UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 20, 2025

Mr. Matt Thomas
Licensing Manager, X Energy, LLC
530 Gaither Road, Suite 700
Rockville, MD 20850

SUBJECT: FINAL SAFETY EVALUATION FOR X ENERGY LLC'S TOPICAL REPORT, "XE-100 LICENSING TOPICAL REPORT REACTOR CORE DESIGN METHODS AND ANALYSIS," REVISION 2 (EPID L-2024-TOP-0012)

Dear Mr. Thomas:

By letter dated March 12, 2025 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML25071A399), X Energy, LLC (X-energy) submitted Revision 2 of its topical report (TR), "Xe-100 Licensing Topical Report Reactor Core Design Methods and Analysis" (CDMA) to the U.S. Nuclear Regulatory Commission (NRC) staff for review. This TR describes the methods and computer codes used to support the Xe-100 reactor core design and preliminary analysis.

The NRC staff's final safety evaluation (SE) for X-energy's CDMA TR is enclosed. The NRC staff concluded that the TR is acceptable, subject to the limitations and conditions documented in the SE. The NRC staff requests that X-energy submit an accepted version of the CDMA TR within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE.

If you have any questions, please contact Denise McGovern at (301) 415-0681 or via email at Denise.McGovern@nrc.gov.

Sincerely,

Signed by Philpott, Stephen
on 08/20/25

Stephen Philpott, Acting Chief
Advanced Reactor Licensing Branch 2
Division of Advanced Reactors and Non-Power
Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Project No. 99902071

Enclosure:
As stated

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SUBJECT: FINAL SAFETY EVALUATION FOR X ENERGY LLC'S TOPICAL REPORT, "XE-100 LICENSING TOPICAL REPORT REACTOR CORE DESIGN METHODS AND ANALYSIS," REVISION 2 (EPID L-2024-TOP-0012) DATED: AUGUST 20, 2025

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FINAL SAFETY EVALUATION FOR X ENERGY LLC'S TOPICAL REPORT, "XE-100 LICENSING TOPICAL REPORT REACTOR CORE DESIGN METHODS AND ANALYSIS," REVISION 2 (EPID L-2024-TOP-0012)

SPONSOR AND SUBMITTAL INFORMATION

Sponsor: X Energy, LLC (X-energy)

Sponsor Address: X Energy, LLC
530 Gaither Road, Suite 700
Rockville, MD 20850

Project No.: 99902071

Submittal Date: March 12, 2025

Submittal Agencywide Documents Access and Management System (ADAMS) Accession No.: ML25071A399 (package)

Supplement and Request for Additional Information (RAI) response letter Date(s) and ADAMS Accession No(s): N/A

Brief Description of the Topical Report:

On April 8, 2024, X Energy, LLC (X-energy) submitted Topical Report (TR) 006889, "Xe-100 Licensing Topical Report Reactor Core Design Methods and Analysis," (hereafter referred to as the CDMA or TR), Revision 1, for review by the U.S. Nuclear Regulatory Commission (NRC) staff (ADAMS Accession No. ML24099A183 (package)). X-energy submitted Revision 2 of this TR on March 12, 2025 (ML25071A399 (package)). The TR describes the methodology used for the preliminary analysis of reactor core physics characteristics of the Xe-100 reactor. The CDMA methodology includes two analysis tools, very superior old programs (VSOP) and STAR-CCM+, that simulate the behavior of the reactor core at the beginning of life, startup, power ascension, and at equilibrium conditions. The methods described in this TR provide input parameters for the preliminary analysis of design basis accidents (DBAs) described in TR 007834, "Xe-100 Licensing Topical Report Transient and Safety Analysis Methodology, Revision 2" (ML25077A288). The CDMA methodology follows the risk-informed, performance-based design and licensing basis methodology described in Regulatory Guide (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," (ML20091L698).

VSOP is a computer code used for the numerical simulation of High Temperature Gas-Cooled Reactor (HTGR) neutronics that incorporates aspects of cross-section processing, reactor core and pebble geometries, neutron spectrum evaluation, neutron diffusion calculation, fuel burn-up, fuel shuffling, reactor control, and thermal hydraulics. STAR-CCM+ is a software package with a wide range of potential use cases in computational fluid dynamics (CFD) simulations. In the preliminary analysis of the Xe-100 reactor, STAR-CCM+ provides CFD simulations to analyze the pebble flow characteristics through the Xe-100 reactor core for use in the VSOP calculations. The CDMA provides an overview of the Xe-100 reactor core design along with preliminary calculational details of VSOP and STAR-CCM+ models, exemplary reactor core physics results, plans for verification and validation (V&V) of the computational tools, and a description of the quality assurance (QA) plan.

REGULATORY EVALUATION

Regulatory Basis:

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34(a)(1)(ii)(D) requires, in part, that an applicant for a construction permit (CP) perform an evaluation and analysis of a postulated fission product release to evaluate the offsite radiological consequences. This evaluation must determine that:

- An individual located at any point on the exclusion area boundary for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).

- An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE.

Under 10 CFR 50.34(a)(4) an applicant for a CP must perform a preliminary analysis and evaluation of the design and performance of structures, systems, and components (SSCs) with the objective of assessing the risk to public health and safety resulting from the operation of the facility and including determination of margin of safety during normal operations and transient conditions. These analyses are associated with the principal design criteria (PDC) and associated SSC design bases, which are required by 10 CFR 50.34(a)(3).

X-energy submitted TR, "Xe-100 Principal Design Criteria," Revision 3 (ML24047A308) which was reviewed and approved by the NRC staff, with limitations and conditions, as documented in its safety evaluation (SE) (ML24284A012). Based on the Xe-100 design overview, provided in various subsections of the CDMA, section 2, "Xe-100 Design Overview," the NRC staff identified the following PDC, including required functional design criteria (RFDC),[1] as applicable to the Xe-100 reactor core design methods and analysis:

- Xe-100 PDC 10, "Reactor design," requires that the reactor system and associated heat removal, control, and protection system be designed with appropriate margin such that specified acceptable system radionuclide release design limits are not exceeded. Demonstrating adequate reactor design generally includes, in part, the use of safety

[1] RFDC, as used in the Xe-100 PDC TR, is defined in NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Revision 1 (ML19241A472), which is endorsed by Regulatory Guide 1.233, "Guidance for a Technology Inclusive, Risk-Informed, and Performance-Based, Methodology and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors," (ML20091L698).

analysis methodologies (potentially including source term and consequence analysis methodologies).

- Xe-100 PDC RFDC 11, “Reactor inherent protection,” requires that the reactor core and associated systems be designed with sufficient negative reactivity feedback characteristics such that, in the power operating range, the net effect compensates for a rapid increase in reactivity, adequately controls heat generation, and ensures fuel performance and radionuclide release limits are not exceeded during design basis events (DBEs) or DBAs.

- Xe-100 PDC RFDC 16, “Functional containment design,” requires that the design of the reactor fuel particles and pebbles provide barriers as part of the reactor functional containment to control the release of radioactivity to the environment to ensure that the functional containment design limit is not exceeded during DBEs and DBAs.

- Xe-100 PDC 20, “Protection system functions,” requires, in part, that the protection system be designed to sense conditions and initiate the operation of necessary systems and components to perform required safety functions.

- Xe-100 PDC RFDC 26, “Reactivity control systems,” requires that the reactor core be designed to include movable poisons that can insert and maintain safe shutdown during DBEs and DBAs.

- Xe-100 PDC 28, “Reactivity limits,” requires that the reactor core, including the reactivity control systems, be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that DBEs and DBAs can neither: (1) result in damage to the reactor Helium pressure boundary greater than limited local yielding, nor (2) sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the core.

The NRC staff’s SE pertaining to TR Xe-100 Principal Design Criteria, Revision 3, includes a condition that applications referencing the TR must confirm that the PDCs remain appropriate for the design (ML24284A012). Therefore, the NRC staff determined that the list of PDCs identified above also needs to be confirmed to ensure the regulatory basis for the CDMA methods remains appropriate. Accordingly, the NRC staff imposed Condition 1 requiring that an applicant referencing the CDMA, needs to confirm or update the regulatory basis relevant to the use of CDMA methods (i.e., relevant PDCs).

Under 10 CFR 50.34(a)(8) an applicant for a CP must describe the research program to resolve any safety questions associated with safety features or components. Such research and development may include obtaining sufficient data pertaining to the safety features of the design to assess the analytical tools used for safety analysis in accordance with 10 CFR 50.43(e)(1)(iii).

Under 10 CFR 50.43(e)(1)(iii) (for applications for an operating license, design certification, combined license, standard design approval, or manufacturing license which differ significantly from light water reactor designs that were licensed before 1997) sufficient data must exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. As described in item 4, “Licensing Basis Events,” of section C, “Staff Regulatory Guidance,” to RG 1.253, Revision 0, “Guidance for a

Technology-Inclusive Content-of-Application Methodology to Inform Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors," (ML23269A222):

> It is also important to note that for CP applicants, the requirements of 10 CFR 50.43(e)(1)(iii) to ensure that sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses do not apply. Accordingly, CP applicants are not required to provide evaluations of the safety margins using approved evaluation models.

RG 1.231, Revision 0, "Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Safety-related Applications for Nuclear Power Plants," (ML16159A130) provides methods that the NRC staff considers acceptable in meeting regulatory requirements for acceptance and dedication of commercial-grade design and analysis computer programs used in safety-related applications for nuclear power plants. This RG endorses Revision 1 of Electric Power Research Institute Technical Report, "Plant Engineering: Guideline for the Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Nuclear Safety-Related Applications, Revision 1" (ML14085A084), with respect to the acceptance of commercial-grade design and analysis computer programs associated with basic components for nuclear power plants.

TECHNICAL EVALUATION

CDMA section 1.2, "Scope and Document Layout," provides the breakdown of sections contained within the TR. Accordingly, the NRC staff's review of the TR is addressed in the following sections of this SE:

- Section 1.2 discusses the evaluation of the Xe-100 reactor core design and operating approach that define the modeling requirements of the reactor core physics methods.

- Section 1.3 discusses the evaluation of results from the exemplary calculations performed using VSOP in support of the safety analysis and to demonstrate the capability of the code to model important physical characteristics of the Xe-100 reactor.

- Section 1.4 discusses the evaluation of validation and verification plans including data qualification for VSOP and STAR-CCM+ codes for its use in the reactor core physics analysis for the Xe-100.

- Limitations and Conditions discusses an additional consideration associated with the limitations pertaining to the CDMA for the evaluation of the reactor core design methods and requirements.

The NRC staff recognizes that fully assessed and approved codes and methods are not required to perform preliminary analyses for a CP (see Regulatory Basis), however 10 CFR 50.34(a)(8) requires a CP applicant to provide sufficient information on research and development programs needed to resolve any safety questions associated with safety features or components. The safety questions must be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility.

1.0 <u>Introduction and Outline</u>

The CDMA is divided into nine sections that the NRC staff evaluated in this SE as follows:

- Section 1, “Introduction,” of the CDMA provides an introduction to the TR and clarifies the scope for which NRC staff approval is being requested. The NRC staff have no regulatory determinations associated with the review of Section 1.

- Section 2 of the CDMA provides an overview of the Xe-100 design including the key reactor characteristics, fuel properties and pertinent operating parameters. The NRC staff considered the information provided in this section during its review, but have no regulatory determinations associated with the Xe-100 design overview and characteristics.

- Section 3 of the CDMA describes aspects of the Xe-100 reactor core design and operating approach that define the modeling requirements of the core physics methods. The NRC staff evaluates the reactor core physics in SE section 1.2.

- Section 4, “Results from Core Physics Methods,” of the CDMA provides results from the reactor core physics methods and codes described in section 3 of the CDMA. The NRC staff evaluates the results in SE section 1.3.

- Section 5, “Code Validation,” of the CDMA provides an overview of the validation plans for VSOP and STAR-CCM+, including plans to include comparisons to experimental datasets. The NRC staff evaluates the code validation methods in SE section 1.4.

- Section 6, “Verification Plan and Verification Method,” of the CDMA provides a high-level overview of planned verification activities for VSOP and STAR-CCM+. The NRC staff evaluates code validation methods in SE section 1.4.

- Section 7, “Quality Assurance,” of the CDMA states that activities described in this section are performed in accordance with the X-energy QA Program Description which has been reviewed and approved by the NRC staff (ML20343A088). The NRC staff have no regulatory determinations associated with the application of the QA program to the CDMA as part of this TR review.

- Section 8, “Conclusions and Limitations,” of the CDMA states that, “Until validation and verification of the VSOP and STAR CCM+ codes as described in [CDMA] are complete and have been approved by the NRC, the codes and methodology described [in CDMA] cannot be used to support a final safety analysis report.” The NRC staff considers this limitation throughout the technical evaluation in SE Technical Evaluation section.

- Section 9, “Cross References and References,” of the CDMA contains the reference list. The NRC staff have no regulatory determinations associated with the review of section 9.

This SE provides the NRC staff’s evaluation regarding the acceptability of the methods and codes identified in the TR as an appropriate means to perform steady-state and normal operations physics analysis as well as to provide inputs to support safety analyses of the Xe-100 reactor for future licensing actions. The NRC staff notes that, while the technical description of the calculational models in the TR describes approaches to performing the described calculations based on generally acceptable engineering techniques, the V&V of these models has not been completed. Therefore, the NRC staff’s review and conclusions regarding the CDMA modeling approach are limited to support for performing preliminary analysis. The acceptability of the codes and calculational models described herein for applicability to the Xe-100 design will be evaluated upon the completion of described V&V activities as described in the CDMA, under a future licensing application that references this topical report. Accordingly, the NRC staff imposed Limitation 1 to limit the applicability of the CDMA to the preliminary analysis of the Xe-100.

1.1. Xe-100 Design Overview

As described in the topical report, the Xe-100 is a 200 MWt pebble bed HTGR fueled by spherical graphitic matrix fuel elements (pebbles) that contain uranium oxycarbide (UCO) tristructural isotropic coated nuclear fuel particles. The cylindrical pebble bed is surrounded with graphite reflector blocks on all sides, located within the reactor pressure vessel. Two banks of neutron absorbing control rods (a control bank and a shutdown bank) are insertable in the side reflectors, near the pebble bed reactor core. The reactor core is normally cooled by Helium gas pumped through the core and steam generator by Helium circulator pumps. During operation, fuel pebbles are continually withdrawn from the bottom of the core and either removed from circulation for disposal or reinserted at the top of the core, depending on the fuel pebbles' determined burn-up and remaining life. New fuel pebbles are added to replace spent fuel pebbles, as required to maintain the appropriate operational characteristics of the reactor core. In-depth details of the reactor and core design, pertinent reactor characteristics and details of fuel and control elements are provided in section 2 of the CDMA.

While important to the context of the methodologies presented in the TR and reviewed in this SE, the Xe-100 design features are not within the review scope of this SE.

1.2. Overview of Core Physics

The aspects of Xe-100 reactor core design and operating approach that define the modeling of the core physics methods are outlined in section 3 of the CDMA. The section also describes the modeling techniques and capabilities of the VSOP code, including a summary of the code's general analysis. Additional modeling details regarding the treatment of specific technical phenomena such as nuclear cross-section processing, resonance integrals, neutron escape probabilities, material representations within models, and neutron diffusion calculations are provided. Further, aspects of operational feedback such as thermal hydraulic feedback, fuel burn-up, and fuel management during reactor operation are discussed.

The NRC staff's evaluation of the core physics methods focused on assessing the acceptability of the core physics model. This evaluation includes defining the purpose of the analysis, identification of pertinent systems for the model, identification of the components and component geometries that are essential to the evaluation, and identification of key phenomena and processes.

The NRC staff reviewed the modeling methodology to determine whether the calculational approach is adequate to simulate the described Xe-100 reactor core conditions and generate appropriate results for use in safety analysis methods input to support the licensing of the Xe-100 design. The following SE sections detail the NRC staff's review of the CDMA methodology described.

Nuclear Data

CDMA section 3.2.2, "Cross-Section Libraries," states that VSOP data libraries are derived from Evaluated Nuclear Data Files (ENDF)/B-V and JEF-1. The NRC staff notes that individual nuclear data libraries are calibrated based on results from critical experiments; therefore, the use of nuclear cross-section information from different data libraries can lead to errors. However, the NRC staff recognizes that combining data libraries is a common practice when a single library lacks all necessary nuclear data (e.g., ORIGEN libraries in SCALE 6.3.2 includes nuclear data from ENDF/B-VII.1 and JEFF-3.0/A (Reference 1)). Accordingly, the NRC staff determined that the VSOP cross-section library based on ENDF/B-V and JEF-1, can be used in the Xe-100 core diffusion calculations, pursuant to methodology assessment, because: (1) ENDF and JEF are internationally recognized and commonly used nuclear cross-section libraries, (2) it is standard practice in nuclear applications to combine libraries for depletion analysis, and (3) any bias or uncertainty introduced by the nuclear data will be addressed by the final design methodology assessment.

The NRC staff considered the calculational techniques described in the CDMA that address physically sensitive phenomena such as resonance absorption and geometric escape. The NRC staff noted that cross-section processing techniques include the calculation of resonance integrals and neutron escape probabilities. The calculations presented rely on solutions of higher-order equations (i.e. solution of the neutron transport equations) to adequately represent the physically sensitive phenomena. The calculations are performed utilizing the adjustment of cross-section data such that the broader diffusion-based core calculation can rely on the adjusted data to capture the underlying sensitive physical phenomena. The NRC staff noted that the calculational techniques presented in the CDMA are acceptable for preliminary analysis because they are standard approaches that are used within the nuclear industry to enhance the accuracy of diffusion-based calculational models.

Core Design and Calculations

The NRC staff considered the core design and modeling calculations performed using the VSOP code, as described in section 3.2.3, "Core Design," through section 3.2.6, "Modeling of Neutron Absorbers (Control Rods)," of the CDMA. Specifically, the NRC staff noted that the methodology employs a spatial mesh that is strategically divided into geometric regions to model fuel pebble movement as a function of core position, which the TR states is expected to directly influence neutronic parameters of interest. The NRC staff determined the incorporation of pebble movement in the neutronics model at this fundamental level to be an acceptable approach, because it enables the model to capture the effects of fuel pebble residence time and resultant burn-up and allows the prediction of reactor core nuclear performance over varying periods of core life from the initial startup to equilibrium conditions. Section, "Safety Analysis Inputs," below, discusses the kinetics parameters obtained from VSOP for use to transient and safety analysis.

The NRC staff noted that the flow channel and region configuration that capture pebble movement through the reactor core consists of a spatial mesh (CDMA figure 9) that differs from the spatial mesh used in the reactor core neutronics calculations (CDMA figure 10). The CDMA describes that the flow channel information is used to generate fuel "batches" to represent the pebbles from different passes through the reactor core in different locations. The neutronic mesh employs a finer spatial discretization that is superimposed on this mesh and thus accounts for batch-specific histories of the fuel constituents while more discretely modeling specific regions of the reactor core as needed, such as reflectors and absorber rods. The finer neutronics mesh is employed for performance of the diffusion calculations, and the solution informs the batch-wise data for burn-up, fuel shuffling, and decay heat production parameters that are used in steady-state and quasi-steady-state transients. The NRC staff determined the spatial modeling approach to be acceptable because it supports the model's ability to perform core neutronic calculations with fine spatial discretization which is superimposed on the mesh while also informing the batch-wise data to capture the effects of fuel movement through the reactor core during operation.

The CDMA states that the VSOP calculation performed collapses fine-group neutronic data to create broad-group macroscopic cross-sections that reflect cell zones (neutronic mesh), predicted cell flux spectra, and other aspects such as self-shielding. The NRC staff noted that use of such an energy collapsing approach is an expected component of the calculational scheme used to support subsequent diffusion calculations. The NRC staff determined that the collapsing approach described in the CDMA is acceptable because it is consistent with standard cross-section homogenization approaches (Reference 2). The broad-group cross-sections are used by VSOP code when solving the diffusion equation using a finite-difference technique to obtain core flux and power distributions. The NRC staff considered the formulations described in the CDMA and determined that they represent the diffusion-based solution technique and that their alignment with accepted engineering first principles provide confidence that the underlying physics of the reactor core are adequately modeled.

The NRC staff considered the described approach to modeling neutron absorbers (control rods) in VSOP code. The CDMA states that regions with strong absorbers typically present challenges to methods that rely on diffusion-theory solutions, as used by VSOP. A solution technique based on incorporating an equivalent Boron-10 concentration to the model is described in the CDMA as a means for modeling such absorbers. The section describes a method typically used to generate appropriate Boron-10 concentrations (referred to as the Method of Equivalent Cross Sections (MECS)) but states that it is not used in evaluating the Xe-100 design due to associated challenges with core anisotropy in the radial plane. Alternatively, CDMA section 3.2.6 mentions that a 3-D Monte Carlo N-particle (MCNP) model is used for calculating the worth of control rods at equilibrium reactor core conditions and further generating equivalent Boron-10 concentrations for use in the VSOP model. This is done to avoid the challenges associated with the MECS approach. The section further states that the pertinent model parameters such as geometrical and material data, pebble flow channel information and the temperature profiles for the MCNP model were adopted form the VSOP model. The total rod worths in the VSOP model are then matched in MCNP by adjusting the equivalent Boron-10 concentrations. The section states that the method of equivalent boron replaces the transport and 3D diffusion calculations and removes the assumptions that are required for the MECS methodology.

The NRC staff determined the methodology described for calculating the worth of control rods and further generating equivalent Boron-10 concentrations to be acceptable because the interactive exchange of data between MCNP and VSOP preserves the reaction rates which preserves overall reactivity balance and associated control rod worths between the higher-order MCNP calculation and the lower order VSOP calculation.

Burn-up Calculations

The CDMA states that the VSOP code calculates burn-up in each mesh within the reactor core based on a combination of information from multiple aspects of the reactor core simulation. Specifically, the fuel shuffling and movement, the spatial and energy dependent neutron flux, and resulting local depletion is evaluated over pre-selected time periods consisting of multiple time steps during which the diffusion and spectrum calculations are performed. The results of the calculations are used to predict and track both heavy metal isotope burn-up and fission product isotopic buildup and decay. The NRC staff noted that the burn-up and fission product chain calculations included in the VSOP model are constrained to include only the isotopes found to result in the vast majority of neutronic feedback effects, specifically 28 heavy metal isotopes and 44 fission product chains. The NRC staff concluded that the isotopic tracking approach described in the CDMA for use in VSOP code is acceptable because it focuses on isotopes of highest importance to reactivity balance and the methodology assessment (see SE section 1.4, “Code Validation and Verification”) will address any uncertainties or biases introduced by the simplification.

Thermal Hydraulics

A thermal hydraulics model included in the VSOP code provides temperature feedback to the neutronic models so that the simulation can account for the temperatures of the material constituents. The NRC staff reviewed the thermal hydraulic model in the CDMA and notes that it describes the first principal engineering concepts of thermal hydraulics with assumptions where appropriate. Specifically, it solves coupled conservation of mass, momentum, and energy equations for the moderator and fuel regions of the reactor core, with assumptions the NRC staff determined are reasonable. The NRC staff concluded that this approach is acceptable for supporting thermal hydraulic feedback to the reactor core neutronics calculations because it provides a physics-based mechanism that is relied upon to ensure that the neutronics calculations account for material temperatures. The NRC staff noted that the thermal hydraulic model described in CDMA section 3.2.8, “Thermal Hydraulics,” states that the Helium bypass flow fraction is included in the VSOP modeling but did not provide any specific information on the calculation of the bypass flow value which the NRC staff determined is important to modeling of the thermal hydraulics phenomena and its feedback to neutronic model. Consistent with the requirements of 10 CFR 50.34(a)(4), the NRC staff does not expect the details of the implementation of bypass flow modeling in the reactor core design methodology along with the associated basis for the CP stage but expects to review such details in the final design methodology.

Decay Heat

The CDMA states that VSOP code tracks the fuel life history as discussed in the previous sections and calculates decay heat based on the DIN 25485 standard that was derived for pebble bed HTGRs with similar fuel cycles (Reference 3). The CDMA further states that this standard has been used internationally for pebble bed reactors and in many core physics codes. Further, the DIN 25485 standards consider contributions of fission products from the four isotopes: ^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu, actinides and the isotopes resulting from neutron capture of the fission products. The CDMA further states that the standard considers the composition and power change in fuel operation by accounting for the flow of pebbles through the different power regions of the reactor and production and utilization of the fissile isotopes.

While the use of DIN 25485 standard for calculating decay heat is reasonable based on its application in other reactors and core physics codes, as stated by the licensee, the NRC staff notes that additional details are needed for how the calculation is performed for a validation basis for the standard DIN 25485 for use in Xe-100 applications. Consistent with the requirements of 10 CFR 50.34(a)(4), the NRC staff does not expect the details on calculation of decay heat using DIN 25485 and its validation for use in Xe-100 applications for the preliminary design but expects the final design methodology to include such details.

Fuel Management

CDMA section 3.2.10, "Fuel Management," provides details pertaining to the VSOP simulation of the fuel movement phenomena in the reactor core to reflect the online fuel management approach inherent to the Xe-100 design. Specifically, fuel pebbles move slowly downward through the reactor core because fuel pebbles are removed from the bottom of the reactor core, analyzed, and if appropriate, reinserted at the top of the core. As previously discussed in the SE, VSOP uses a spatial mesh to track pebble movement through the reactor core. The spatial mesh is discretized into radial flow channels with multiple layers in the axial direction. When pebbles exit the reactor core, they are separated into simulated storage boxes based on the number of passes through the core. On a batch-wise basis, the simulated storage boxes are volume averaged for isotopic content, and the pebbles are then re-introduced at the top of the core assuming each pebble consists of the averaged isotopics. The TR describes that this averaging approach is necessary because the radial location that any specific re-introduced pebble will flow through the reactor core during its next pass is stochastic in nature, as all pebbles are inserted at the center of the reactor core and randomly distribute radially. The NRC staff notes that the approach of volume averaging batch-wise pebble re-introduction phenomena is a reasonable approach to tracking cumulative batch-wise pebble isotopics for the purpose of ensuring that the VSOP code adequately predicts the overall reactor core conditions, despite the stochastic nature of specific pebble re-introduction because the volume averaging of the isotopic content accounts for fuel burnup, fission product buildup, and neutron absorption which provides confidence that overall reactor core isotopics are conserved and modeled appropriately without requiring discrete tracking of individual pebbles.

CDMA section 3.2.10 states that pebble flow used in VSOP is an input from STAR-CCM+ and that pebble flow is modeled as five radial flow channels in VSOP, with the pebble velocity within each channel being viewed as radially independent. The NRC staff agree with the approach that only the vertical component of pebble movement needs to be considered within VSOP because the shape of the radial flow channels is specifically defined such that radial pebble movement between channels can be neglected.

The NRC staff notes that while the volume averaging method for batch-wise pebble re-introduction phenomena for tracking cumulative batch-wise pebble isotopics is described in the licensing topical report, no details are provided in the CDMA on estimation of burn-up experience by individual pebbles to establish a burn-up limit for reinsertion below the maximum burn-up limit. Consistent with the requirements of 10 CFR 50.34(a)(4), the NRC staff does not expect details of the reinsertion burn-up limit for individual pebbles for the preliminary design but expects the final design methodology to include such details.

VSOP Calculation Capabilities

CDMA section 3.2.11, "VSOP Calculation Capabilities," states that VSOP calculates flux and power distribution for the range of operational conditions between startup and equilibrium reactor core conditions. Reactivity coefficients, control rod worths, and Xenon feedback constants are calculated by VSOP. Additionally, as presented in CDMA section 3.2.11, the kinetics parameters used in the downstream safety analyses are also calculated by VSOP code. The NRC staff reviewed the formulations used to determine kinetics parameters such as delayed neutron fraction, delayed neutron decay constants, and neutron mean generation time and determined them to be consistent with standard definitions for kinetics parameters (Reference 4). Hence, the NRC staff concluded that the kinetics parameters calculated by VSOP code are acceptable for use in the downstream safety analyses.

STAR-CCM+

CDMA section 3.3, “Overview of STAR-CCM+,” provides details regarding the STAR-CCM+ code and analysis that is used to simulate pebble movement through the Xe-100 reactor core and provide VSOP with the appropriate spatial mesh and pebble flow characteristics to allow for tracking and simulation of fuel utilization and isotopics throughout reactor core operations, as previously discussed.

STAR-CCM+ is a widely used software package which provides an environment for detailed CFD simulations to be configured, executed, and further interrogated. The STAR-CCM+ model for the Xe-100 reactor core uses Discrete Element Method (DEM) for pebble flow analyses, specifically to model the interaction between pebbles in contact in the reactor core. The STAR-CCM+ model uses Hertz-Mindlin Contact Model to model the sliding friction and momentum transfer between the pebbles in the reactor core and a Force Proportional Rolling Contact Model to model the rolling friction of discrete elements in contact. The Xe-100 reactor core design primarily uses STAR-CCM+ to determine the pebble flow characteristics through the reactor core for specific radial flow channels as an input into the VSOP code system. Based on the information provided in the CDMA and pursuant to the code validation plan discussed in section 5.2, “STAR-CCM+,” of the CDMA, the NRC staff concluded that the use of STAR-CCM+ to generate pebble flow characteristics through the reactor core as an input for the VSOP code for preliminary analysis is acceptable because DEM, as implemented in STAR-CCM+, is capable of modeling pebble flow through the reactor core and these modeling capabilities will be assessed as part of the validation plan.

The NRC staff notes that stability of the Xe-100 reactor from operational phenomena, such as Xenon oscillations, is not evaluated here and is expected to be analyzed as part of the final reactor core design and analysis.

Safety Analysis Inputs

CDMA section 3.2, “Overview of VSOP Computer Code System,” identifies the input parameters used in the safety analysis that are obtained from the reactor core design methodology as:

- Reactivity coefficients (Fuel temperature coefficient, moderator temperature coefficient, reflector temperature coefficient)
- Xenon feedback coefficients
- Control and shutdown element worth (integral and differential)
- Power distribution (peaking factors, axial and radial power profiles)
- Kinetics parameters (delayed neutron fractions and delayed neutron decay constants, neutron mean generation time)

CDMA section 4 contains results from example calculations that provide representative values for these parameters and is discussed in section 1.3, “Results from Core Physics Methods,” of this SE. The NRC staff expects that a final reactor core design and analysis methodology will include a set of biases and uncertainties to be applied to these input parameters to provide assurance of adequate design margin.

Core Physics Overview Conclusions

Based on the information provided in CDMA section 3 as described in SE section 1.2, “Overview of Core Physics,” above, the NRC staff determined that the CDMA provides sufficient level of details to establish requirements for the reactor core physics model capability including clearly identifying purpose of the analysis, identification of pertinent systems, components and geometries that are essential to the evaluation and identification of key processes. Hence, the NRC staff concludes that the codes and methods described as described in section 3 of the CDMA, along with the conditions and limitations specified herein, are acceptable for the preliminary analysis of the design and performance of Xe-100 in accordance with 10 CFR 50.34(a)(4).

1.3. Results from Core Physics Methods

The NRC staff reviewed the results provided in CDMA section 4 and notes that the information provided is exemplary in nature and is intended to demonstrate the general capabilities and outputs of the reactor core analysis methods described in the report. The results from reactor core physics calculations are presented for key phenomena including reactivity coefficients, Xenon feedback, shutdown reactivity worth, axial and radial reactor core power distributions as well as a representative six-group delayed neutron data. Results are also provided for the typical STAR-CCM+ relative velocity for each of the radial channels that is input into VSOP code.

While the results presented from the calculational models in the CDMA demonstrate the capability of the codes to model important physical characteristics of the Xe-100 reactor depth and support input generation for the downstream safety analysis, the NRC staff notes that given the exemplary nature of the results presented, they are not intended to address any specific safety analysis needs for licensing a Xe-100 reactor. Therefore, while the NRC staff recognized that the results demonstrate and support the capability of reactor core analysis methods, the NRC staff did not make a determination regarding the acceptability of the specific results presented in CDMA section 4.

1.4. Code Validation and Verification

An overview of validation plans for VSOP and STAR-CCM+ is described in section 5 of the CDMA. As described in CDMA section 5.3, “Validation Reports,” the detailed validation exercises will be documented in standalone reports in the future. Similarly, the planned verification activities to support the assessment of the evaluation model adequacy are described in CDMA section 6, “Verification Plan and Validation Method.”

The NRC staff’s evaluation for the overview of the code V&V focused on the development of an appropriate assessment base for the evaluation model. This included specifying the objective of the validation, determining the adequacy of the validation models, identifying existing test databases and developing uncertainties in the models and methods where appropriate.

VSOP Validation

Three experimental data sets are identified and planned to be relied upon to support validation of the VSOP code. Each dataset is discussed, including pertinent design parameters and experimental data available for comparison with respect to the Xe-100 design. Some parameters of the experimental datasets available for comparison to support a code validation include criticality, temperature coefficients, control rod worths, steady-state operations, power ascension operations, kinetics parameters, reflector worth, pebble packing fraction, and mixed-cores of graphite moderator, poison, and fuel pebbles. Of these, parameters of interest were specified for comparison to each of the experiments identified. Specifically, the parameters of interest focused on significant code results, including K-effective, control rod worths, reflector worths, and kinetics.

Additionally, a code-to-code benchmarking is identified for inclusion in the validation plan. Specifically, CDMA table 10, “VSOP Code-to-Code Benchmarks,” identifies benchmarks that will be performed to compare VSOP against higher-order codes.

The NRC staff considered the applicability of each validation case and whether the test cases and parameters of interest could reasonably be expected to provide an adequate basis for the validation of VSOP. The NRC staff notes that the experimental tests identified for comparison are similar in design to the described Xe-100 and, therefore, are expected to provide useful comparisons to support the validation of the VSOP methods described in the TR. Additionally, the planned validation cases and associated parameters of interest for comparison appear to cover the range of parameters that VSOP contributes to inform the overall transient safety analysis evaluation model and support the safety analyses. The NRC staff expects that the final reactor core design and analysis methodology will justify test applicability to the Xe-100 reactor design (e.g., similar materials, neutron spectra, importance of neutron leakage, etc.).

STAR-CCM+ Validation

The CDMA states that the validation basis for STAR-CCM+ will rely on the comparison to the ANABEK experiment, which focused on characterizing pebble flow behaviors at scale. Scalability between the ANABEK experiment and the Xe-100 design is considered based on ratios associated with the reactor core dimensions and shape, and the spherical pebble dimensions, as shown through the experiment by D. Bedenig, as described in CDMA section 5.2.2, “STAR-CCM+ DEM Validation Data.” CDMA section 5.2.2 states that the ANABEK experiment has geometric parameters that are similar to the Xe-100 design with respect to pebble flow because the ANABEK experiment and Xe-100 design have similar values for: (1) the ratio of the core height to the diameter, (2) the ratio of the core diameter to the pebble diameter, (3) the ratio of the defuel chute diameter to the pebble diameter, and (4) the outlet funnel’s conical angle to the horizontal plane.

The comparison will focus on the non-dimensional retention time ratio that describes the pebbles’ relative in-core time through the centerline of the reactor core versus at the wall, and the residence time ratio, which describes a measure of pebble residence time normalized to the recirculation frequency. The approach used in this validation will be similar to the approach described in Xhonneux et. al. (CDMA, Reference 14), with a preliminary validation goal of +/- 10 percent for acceptability. The NRC staff concluded that this approach is acceptable since it is based on comparison to experimental evidence and is an approach that has been demonstrated successfully in the work referenced in the CDMA. Further, the acceptance criterion provided appears to be commensurate with the sensitivity of the overall methodology to the results of STAR-CCM+ pebble flow modeling.

Verification Plan

The planned verification activities along with the validation method are described in CDMA section 6 and support the assessment of the evaluation model adequacy. Specifically, each validation exercise, and associated reports, will be examined to confirm that the aspects of the activities listed in CDMA section 6.1, "Verification Plan," are met. The NRC staff notes that the checklist provides assurance that inputs, models, correlations, and other aspects of the analyses of the experiments are performed in a logical manner to provide reasonable justification for the modeling and the results. Two of the listed verification checks are further expanded upon in the CDMA, and are of notable importance to the NRC staff, as described below.

Verification that the benchmark data is suitably qualified for use in the validation exercise is discussed in CDMA section 6.2.1, "Qualification of Data," where the supporting checks and considerations, to be documented, are listed. The NRC staff notes that the listed checks generally ensure that the experimental data and any differences between the experiments and models are well understood and are appropriately justified.

The code accuracy calculations of the bias and variation in the bias is described in CDMA section 6.2.3, "Code Accuracy Calculations." The bias consists of measured minus predicted residuals that are averaged over the analysis time steps, and the variation in bias consists of the standard deviation of the residuals across the set of time steps of interest.

The NRC staff notes that the TR lists a limitation that the VSOP and STAR-CCM+ codes, as described in the report, cannot be used to support a final safety analysis report until the V&V of the codes are complete and have been approved by the NRC. The NRC staff determined that this applicant-imposed limitation is acceptable since it is consistent with 10 CFR 50.34(a)(4) and its applicability to preliminary design information expected during the CP application.

Validation and Verification Conclusions

The NRC staff concluded that the code validation method and verification plan appropriately described the objective of the V&V including determining adequacy of the validation methods, identifying existing test databases and performing evaluations to establish model adequacy.

While the NRC staff determined that the V&V method is acceptable based on the description provided, the NRC staff notes that the final approval of such will be based on the NRC staff's review of the detailed V&V of the methods and results for VSOP and STAR-CCM+ during a future licensing application for the final design. Additionally, the NRC staff expects that a final core design and analysis methodology will include a process to update biases and uncertainties based on plant surveillance information.

LIMITATIONS AND CONDITIONS

The NRC staff's conclusions pertaining to the CDMA are subject to the following limitations and conditions:

Limitation 1	Application of this TR is limited to the preliminary safety analysis of the Xe-100 design. SE section 1.0, "Introduction and Outline," describes the basis for this limitation.
Condition 1	A CP application referencing this TR must confirm or otherwise justify the regulatory basis for the assessments that use CDMA methods. SE section "Regulatory Evaluation" describes the basis for this condition.

CONCLUSIONS

The NRC staff approves the use of TR, “Xe-100 Licensing Topical Report Reactor Core Design Methods and Analysis,” Revision 2, for the preliminary analysis of the Xe-100 reactor core subject to the limitations and conditions identified in the SE Limitations and Conditions section above. This determination is based on the following:

- The use of the codes and methods described in the CDMA is preliminary and still under development. Hence the information provided is subject to change and not considered complete. Subject to Limitation 1 of the SE, the final acceptability of the codes and methods described in this TR will be based on the review of detailed calculational methods, any applicable testing and associated V&V along with the conclusions derived from these methods during future licensing action for the construction or operation of a facility, as appropriate.

- The use of codes and methods as described in the CDMA is acceptable for the preliminary design and analysis of the Xe-100 reactor core in accordance with 10 CFR 50.34(a)(4) and, pursuant to Condition 1, would support demonstration of compliance with Xe-100 PDC 10, Xe-100 PDC RFDC 11, Xe-100 PDC RFDC 16, Xe-100 PDC 20, Xe-100 PDC RFDC 26 and Xe-100 PDC 28 because: (1) CP applications are not required to provide evaluations of the safety margins using approved methods (see SE Regulatory Evaluation section), and (2) sufficient justification would be provided in a CP application to ensure that the codes and methods used for the analysis of the Xe-100 reactor are capable of providing reasonable assurance of acceptability.

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