiation

d. Condensation
    - Direct Condensation
    - Tagami Blowdown
    - Specified Condensation
    - Mist/diffusion layer model

e. Specified Revaporization

f. Built In Heat Transfer Package

g. Thermal Boundary Conditions

2. Momentum Source Term

The Momentum source terms include the fluid drag due to wall friction and losses through orifices or across obstructions and calculated as described in Section 9.2 of Reference [68]. Additional drag can be caused by the following.

- Floor Drag
- Wall Adhesion
- Compressibility Option which includes orifice vapor phase compressibility, liquid and drop phase flashing, and nozzle vapor phase compressibility
- Steam Injection into a Condensing Bowl

B.2.7. Drop Fields

Any number of drop fields can be used to track and simulate drop behavior in a GOTHIC model. It is assumed that the drops for a given field in a given cell have a log normal size distribution and that within each cell, each field is characterized by an average drop diameter and the geometric standard deviation (GSD).

Depending on the field option settings, the drop behavior may include agglomeration and deposition by various mechanism, evaporation, condensation and entrainment from pools and films. The combined effect of these mechanisms causes the average drop diameter and the GSD to change as the transient progresses.

If the drops are from a single source, (e.g., spray system or break), a single drop field is typically adequate to model the drop behavior. If the drops are from multiple sources (e.g., drops from a break followed by drops from spray system), the difference in the average diameter of the two drop source makes it difficult to model the drops with a single field. When the two fields are combined together in a single, the average drop diameter results in drop behavior that is not truly representative of either field. In these cases, multiple drop fields are recommended. Multiple drop fields may also be useful where there is interest in tracking the distribution of a constituent with a particular drop source combined with other drop sources.

The assumptions for the drop models, and equations for the size distribution, solid component in droplet fields, drop diameters, drop drag coefficients, drop terminal velocity, source terms, and size distribution integration formulas are found in Section 10 of Reference [68].

B.2.8. Stress and Diffusion Terms

The stress tensor in the surface stress term of the momentum equation includes the effects of static pressure, viscous shear and turbulent diffusion of momentum. The components of the mass and energy diffusion that are due to turbulence are closely related to that for turbulent momentum diffusion. Inclusion of both molecular and turbulent diffusion in the mass diffusion coefficient and in the energy diffusion coefficient is optional in GOTHIC.

The fluid-fluid shear stress terms are applied only to the continuous phase. When the flow is primarily drop/vapor, the stresses are applied to the vapor phase. When the flow is bubble/liquid, the stresses are applied to the liquid phase. The stress tensors are calculated for each phase but are ramped to zero when the phase becomes discontinuous.

Details of specific equations and on the following stress and diffusion terms are found in Sections 11.1 through 11.6 of Reference [68].

- Turbulence Effects
- Mixing Length Model
- Two-Equation Model
- Strain Rate Norm
- Liquid Component Diffusion
- Numerical Stabilizers

B.2.9. Fluid Properties

Formulas for calculation of ice, liquid, steam, gas component, and vapor mixture properties and the corresponding table data are coded in GOTHIC S. References are provided for most of the properties. There are a few properties, however, for which no specific reference is available. In most cases where a reference is not available, the properties in question were built into COBRANC when it was acquired as the starting point for the development of FATHOMS and GOTHIC S, as outlined in Section 1.3 of Reference [68]. Therefore, these property correlations and tables in GOTHIC S have received considerable scrutiny over many years of applications. In addition, as part of the GOTHIC Design Review, many property correlations and tables in GOTHIC S were compared to other sources of property data and found to agree quite well.

Sections 14.1, 14.2 and 14.3 of Reference [68] describe the fluid property equations coded into GOTHIC S for the solid (ice), liquid (water) and vapor (steam) phases of water. GOTHIC also includes an option to use table based properties for water, as well as alternate fluids. Table based properties are discussed in Section 14.4 of Reference [68].

B.2.10. Finite Volume Equations

The finite-volume equations for the mass, energy and momentum balances are written such that they may be solved on a rectangular mesh or in lumped parameter form. On a rectangular mesh, the full three dimensional `form of the momentum equations is solved.

Mass and energy balances are maintained for each finite volume. Lumped parameter finite volumes are defined by the total free volume, height, and hydraulic diameter. The actual shape of a lumped volume is unspecified. Subdivided volumes can be divided into any number of finite volumes on a rectangular grid. The finite volumes in a subdivided volume are defined by the x, y and z dimensions, and volume and area porosity factors. The porosity factors are values between zero and 1. The volume porosity is the fraction of the cell volume occupied by the fluid. The area porosity factors specify the fraction of each cell face area that is open to flow.

The formulation of the finite volume equations for the mass, energy, and momentum balances is described in Section 16 of Reference [68]

B.2.11. Solution Algorithm

GOTHIC models three types of phases (vapor, liquid and drops) and one, two or all three can be included in a model. The list below describes the solution procedure when all three phases are included. When one or more phases is excluded, the corresponding equations are not solved and the coefficients relating to the missing phases are zero. Details on the following solutions are found in Section 17 of Reference [68].

- Solution of the Momentum Equations
- Linearization of the Mass and Energy Equations
- Solution of the Pressure Matrix
- Unfolding of Primary and Secondary Variables
- 3D Flow Connectors
- Network Solutions

- Leakage
- Oscillating Flow Control
- Time step Control
- Damping and Ramps
- Phase Change Rate Limits
- Variable Limits