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NUCLEAR TECHNOLOGIES AND MATERIALS ADVANCED REACTOR CONCEPTS-20

FAST MODULAR REACTOR PRE-APPLICATION REGULATORY ENGAGEMENT PLAN

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ACRONYMS

Acronym	Definition
ADAMS	Agency wide Documents Access and Management System
AFQ	Accelerated Fuel Qualification
ANL	Argonne National Laboratory
AOO	Anticipated Operational Occurrence
ARC-20	Advanced Reactor Concepts-20
ARDC	Advanced Reactor Design Criteria
ARDP	Advanced Reactor Demonstration Program
ATF	Accident Tolerant Fuel
ATR	Advanced Test Reactor
BDBE	Beyond Design Basis Event
CFR	Code of Federal Regulations
COL	Combined License
DBE	Design Basis Event
DiD	Defense-In-Depth
DLOFC	Depressurized Loss of Forced Cooling
DOE	Department of Energy
DPA	Displacement Per Atom
ECN	Engineering Change Notice
EM^2	Energy Multiplier Module
EPRI	Electric Power Research Institute
ESP	Early Site Permit
FMR	Fast Modular Reactor
FRA	Framatome
GA	General Atomics
GA-EMS	General Atomics Electromagnetic Systems
GDC	General Design Criteria
GFR	Gas-cooled Fast Reactor
I&C	Instrumentation and Control
INL	Idaho National Laboratory
LBE	Licensing Basis Event
LFR	Lead-cooled Fast Reactor
LMP	Licensing Modernization Project
LOCA	Loss of Coolant Accident
MHTGR-DC	Modular High Temperature Gas-cooled Reactor Design Criteria
MSR	Molten Salt Reactor
MWe	Megawatt electric
NEI	Nuclear Energy Institute
NG	Nuclear Generation

Acronym	Definition
non-LWR	non-Light Water Reactor
NRC	Nuclear Regulatory Commission
NSR	Non-Safety Related
NSRST	Non-Safety Related with Special Treatment
PCU	Power Conversion Unit
PDC	Principal Design Criteria
PIE	Post-Irradiation Examination
PRA	Probabilistic Risk Assessment
QA	Quality Assurance
QAP	Quality Assurance Program
QAPD	Quality Assurance Program Description
QMP	Quality Management Plan
RCCA	Rod Cluster Control Assembly
RCCS	Reactor Cavity Cooling System
REP	Regulatory Engagement Plan
Rev	Revision
RG	Regulatory Guide
RIA	Reactivity Initiated Accident
RVCS	Reactor Vessel Cooling System
SAFDL	Specified Acceptable Fuel Design Limit
SAM	System Analysis Module
SFR-DC	Sodium-cooled Fast Reactor Design Criteria
SMR	Small Modular Reactor
SNL	Sandia National Laboratory
SR	Safety Related
SSC	Structures, Systems, and Components
TI-RIPB	Technology-Inclusive, Risk-Informed, and Performance-Based
TREAT	Transient Reactor Test Facility
UWM	University of Wisconsin at Madison

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1 INTRODUCTION

General Atomics Electromagnetic Systems (GA-EMS) is pursuing design, licensing, and commercialization of a 50-megawatt electric (MWe) Fast Modular Reactor (FMR), with demonstration by 2030 and deployment by the mid-2030s. Over the next three years, GA-EMS will develop the conceptual design of the FMR plant with verification of key metrics in fuel, safety, and operational performance [1]; improve the technology readiness of the key systems and components; and develop a licensing approach for this advanced reactor technology. This is pre-application Regulatory Engagement Plan (REP) for the project.

This document describes the overall conceptual design of the GA-EMS FMR reactor system, the licensing strategy, and pre-application licensing submittals for review by the U.S. Nuclear Regulatory Commission (NRC) advanced reactor staff. It is mainly focused on early activities of the GA-EMS FMR design and analysis to support critical development decisions and the development of pre-application submittals for NRC review and feedback. Some important objectives of this pre-application regulatory engagement plan include facilitating the mutual understanding of the GA-EMS FMR design team and the NRC, initiating discussions of some key features and challenges of this novel technology, and fostering strategic thinking for the development and delivery of this advanced reactor design. It provides the NRC staff with information on a summary level, with a goal to identify and resolve issues as early as practical during the development of the GA-EMS FMR technology.

The enclosed information is preliminary and pre-decisional and is subject to change during detailed planning and project execution. The provision of the enclosed information is for planning and familiarization purposes in support of pre-application discussions with the NRC.

2 CONCEPTUAL DESIGN OF GA-EMS FAST MODULAR REACTOR

General Atomics (GA) has been at the forefront of innovation in nuclear energy since its founding in 1955. The GA TRIGA® (Training, Research, Isotopes, General Atomics) reactor is the most widely used non-power nuclear reactor in the world. GA has developed and installed 66 TRIGA reactors at universities, government and industrial laboratories, and medical centers in 24 countries. GA-EMS continues to develop and deploy advanced reactors of the state of the art in efficiency, safety, and economics. For example, GA-EMS has developed Energy Multiplier Module (EM^2) Small Modular Reactor (SMR) that addresses the most challenging problems of nuclear power generation such as economics, safety, waste, and nonproliferation. GA-EMS is also a leader in developing high-temperature nuclear materials, including the cutting-edge technology of Accident Tolerant Fuel (ATF). For example, the GA-EMS SiGA® cladding allows fuel rods to withstand temperatures of over 3000 °F (1700 °C), more than twice that sustained by the metal cladding of the current commercial reactor fuels.

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2.1 FMR Project Overview

The U.S. Department of Energy (DOE) has selected GA-EMS FMR for its Advanced Reactor Demonstration Program (ARDP), specifically Advanced Reactor Concepts-20 (ARC-20), to develop the conceptual design with verifications of key metrics in fuel, safety, and operational performance. GA-EMS proposed to develop a new 50 MWe Gas-cooled Fast Reactor (GFR) that provides safe, carbon free electricity, capable of incremental capacity additions. This modular reactor system is factory-built and assembled on-site to keep the cost of capital low. Its dry-cooled reactor and low-water-usage facilitates allow a FMR power plant to be sited in nearly any location.

GA-EMS brings a distinguished team to this FMR project, with a strategic partner of Framatome (FRA), as well as technical support from Idaho National Laboratory (INL), Argonne National Laboratory (ANL), Sandia National Laboratory (SNL), Electric Power Research Institute (EPRI), and the University of Wisconsin at Madison (UWM). This team brings together expertise needed for the conceptual design of all critical systems and subsystems of the FMR plant.

The FMR will be designed with the following features:

- Enhanced safety and ease of operation
- Fast load-follow response and overall high efficiency
- Passive safety systems with less reliance on additional pumps and alternating current power for accident mitigation
- Modularity of fabrication with higher quality standards
- Smaller radioactive inventory and reduction of the source term
- Underground location of the reactor unit providing more protection from natural (e.g., seismic) or man-made (e.g., aircraft impact) hazards

2.2 Plant Description

The FMR is a helium-cooled, fast reactor with a core outlet temperature of 800 °C. It is a grid-capable power source with a net electrical output of 50 MW. The prototype plant consists of a single reactor module sited below grade. Figure 1 shows an artistic rendering of the plant, of which the security border, access points, and switchyard are not shown (and will be included in the design). The limited number of structures give a sense of compactness and simplicity of the plant. Dry cooling capability is a distinctive feature that allows the plant to be built in remote regions. High temperature output can extend the application to mining or desalination. The entire system is designed for compactness and ease of construction.

One of the goals of the FMR is to have fast grid response capability, up to a 20% per minute power ramping for load following, using the direct helium Brayton cycle. This will be achieved by automatic control of the reactor power synchronized with turbomachinery that keeps the reactor at a constant temperature to mitigate thermal cycle fatigue typical with most load-following operations. With the direct thermal cycle, the overall thermal efficiency will be as high as 45% during normal operation.

The major systems and components are installed underground. The maintenance hall of the reactor building is set at grade and provides access to the reactor vessel and Power Conversion Unit (PCU). All water, helium, and fire suppression systems are protected in an underground fluid transport network structure connected to the reactor. Dry cooling structures are located adjacent to the reactor to minimize the transport distance of fluids. Support equipment for helium and water processing, cooling, fire suppression, and backup power systems are all contained within the auxiliary building. The administration building provides an operational work space hub for staffs.

Figure 1. GA-EMS 50 MWe FMR Plant

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2.3 Power Operation

The high temperature Brayton cycle provides power and maneuverability. The primary coolant flow path for power operation begins with 800 °C helium flowing from the core at 74 kg/s to the PCU through the inner concentric duct as illustrated in Figure 2. It expands over the turbine (at 543 °C, 3.4 MPa) to the recuperator and then flows to the precooler, compressor, and intercooler (at 48 °C, 4.8 MPa), which are the cold sinks. The helium is pressurized to 7.1 MPa at 110 °C through the intercooled compressor and returned to the cold side of the recuperator and then to the outer cross-duct annulus. The compressed helium at 509 °C exits the cross-duct into the reactor and flows around and down through the inner annular surface of the reactor vessel to the lower plenum below the core.

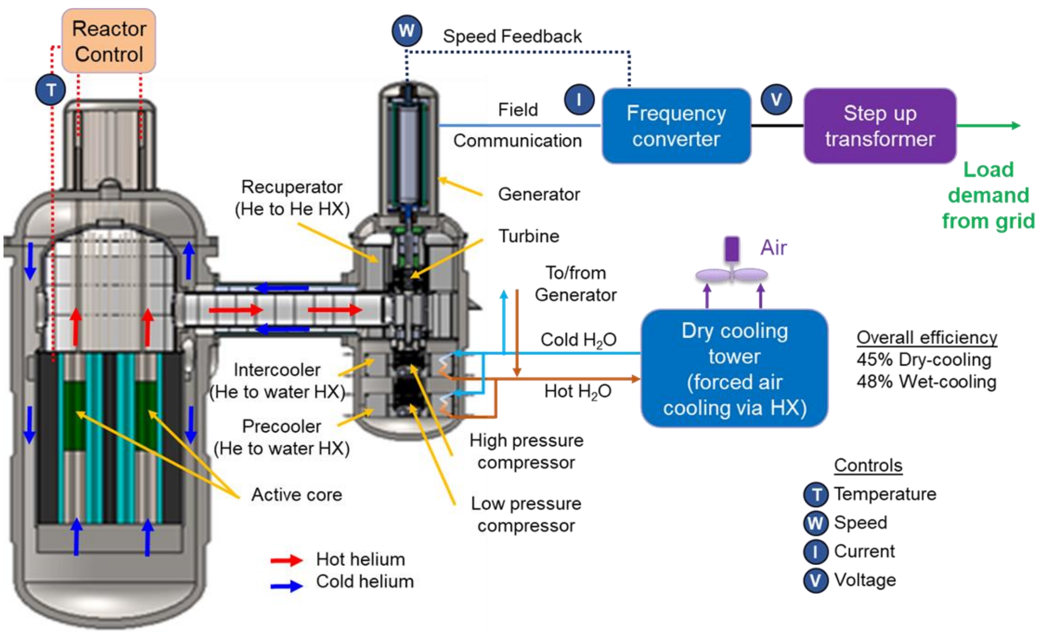

Figure 2. FMR Primary Coolant Flow and Power Generation

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2.4 Major Systems

The major systems and components are underground, as illustrated in Figure 3. The vessel system consists of the reactor vessel, power conversion system vessel, cross duct vessel, re-circulator vessel, and maintenance cooling system vessel. The reactor vessel consists of a permanent upper head, a removable upper center head, gaskets, and a flanged pressure vessel. It contains the fuel assemblies, reflector assemblies, a B_4C shield, core shroud, lower core support structure, reactivity control devices, and in-core instrumentation.

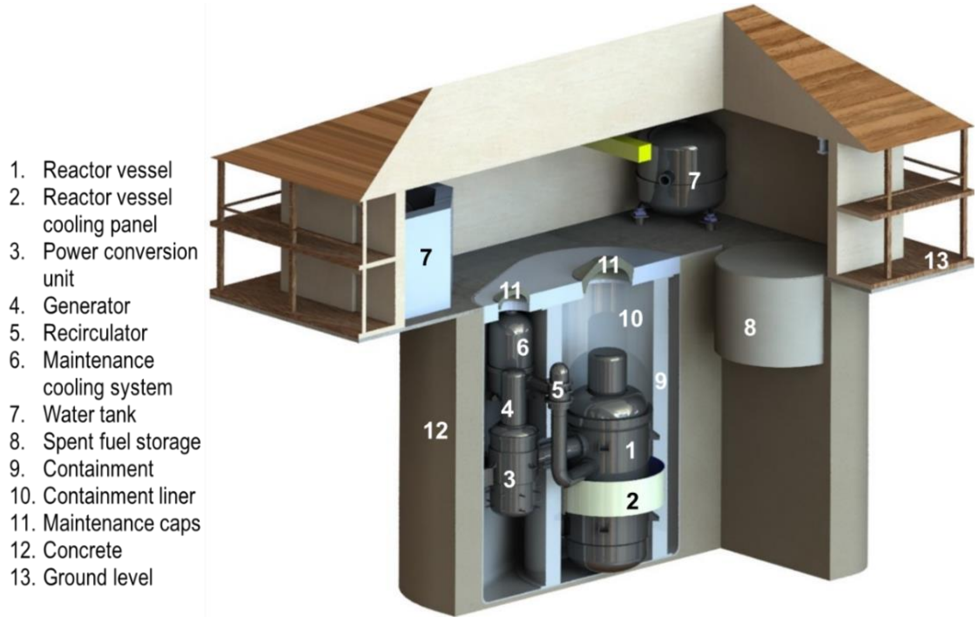

Figure 3. Nuclear Island Components

The fuel assemblies are composed of fuel rods, a central SiC composite guide tube, and spacer grids. The spacer grids will be joined to the central SiC composite guide tube to form a support structure for the fuel rods during handling and operation. The fuel rods are composed of SiC composite cladding, UO_2 pellets, Zr_3Si_2 pellets, graphite pellets, pellet support structure, fission gas plenum, and end plugs. Some major design parameters are given in Table 1.

Table 1. FMR Design Parameters

Parameter	Nominal Value
Plant electrical power	50 MW
Reactor thermal power	112 MW
Core pressure	7 MPa
Core inlet temperature	509 °C
Core outlet temperature	800 °C
Fuel material	UO_2
Number of fuel assemblies	144
Fuel rods per assembly	120
Fuel assembly configuration	Hexagonal
Spacer grid material	SiC composite
Support structure material	SiC composite
Fuel cladding OD (mm)	9.5
Fuel average discharge burnup	~100 GWd/tU
Core average power density	21 W/cm^3

The Reactor Vessel Cooling System (RVCS) is a passive cooling system designed to protect the reactor core and vessel in case of Loss of Coolant Accident (LOCA) accompanied by loss of active cooling. The RVCS system is passive in that cooling water is circulated down to the reactor level through the natural convection created by the temperature gradient between water absorbing the heat from the reactor and the cool reservoir tank. The above grade safety-related structures, such as the RVCS water tanks and control room, will require reinforced concrete structure for protection against external hazards. A maintenance cooling loop, including heat exchangers and re-circulator, will be designed to transfer the heat of the hot helium in the reactor to one of the water towers used by the RVCS through a helium-to-water heat exchanger.

3 FAST MODULAR REACTOR LICENSING STRATEGY

3.1 Licensing Engagement Overview

The U.S. NRC recommends the pre-application regulatory engagement to provide for early identification of regulatory requirements for advanced reactors and to provide all interested parties with a timely, independent assessment of the safety and security characteristics of advanced reactor designs.

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In order to provide a smooth path forward and to support a timely and effective NRC review, the GA-EMS team will engage NRC early in the pre-application process to identify and resolve issues, work collaboratively with NRC and the project team, and submit high-quality topical reports. Early engagement with NRC is an important licensing strategy, with the following objectives.

- Facilitating the mutual understanding of GA-EMS FMR design team and the NRC
- Early identification of regulatory requirements for advanced reactors
- Initiating discussions of some key features and challenges
- Assessment of the safety and security characteristics of FMR reactor design
- Fostering strategic thinking for the development of the FMR systems
- Scheduling the delivery of the application submittals
- Minimizing complexity in the regulatory review process
- Adding stability and predictability in the licensing and regulation of the FMR plant.

GA-EMS will refer to three major documents for the pre-application regulatory engagement:

- The NRC Regulatory Guide (RG) 1.232 establishes guidance for developing Principal Design Criteria (PDC) of non-Light Water Reactor (non-LWR) in support of the regulatory requirements [2]. This RG also describes guidance for modifying and supplementing General Design Criteria (GDC) to develop PDC that address non-LWR design concepts in three categories: Sodium-cooled Fast Reactor Design Criteria (SFR-DC), Modular High Temperature Gas-cooled Reactor Design Criteria (MHTGR-DC), and a design-neutral category, Advanced Reactor Design Criteria (ARDC).
- The development of the pre-application regulatory engagement plan followed the instructions from Nuclear Energy Institute (NEI) guidance document NEI 18-06. This guidance document also provides guidance and candidate topics regarding engagement with the NRC staff prior to submittal of an application for NRC review and approval [3].
- Guidance document NEI 18-04 presents a Technology-Inclusive, Risk-Informed, and Performance-Based (TI-RIPB) process for selection of Licensing Basis Events (LBEs); safety classification of Structures, Systems, and Components (SSCs) and associated risk-informed special treatments; and determination of Defense-in-Depth (DiD) adequacy for non-LWRs [4]. This guidance provides applicants one acceptable method for establishing the aforementioned topics as part of demonstrating a specific design provides reasonable assurance of adequate radiological protection.

- RG 1.233 gives a list of applicable regulations and related guidance for licenses, permits, certifications, and approvals for advanced non-LWR designs [5]. This RG endorses NEI 18-04 as the technology-inclusive methodology and a common approach for selecting LBEs, classifying SSCs, and assessing DiD.

Specifically, GA-EMS will submit principal design criteria, LBE selection, and the safety cases with pre- Probabilistic Risk Assessments (PRAs). In developing these and other licensing submittals, the GA-EMS team will follow RG 1.232, RG 1.233, and NEI 18-04 for the licensing basis and content of applications for licenses, certifications, and approvals.

GA-EMS will analyze the FMR design to demonstrate that it meets the safety objective by following the NRC Licensing Modernization Project (LMP) approach [4] that has been formulated for advanced reactors, further discussed in Section 3.3. The LMP uses probabilistic insights for the selection of LBE; classification of SSCs (to assure that safety-significant components have the pedigree to assure fulfillment of their function); and determination of the adequacy of DiD.

The DiD includes multilayer protection from fission products, for example, ceramic fuel pellets, cladding, reactor vessel and piping, containment, exclusion area, low population zone and evacuation plan, and population center distance. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon.

There are different views on the makeup of these layers, and the number of layers. Two broadly differing views regarding the layers of defense are the following: One view is that the multiple layers are actual physical barriers, as seen in early characterization of DiD. These physical barriers were generally viewed to be the fuel rod cladding, the reactor vessel, and the containment. This view of barriers is more focused on mitigation rather than prevention. The other view is that the layers came to be more functional in nature, and not limited to physical barriers. The layers address both prevention and mitigation and generally involve measures to prevent an adverse event from occurring and mitigating the consequences if the event were to occur.

There are two proposals that define five-layer structures:

- Proposal 1 includes (a) accident prevention, (b) safety systems, (c) containment, (d) accident management, and (e) siting and emergency plans.
- Proposal 2 incudes (a) physical protection against intentional acts, (b) stable operations to limit the frequency of events, (c) protective systems to mitigate initiating events that are both reliable and capable of preventing and mitigating, (d) barrier integrity to ensure adequate barriers to protect from accidental radionuclide release, and (e) protective actions to protect the public should radionuclides penetrate barriers.

Early establishment of the DiD concept ensures that DiD adequacy will be achieved as described in NEI 18-04 and provides some assurance prior to the performance of a full scope PRA and assessment of the Frequency-Consequence target.

In the longer term, the licensing approach for the demonstration and commercialization of the FMR concept is to construct and operate a demonstration unit through two-step 10 CFR Part 50 licensing with prototype testing features. Design certification and combined licenses through 10 CFR Part 52 will be applied for the commercial units, which are expected to be available in the mid-2030s, with a demonstration unit planned for 2030.

3.2 Technical Topic Candidates

The following are possible topics to be discussed with the NRC:

a) Safeguards and Security
   - Design considerations for physical security
   - Detection, assessment and notification vs interdiction and neutralization
   - Security staffing
   - Material control and accounting for unconventional fuel

b) Emergency Planning
   - Alternative approaches to establishing emergency planning zones
   - Dose consequence criteria and analytical methods

c) Fuel Qualification
   - Planned approach to fuel qualification
   - Use of historic/legacy data, including quality assurance implications
   - Testing and analytical approaches
   - Novel fuel forms

d) Seismic
   - Anticipated challenges (e.g., high site seismicity, changing seismic characterizations)
   - Consideration of seismic isolators
   - Beyond-design-basis considerations

e) Flooding
   - Anticipated challenges (e.g., site-specific analysis, tsunami evaluations)
   - Susceptibility of design to flooding impacts
   - Beyond-design-basis considerations

f) Instrumentation and Control (I&C), Digital I&C
   - Common cause failure due to software failure, inter-channel communication, communication between safety and non-safety systems

- Cybersecurity

g) Accident Analysis Methodology
  - Use of phenomena identification and ranking tables
  - Development of codes and models
  - Consideration/use of test data
  - Application of Regulatory Guide 1.203, Transient and Accident Analysis Methods
  - Interface with risk-informed framework
  - Basis for SSC classification

h) PRA and Risk-Informed, Performance-Based Regulatory Framework
  - PRA requirements for 10 CFR 50 applications
  - Use of risk-informed, performance-based regulatory framework for establishment of non-LWR safety basis
  - Potential use of 10 CFR 50.69

i) Quality Assurance (QA)
  - Use of NEI 11-04 and Regulatory Guide 1.28
  - Alternative QA programs and consideration of international standards
  - Timing of QA program approval and implementation, particularly for early development
  - QA program applicability to testing
  - Dedication of legacy data
  - Software validation and verification requirements

j) Concept of Operations
  - Fuel cycle length
  - Refueling and other maintenance/outage approaches
  - Other drivers for operational regimes, surveillance/inspection frequencies, etc.

3.3 LMP Documents

The LMP prepared and provided the following draft white papers on key licensing issues, in the series "Modernization of Technical Requirements for Licensing of Advanced Non-Light Water Reactors," and the NRC staff gave feedback to the LMP on each white paper as noted:

- "Selection of Licensing Basis Events," issued April 2017 (Agency wide Documents Access and Management System (ADAMS) Accession No. ML17104A254), and related staff questions and comments (ADAMS Accession No. ML17145A573)

- “Probabilistic Risk Assessment Approach,” issued June 2017 (ADAMS Accession No. ML17158B543), and related staff questions and comments (ADAMS Accession No. ML17233A187)
- “Safety Classification and Performance Criteria for Structures, Systems, and Components,” issued October 2017 (ADAMS Accession No. ML17290A463), and related staff questions and comments (ADAMS Accession No. ML17319A210)
- “Risk-Informed and Performance-Based Evaluation of Defense-in-Depth Adequacy,” issued December 2017 (ADAMS Accession No. ML17354B174), and related staff questions and comments (ADAMS Accession No. ML18024A595)

4 PRINCIPAL DESIGN CRITERIA

Regulatory Guide 1.232 establishes guidance that non-LWR designers may use to develop PDC in support of the regulatory requirements. This RG describes the NRC’s proposed guidance on how the GDC in Appendix A, “General Design Criteria for Nuclear Power Plants,” of Title 10 of the Code of Federal Regulations, Part 50 “Domestic Licensing of Production and Utilization Facilities” (10 CFR Part 50) apply to non-LWR designs. The GA-EMS reactor design and analysis team will follow RG 1.232 to develop PDC for the FMR, with strategically considering the following:

- The PDC establish high-level requirements for the design basis of SSCs.
- The PDC establish acceptance criteria used in the evaluation of the design.
- PDC development is informed by risk-insights that may be obtained through a PRA.

The team will also justify how the PDC provide reasonable assurance that the FMR can operate without undue risk to the health and safety of the public.

5 FUEL QUALIFICATION PLAN

The FMR fuel is considered to represent a new and unique feature which has a significant bearing on the consequences of the release of radioactive materials. GA-EMS will provide a technical report for the FMR fuel qualification and request NRC review and feedback that the fuel qualification methodology described in the technical report is sufficient to demonstrate the fission product retention capabilities of the FMR fuel. The following sections describe the NRC regulations and PDC that are relevant to the FMR fuel design.

5.1 Regulations Relevant to Fuel Qualification

The regulations that are relevant to the fuel are contained in 10 CFR 50.34 “Contents of applications; technical information,” and 10 CFR 50.43(e) “Additional standards and provisions affecting class 103 licenses and certifications for commercial power.”

10 CFR 50.34

- 10 CFR 50.34(a) defines the minimum technical content of the fuel and all supporting SSCs to be included in a preliminary safety analysis report with the application for a construction permit.
- 50.34(a)(1)(ii)(C) requires information on the extent to which the reactor incorporates unique, unusual, or enhanced safety features having a significant bearing on the probability, or consequences of accidental release of radioactive materials.
- 50.34(a)(1)(ii)(D) requires information on the safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur, with a special attention to plant design features intended to mitigate the radiological consequences of accidents.
- 50.34(a)(2) requires a summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

10 CFR 50.43(e)

This regulation requires that designs significantly different from light-water reactors licensed before 1997 will be approved only if:

- (1)(i) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof; (ii) Interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof; and (iii) Sufficient data 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; or
- (2) There has been acceptable testing of a prototype plant over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If a prototype plant is used to comply with the testing requirements, then the NRC may impose additional requirements on siting, safety features, or operational conditions for the prototype plant to protect the public and the plant staff from the possible consequences of accidents during the testing period.

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5.2 Principal Design Criteria Relevant to Fuel Qualification

The following PDC are most relevant to the qualification of the fuel:

- PDC 10: The reactor core and associated heat removal, control, and protection systems shall be designed with appropriate margin to ensure that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.
- PDC 11: The reactor core and associated systems that contribute to reactivity feedback shall be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Other relevant PDCs that influence the definition of fuel performance requirements include PDC 2, 26, and 34.

5.3 Requests to NRC

GA-EMS will request NRC feedback of the fuel qualification methodology to be described in the fuel qualification technical report. This qualification methodology will be based on a combination of analysis, testing, and operating experience to meet the requirements of 10 CFR 50.34(a)(1)(ii)(C), 10 CFR 50.43 (e)(1), PDC 10, and PDC 16. GA-EMS will request that the NRC review and provide feedback on this methodology and expects that, when the acceptance criteria of the methodology are met, the fuel described in the fuel qualification technical report is qualified for use by applicants for licensing the FMR fuel under 10 CFR 50 and 10 CFR 52.

The design and qualification of the FMR fuel is such that the SAFDLs are met during any conditions of normal operation and Anticipated Operational Occurrences (AOOs). The UO_2 fuel pellets within SiC composite cladding of the FMR fuel produce much less linear power when compared with those of the commercial LWR, and thus provide a better retention of fission products and contribute to a functional containment, as defined in RG 1.232. Qualification of the FMR fuel design will confirm that the fuel meets this fission product containment function.

The NRC's Office of Nuclear Reactor Regulation recently issued NUREG-2246 to provide a fuel qualification assessment framework for use with advanced reactor designs that satisfies regulatory requirements [6]. This report considers the use of Accelerated Fuel Qualification (AFQ) techniques as part of the framework.

The AFQ methodology [7] brings together a combination of advanced, physics-informed nuclear fuel performance modeling and simulation with targeted experiments. The FMR project will conduct specific tasks to qualify the fuel design, including fabrication of UO_2/SiC composite rodlets, irradiation tests, and Post-Irradiation Examination (PIE). The focus of these experiments is to observe and confirm the fuel failure mechanism and draw a threshold line for operation from limited number of integral tests supported by high-fidelity modeling and simulation. The major activities are as follows:

- Fabrication and characterization of UO_2 fuel encapsulated by SiC composite cladding.
- Fuel irradiation tests in the Advanced Test Reactor (ATR) under high-burnup and high-temperature conditions including two different size rodlets. Small rodlets are for the accelerated irradiation. The thermal and mechanical failure mechanism will be investigated from the small rodlet experiments, and scaling analysis will be conducted against the standard size rodlet.
- The transient tests will be conducted in the Transient Reactor Test Facility (TREAT) to investigate the fuel behavior under transients such as the Reactivity-Initiated Accident (RIA) and LOCA. The tests will be conducted with fresh fuels at this phase of the project.
- The minimum efforts in PIE are the measurement of fission gas release from the rodlet to confirm the hermeticity of the SiC composite cladding. The measurements of rodlet dimensional change will be used to validate the fuel performance models and data. The AFQ approach of the FMR fuel will fully utilize the existing database of the UO_2 fuel, developed for commercial reactors as well as tested in research reactors [8, 9], and high Displacement Per Atom (DPA) SiC composite [10] and high-fidelity modeling and simulation technique developed for the ATF [11].

As the engineering-scale tools become well developed, they will be used to finalize the fuel design specification used in the final fuel qualification integral tests. It is expected that the AFQ methodology would significantly reduce the time to qualify the new fuel by reducing the number of costly integral tests and/or by identifying specific separate effect tests.

6 SAFETY ANALYSIS PLAN

6.1 Safety Analysis Codes

MELCOR

MELCOR is a severe reactor accident analysis code developed by SNL for the US NRC. It was initially developed for assessment of boiling-water and pressurized-water reactors and has been further developed for non-water reactors. Accident scenarios include loss of coolant, core heat up, core degradation, core material melting and relocation, hydrogen production, transport, and combustion. With its capability to calculate fission product release and transport behavior, the MELCOR code is a prime tool to estimate the severe accident source terms and to analyze their sensitivities and uncertainties.

GA-EMS will use the MELCOR code for safety analysis and severe accident source terms analysis of the FMR in collaboration with UWM and SNL, respectively.

System Analysis Module (SAM)

SAM is a modern system analysis tool being developed at ANL for advanced non-LWR safety analysis. It aims to provide fast-running, whole-plant transient analyses capability with improved-fidelity for SFR, Lead-cooled Fast Reactor (LFR), and Molten Salt Reactor (MSR). It is being developed as a system-level modeling and simulation tool that has higher fidelity but that is also computationally efficient.

As proposed by the UWM team, the SAM code may be used as a backup software to perform safety analyses for some of the key LBE cases to demonstrate that the FMR design meets the design criteria.

6.2 Safety Analysis Events

It is one of the objectives of this pre-application regulatory engagement plan that selection of LBEs for the FMR design, by following the guidance in NEI 18-04, will demonstrate the effectiveness of the selection process in order to facilitate the determination of risk significant LBEs and SSCs and the evaluation of DiD adequacy. Also, it is very important that assertion be achieved by the FMR design team, that use of the TI-RIPB process for selection of LBEs is endorsed by the NRC policies and compatible with the latest regulatory framework.

GA-EMS has more than 25 years of experience in safety analysis of helium-cooled reactors. Some events selected for safety analysis are based on the previous experience of GA-EMS on the design and analysis of helium-cooled fast reactors [12].

The following categories of events are included in the selection of LBEs.

- AOOs
- Design Basis Events (DBEs)
- Beyond Design Basis Events (BDBEs)

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AOOs are conditions of normal operation that are expected to occur one or more times during the life of a nuclear power unit. Event sequences with mean frequencies of 1×10^{-2}/plant-year and greater are classified as AOOs. AOOs take into account the expected response of all SSCs within the plant.

DBEs are infrequent event sequences that are not expected to occur in the life of a nuclear power unit. Event sequences with mean frequencies of 1×10^{-4}/plant-year to 1×10^{-2}/plant-year are classified as DBEs. For DBEs, the plant must be designed to ensure functions of safety-related electric equipment that ensures the integrity of the reactor coolant pressure boundary; the capability to shut down the reactor and maintain it in a safe shutdown condition; or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures.

BDBEs are rare event sequences that are not expected to occur in the life of a nuclear power plant. Event sequences with mean frequencies of 5×10^{-7}/plant-year to 1×10^{-4}/plant-year are classified as BDBEs.

6.2.1 Loss of Turbine Blade

Gas turbine engines are considered to be highly reliable in that failures in service are rare. A hypothetical accident, loss of a turbine blade, is selected as one of BDBEs. This event results in a turbomachine shutdown following a reactor trip.

The loss of a turbine blade event is initiated from the steady state full power operation conditions of the FMR. The turbomachine shaft high vibration signal will result in a protection trip of the reactor. It will also cause other actions of the reactor control and protection system, as follows:

- Generator load decreases rapidly and is decoupled from the grid when the load reaches a minimum setpoint
- Reactor trip
- Control signal closes Helium Service System

The failed blade will not cause consequential damage of any additional components such as valves, disks, or nozzle vanes in the remaining stages of the gas turbine, and neither will the broken blade part cause any damage of the instrumentation and control system components.

The important transient parameters of interest are as follows:

- The maximum pressure differential across the reactor core
- The potential for flow reversals in the reactor core
- The maximum temperatures in the systems, structures, and components

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6.2.2 Precooler Tube Rupture

The precooler tube rupture can cause water ingress into the core. The objective of the calculation is to estimate the amount of water that can enter the primary system due to a precooler tube rupture. Water ingress into the FMR core can cause reactivity increase. Key parameters of interest during this event are as follows:

- The rate of moisture ingress into the vessel system
- The transport of moisture to the reactor core
- The primary and secondary system pressure and temperature response during the transient

6.2.3 Depressurized Loss of Forced Cooling

Depressurized Loss of Forced Cooling (DLOFC) is initiated by a breach of the primary coolant pressure boundary. This event is characterized by a primary system depressurization and elevated system temperatures.

A spectrum of primary coolant leak sizes will be considered to determine a bounding case. This is a design basis event that is considered Category III, potentially severe accidents of extremely low probability, postulated to establish the performance requirements of engineered safety features. A study of the accident frequency is required to make the final classification of this accident. The key result of the analysis of this event is the peak fuel temperature.

The event sequence of DLOFC is as follows:

- Breach of the primary coolant pressure boundary causes primary coolant depressurization.
- Reactor trips are automatically based on either low primary coolant pressure or high radiation levels or high pressure in the Containment. All the operating control rods are inserted making the reactor subcritical.
- Power Conversion System is assumed to fail immediately after and independently of the initiating event.
- Maintenance Cooling System is also assumed to fail to start immediately and independently, causing a complete loss of all active cooling systems.
- Reactor Vessel Cooling System, which is always in operation, passively removes the core decay heat from the reactor vessel by conduction and radiation to the air naturally circulating within the RVCS.
- Containment leaks at its designed leak rate a ground level mixture of air, primary coolant helium, and radionuclides to the environment.

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6.2.4 Subcriticality Analysis

It is imperative that the core subcriticality be maintained after the reactor cooling is transferred to long-term cooling with RVCS. This demonstrates conformance to the long-term cooling acceptance criteria.

6.2.5 Control-Rod Withdrawal Accident

A control rod withdrawal accident is an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of a control rod which results in a power excursion. Such a transient can be caused by a malfunction of the reactor control or rod control systems. This can occur with the reactor subcritical, at hot zero power, or at power.

A number of faults are postulated that result in reactivity and power distribution anomalies. Reactivity changes could be caused by control rod motion or ejection. Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by static means such as fuel assembly mislocation.

The following incidents are to be considered for the safety analysis:

- Uncontrolled control rod withdrawal from a subcritical or low-power startup condition
- Uncontrolled control rod withdrawal at power
- Control rod ejection accidents

The uncontrolled control rod withdrawal at power event is defined as an uncontrolled addition of reactivity due to the withdrawal of a control rod during power operation.

The maximum reactivity insertion rate is that occurring with the simultaneous withdrawal of the combination of two control rods having the maximum combined worth at maximum speed. Should a continuous rod withdrawal accident occur, the transient is terminated by the automatic features of the protection and safety monitoring systems.

With a source range high neutron flux reactor trip, the trip function is actuated when two out of four independent source range channels indicate a neutron flux level above a preselected, manually adjustable setpoint. It may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when the coincident two out of four intermediate range channels indicate a flux level below a specified level.

It is anticipated that the most severe radiological consequences result from the complete rupture of a control rod drive mechanism housing. Radiological consequences will be reported only for the limiting case.

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7 SOURCE TERM CALCULATION

7.1 Source Term Regulation Framework Overview

Regulatory Guide 1.183 ("Alternative Radiological Source Terns for Evaluating Design Basis Accidents at Nuclear Power Reactors") was developed by NRC to support the final rule that amended 10 CFR Part 21, 50, and 54. The regulatory text that follows in 10 CFR Part 50 is limited to using the phrase "accident source term" in relation to design basis radiological consequence analyses.

This focus on accidents appears in several additional requirement statements in 10 CFR 50.34. For example, 10 CFR 50.34(f)(2)(VII) states that an application shall:

> Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment.

The next use is in 10 CFR 50.67, which details requirements should an operating plant license holder wish to revise their current accident source term used in their design basis radiological analyses:

> The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses, or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

10 CFR 50.2 defines source term as:

> The magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release.

NUREG-1465 proposed revised source terms to the containment based on more realistic assumptions as related to release duration, release quantity, fission product aerosol retention, and chemical forms. The revised source term as proposed in NUREG-1465 is primarily based on experiments and analytic studies applicable to low burnup (less than 40 GWd/t) UO_2 fuel.

7.2 FMR Source Term

Due to adequate shielding around the FMR core and spent fuel containers, fission products in the fuel pose no threat to either plant personnel or the nearby population. The safety of the FMR nuclear power plant is therefore dependent on its fuel's ability to retain fission products under all expected reactor operating conditions. In the FMR core, the principal containment barrier is the SiC composite cladding.

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The objective of the FMR source term calculation is to define a specific FMR accident source term for regulatory application. The approach to be used in this effort is therefore a best estimate source term calculation, rather than a bounding source term determination that can be derived from some existing analysis of the available inventory of fission products. Using the Booth diffusion model, the MELCOR code will need information on fission product release rate from overheating and melting of the FMR fuel. Temperature dependent diffusion coefficients derived from experimental data will be the input to the MELCOR calculations.

Typical assumptions related to the release of radioactive material from the fuel and containment are as follows [13]:

a) Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the primary reactor containment. Ninety-one percent of this 25 percent is to be assumed to be in the form of elemental iodine, 5 percent of this 25 percent in the form of particulate iodine, and 4 percent of this 25 percent in the form of organic iodides.
b) One hundred percent of the equilibrium radioactive noble gas inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the reactor containment.
c) The effects of radiological decay during holdup in the containment or other buildings should be taken into account.
d) The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on an individual case basis.
e) The primary containment should be assumed to leak at the leak rate incorporated or to be incorporated in the technical specifications for the duration of the accident. The leakage should be assumed to pass directly to the emergency exhaust system without mixing in the surrounding reactor building atmosphere and should then be assumed to be released as an elevated plume for those facilities with stacks.

8 PROBABILISTIC RISK ASSESSMENT STRATEGY

The LMP PRA approach report prepared by Idaho National Laboratory contains the historical background, technical justifications and supporting information, and implementation guidance for creating a PRA model fit for providing insights into plant behavior for a given phase of design development for advanced non-LWRs. The LMP PRA approach is reactor technology inclusive and makes use of technology-inclusive risk metrics. The PRA is recommended to be introduced at an early stage of design to incorporate risk insights into early design decisions. The PRA models are initially limited in scope and of a coarse level of detail as constrained by available supporting information. The scope and level of detail of the PRA models are increased as design and site information become available and as design features to protect the plant against internal and external hazards are defined. The TI-RIPB decisions supported by the PRA and

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deterministic safety approaches are reviewed and revised as the risk model definition is brought into focus. The technical adequacy of the PRA for use in LMP applications is addressed by application of the ASME/ANS PRA Standard for Advanced non-LWRs, ASME/ANS RA-S-1.4-2013.

The GA-EMS team will follow the guidance in ASME/ANS RA-S-1.4-2013 to establish PRA technical acceptability. Key technical elements of the guidance will be addressed based on the high-level and supporting requirements of the standard as tailored for the application. Sufficient justification for its use will also be provided, in order to confirm that the design meets the applicable Commission policy-level goals related to PRA and severe accidents.

A PRA white paper will be prepared to include the following contents.

- How to establish appropriate risk metrics to be used in the PRA
- How to address the risk associated with any multi-unit or multi-module aspects of the design
- How to address the reliability of inherent features or passive safety systems that are relied upon for the design or safety case
- How to address the risk associated with different sources of radioactive material both within and outside of the reactor core
- How to identify the appropriate criteria for determining the risk significance of various elements modeled in PRA including initiating events, SSCs, and operator actions by performing PRA importance analyses based on appropriate importance measures
- How to perform sensitivity analyses and how to analyze the uncertainties in the PRA

9 QUALITY ASSURANCE PROGRAM

A Quality Assurance Program (QAP) will be established early in the development of the FMR technology. The GA-EMS team will establish and implement the FMR QAP pursuant to 10 CFR 50 Appendix B, 10 CFR Part 21, ASME NQA-1-2015, and RG 1.28 [14]. The team will also follow the implementation guidance found in NEI 11-04A [15]. NEI 11-04 is structured as a template for use in developing an applicant-specific Nuclear Generation (NG) Quality Assurance Program Description (QAPD) required as part of Early Site Permit (ESP) and Combined License (COL) applications. The QAPD report will provide details of the quality assurance program, as well as quality control of SSC. It may include a Quality Management Plan (QMP) that establishes the quality assurance policy and assigns major functional responsibilities for nuclear power reactor design certification activities conducted by or for GA-EMS. The QMP describes the methods and establishes quality assurance and administrative control requirements that meet 10 CFR 50, Appendix B and 10 CFR 52.

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10 SAFETY CLASSIFICATION

The safety classification process follows the methodology described in NEI 18-04, Revision 1. This methodology is endorsed by RG 1.233, which gives a list of applicable regulations and related guidance for licenses, permits, certifications, and approvals for advanced non-LWR designs. This process will determine the safety classification of each SSC, the level of quality assurance that is required, and the function of the SSC in mitigating and preventing LBEs. It will be used to identify appropriate programmatic controls and determine the appropriate scope and level of detail for information provided in applications.

Criteria are provided to classify SSCs into three safety classes: Safety-Related (SR), Non-Safety-Related with Special Treatment (NSRST), or Non-Safety-Related with No Special Treatment (NSR). Based on the SSC safety functions in the performance of both prevention and mitigation functions, the developer will assign reliability and performance targets which help ensure that selected special treatment requirements are performance-based.

The FMR design is expected to provide enhanced margins of safety while using simplified, inherent, passive, or other innovative means to accomplish its safety and security functions. Programmatic defense in depth will be discussed to ensure its adequacy, including measures to increase confidence in SSC performance during operation and throughout the life of a plant, by quality assurance control, operational procedures and training, and preparedness for emergency plan protective actions.

11 SUBMITTAL SCHEDULE

In coordination with the NRC staff, topics can be prioritized and optimized to address design alternatives or those topics most important to the overall project plan. The schedule for key deliverables of the FMR project is planned as follows.

11.1 QA Program Description Topical Report

A topical report of the GA-EMS FMR QAPD is a document that describes the policies, organizations, design control, procedures, and methods that constitute the QA program relative to the US domestic licensing requirements for nuclear power plants.

Topical report submittal: June 2022

11.2 Principal Design Criteria Report

The PDC for the FMR design will be one early topical report to be submitted as one of the pre-application engagement activities.

Topical report submittal: June 2022

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11.3 Fuel Qualification Plan

This pre-application submittal will provide an overview of fuel performance analysis methodology and fuel qualification plans. Candidate topics to be included in the fuel design description are as follows:

- Fuel: This information could include fuel type; key material and structural parameters; important design constraints (e.g., burnup, heat rate, etc.); discussion of the status of fuel qualification and/or existence or planned development of qualification data; and important or novel aspects of fuel handling.
- Technology Readiness: This information could include fuel testing, validation, and verification.
- Fuel Cycle Considerations: This discussion could include fuel cycle infrastructure, front-end fuel cycle development, or novel approaches to fuel management methodology.
- Means of Radionuclide Retention: This information could include novel approaches to radionuclide retention.

Technical report submittal: January 2023

11.4 Mechanistic Source Term

The source term will include radiological source terms for shielding design. This pre-application interaction will provide an overview of the radionuclide transport methodology.

Technical report submittal: June 2023

11.5 LBE Selection

Selection and evaluation of LBEs will be conducted by following LMP technical requirements for licensing of advanced non-LWRs.

White paper submittal: December 2023

11.6 Safety Approach and Mini-PRA

This pre-application licensing submittal will provide an overview of the safety approach and mini-PRA methodology.

White paper submittal: May 2024

11.7 Safety Classification

The SSCs will be categorized into three classes: SR, NSRST, or NSR. Reliability and performance targets will be assigned to ensure that selected special treatment requirements are performance-based.

White paper submittal: May 2024

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