Attachment 2

Demonstration Sodium-Cooled Fast Reactor GE-PRISM

GE Hitachi Nuclear Energy

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Executive Summary

GEH recommends the pool type sodium fast reactor, PRISM, in "Mod-A," single reactor configuration for a DOE Demonstration reactor mission. This 471 MWt reactor would be coupled to a superheated Rankine power cycle with a single helical coil steam generator producing 165 MWe (35% efficiency). Located around the PRISM reactor will be several ancillary non-reactor technology modules to support testing and demonstration needs of other reactor technologies such as molten salt reactors and super-critical CO_2 power conversion. Each of these would be supported by output from the PRISM reactor.

No U.S. advanced reactor technology has more licensing, testing, design or operation basis than PRISM. PRISM's extensive design development, state of knowledge, operational experience, and conservative design approach (based on the use of technologies proven at EBR-II) provides the highest potential for an aggressive plant project schedule and successful project execution. PRISM has an existing test plan (informed by EBR-II and FFTF operations) which includes instrumentation identified for analytical code validation and startup testing. The regulatory approval process will benefit from the extensive NRC pre-application review performed on the PRISM design between 1987 and 1994 (see NRC document NUREG-1368). These extensive efforts represent over seven years of critical path time that can be avoided versus selection of a different design that has not completed such work.

PRISM supports all of the most promising fuel cycles identified in the recent multi-year DOE fuel cycle evaluation and screening study. The PRISM demonstration reactor will start up on U-Zr fuel which has a Technology Readiness Level (TRL) of 8. U-Zr fuel can be quickly followed by U-Pu-Zr fuel (TRL 7) and fuel bearing minor actinides to support fuel cycle closure using the proliferation resistant pyroprocess. In addition to metallic fuels, PRISM will support irradiation testing of MOX, nitride, thorium and other fuel types with a peak fast flux level in excess of ~3×10^{15} n/cm^2-s.

The PRISM reactor is the most competitive advanced reactor design for near term demonstration. The licensing basis for PRISM is defined. The technology basis is strong. No major R&D such as material development is required. Scaling issues to commercial size are eliminated because the demonstration will be commercial scale. The reactor is cost competitive due to its inherently passive safety systems and small size. Its construction is simplified by its modular reactor building and rail-shippable, factory fabricated reactor vessel. With an aggressive schedule, PRISM has the highest potential to be demonstrated in the 2035 time frame. PRISM's safety performance minimizes safety system complexity and associated capital and operational costs. The design is based on extensive studies conducted to optimize and balance economies of scale with modularization and factory fabrication.

1.0 Introduction

1.1 Background of the Study and Vendor/Designer Input

The U.S. Department of Energy has chartered a national laboratory group to assess the readiness of advanced reactor technologies and to recommend which technologies have sufficient maturity for test or demonstration reactor missions. The information provided herein explains why the PRISM reactor is the right technology for the demonstration reactor.

Section 2 discusses objectives and motivation for selecting the PRISM design as the advanced demonstration reactor. Section 3 discusses technology readiness of PRISM. Section 4 discusses the licensing strategy and the benefits of previous and current work that can be applied to this project. Section 5 provides technical design and performance highlights of the PRISM demonstration plant. Section 6 discusses the safety characteristics of PRISM. Section 7 provides details on economics and schedule. Finally, section 8 highlights the results of the assessment of PRISM against the demonstration reactor metrics.

The present GEH state of knowledge on advanced reactors leverages four previous DOE programs (LMR, ALMR, GENIV and GNEP). For deployment of the PRISM demonstration system, GEH envisions a broad coalition of industry partners along with problem solving expertise of national laboratories to deliver the project on-time and on-budget.

The PRISM reactor can meet the demonstration reactor requirements and the following DOE objectives: "to achieve greater levels of safety and resilience, flexibility of use, sustainability and construction or operational affordability." Given that the PRISM reactor has benefitted from what is probably the largest, and most long-term U.S. investment in advanced reactors, it is expected to considerably reduce the technical risks usually associated with advanced reactor technologies.

1.2 PRISM Design Summary Description and Main Attributes.

The PRISM MOD-A reactor is a 471 MWt fast reactor that uses sodium coolant and metal fuel. The reactor and steam generator are located upon a single seismic isolation pedestal. The reactor uses a superheated steam energy conversion cycle, with a reactor coolant outlet temperature of 930 °F, producing 35% net cycle efficiency. The demonstration reactor core is capable of achieving 12 to 24-month fuel cycles.

Specific design features include a pool configuration for the primary sodium, and the use of electromagnetic pumps for moving sodium. The pumps have already been tested at full scale. The reactor has two intermediate heat transfer loops coupled to a single helical coil steam generator. Transportability of reactor components to the plant construction site is expected to be enhanced by the use of modular construction which is pre-sized for trucks and rail cars. All components, including the reactor vessel and the guard vessel, would be rail shippable thereby reducing construction costs.

The reactor design uses simple, passive safety systems for improved safety at lower cost. For example, the Auxiliary Cooling System (ACS), a backup residual heat removal system, is simply an air duct surrounding the steam generator which removes heat by forced or natural circulation of atmospheric air. Likewise, the Reactor Vessel Auxiliary Cooling System (RVACS) has no moving parts and removes residual heat via natural circulation of atmospheric air from the outside of the reactor vessel without operator action or external power. Both safety systems reduce capital and O&M costs compared to their active system predecessors.

Adjacent to PRISM would be several ancillary non-reactor demonstrations supporting the test and demonstration needs of other reactor technologies such as molten salt reactors and super-critical CO_2 power conversion rigs. Each of these would be supported by the thermal and electrical energy from the PRISM reactor, if desired.

The reactor core is designed to provide an appropriate environment of a fast spectrum in both energy level and flux. Thus, it is capable of supporting some of the material and fuel development testing for the development of other advanced reactors.

The reactor is at commercial size. Therefore, its transient performance will be directly applicable to the licensing process for follow-on commercial units. The PRISM reactor system is designed to provide feedback on design features, construction technology, and operational performance, thereby reducing regulatory and financial risks.

2.0 Demonstration Reactor Objectives and Motivation for PRISM Selection

2.1 Demonstration Facility Objectives

GEH is targeting the following objectives for its proposed demonstration reactor:

- Design and construct a reactor for further development of PRISM and advanced reactor concepts in the areas of safety, operations, and economics;
- Validate advanced reactor passive features, namely: reactor control, decay heat removal, accommodation of reactor vessel leaks, and in-vessel used fuel storage for initial cooling;
- Validate performance and safety analysis codes for commercial licensing by inducing controlled transient conditions;
- Provide the ability to swap out key components with alternative components to validate specific demonstration tests;
- Introduce a comprehensive seismic isolation pedestal to demonstrate seismic isolation of a nuclear reactor facility for reduced equipment qualification costs and increased safety;
- Demonstrate full acceptance of metal fuel and cladding for U.S. commercial application;
- Produce electricity using a licensable, passively safe, and reliable liquid metal cooled reactor as a heat source;
- Provide feedback on licensing, design, construction and operations to inform future commercial decisions.

2.2 PRISM Demonstration Reactor Solution

PRISM's extensive design development, operational experience, and conservative design approach based on the use of technologies proven at EBR-II provides the highest potential for an aggressive plant project schedule and successful project execution. PRISM possesses a high level of design detail backed by testing and operational experience which is a **key advantage**.

Licensing will benefit from the extensive NRC pre-application review performed on the PRISM design between 1987 and 1994 and subsequent work; this is possibly six to ten years of critical path time that could be avoided versus designs that have not yet completed a pre-application review. The "Preliminary Safety Information Document," prepared for pre-application review, describes the PRISM design and safety case in more than 2,500 pages. Subsequent work added ESBWR design certification experience into the precursor documents for a similar PRISM design certification application.

The PRISM design has evolved through various stages of development and testing. The design has been refined through perhaps the largest collection of trade studies ever carried out for a reactor design. PRISM technology is based on an extensive U.S. based advanced reactor testing database that captures:

- Fuel and safety testing at EBR-II and FFTF reactors,
- Passive heat removal testing at Argonne National Laboratory,
- Fuel fabrication at Idaho National Laboratory,
- Component testing of scaled steam generators and full scale electromagnetic pumps at the Engineering Technology Energy Center.

PRISM trade studies, testing history, and the safety evaluation in NRC NUREG-1368, have led to design improvements over time for improved licensability and associated return on investment for U.S. taxpayers and private investors.

The PRISM reactor is relatively small (165 MWe), and it is in commercial form. **This is a key advantage** of PRISM because it essentially eliminates significant regulatory review of follow-on units, while reducing cost, and schedule risks of scaling up a small scale reactor to commercial scale.

PRISM's selections of fuel, materials, and coolant are relatively conservative with respect to technology innovation, reflecting an appreciation for the conservative nature of the new nuclear plant environment. A PRISM development project can catalyze the NRC regulatory licensing process, reinvigorate the supply chain, and in the end convert decades of research into actionable intelligence for licensing and commercialization. GEH envisions the participation of a broad U.S. industry and national laboratory team to bring the project to fruition.

The PRISM reactor is flexible in meeting power grid demands. The PRISM demonstration plant, operating in, "turbine leading operation," can instantaneously respond to changes in grid load demand including up to a 10% step change in power. Further the demonstration plant is designed to be automatically controlled for daily load following at a maximum rate of 2% per minute for changes up to 10% of rated power. A rate of <1% per minute is normally sufficient to meet changes in load demand. Step load changes up to 10% (of reactor electrical output) are automatically controlled at rates up to 60% per minute. No direct operator control actions are required for a PRISM reactor to load follow.

Modern regulatory strategies rely heavily on risk-based decisions and thus a comprehensive probabilistic risk assessment (PRA) of the nuclear plant is necessary to support licensing activities. GEH is presently upgrading a PRA of the PRISM plant to meet current standards and will complete this project by the end of 2017. **The updated PRISM PRA is another key advantage** that can give the demonstration project a head start in licensing.

2.3 Supporting Future Development of Other Advanced Concepts

DOE-NE's Technical Review Panel (TRP) evaluated the viability of various advanced reactor types and documented R&D needs for commercialization. The TRP report identified three "need areas," that apply to a majority of the advanced reactors reviewed:

1. *"Development of licensing approaches for advanced reactor concepts: development of advanced safety analysis tools and the development of a common verification and validation framework for these tools."*
2. *"Accelerated development of Brayton cycle technologies: This will involve efforts to accelerate the demonstration and deployment of Brayton cycle technologies. That program should focus on both the electricity producing technologies and on the coupling to the various advanced reactor technologies."*
3. *"Development of validated advanced reactor analysis methods: This will involve the development of advanced neutronics, thermal-hydraulics, and mechanical analysis tools, and their validation to modern standards."*

PRISM addresses these three need areas as follows:

1. **Licensing:** GEH developed the passive safety licensing approach for PRISM, as well as for its Generation III+ light water reactor (ESBWR). The regulatory framework that was achieved has resulted in licensing approvals for both GEH's Advanced Boiling Water Reactor (ABWR) and Economic Simplified Boiling Water Reactor (ESBWR) designs. The development of advanced safety analysis tools was completed for PRISM during the years that DOE's Advanced Liquid Metal Reactor (ALMR) program was active. The advanced reactor community would benefit from having GEH continue with an "ALMR" type reactor that was evaluated by the NRC in NUREG-1368. Reengagement with the NRC will lead to the faster regulatory policy decisions (which NRC representatives spoke of during the September 1-2, 2015 DOE/NRC Licensing Workshop). GEH believes that such reengagement will be the fastest and surest way to achieve an advanced reactor regulatory framework. GEH (with the support of the U.S. national laboratories) plans to continue to advance safety analysis tools along with the verification and validation framework. Use of PRISM provides clarity regarding what is required to establish the regulatory framework and can be leveraged to address the following important issues:

a. Implementation of Defense-in-Depth for advanced reactors;

b. Passive system behavior and reliability;

c. Establishment of mechanistic source terms;

d. Licensing basis event selection; and

e. Size of Emergency Planning Zone.

Information derived from a detailed examination of the foregoing issues could be utilized by other advanced reactor concepts as well.

2. **Brayton Cycle:** For purposes of promoting the demonstration and deployment of Brayton cycle technologies, the PRISM solution can use an onsite 'energy park' concept. The energy park is intended to support other advanced conversion systems, such as the Brayton Cycle, by reserving footprint in the reactor yard for testing and providing heat and power to these systems.

3. **Reactor Analysis Methods:** The PRISM reactor program can be configured for the collaborative development of advanced neutronics, thermal-hydraulics, and mechanical analysis tools, as well as their validation to modern standards. These tools are the most mature with sodium cooled systems. Three Licensing Topical Reports: Methods, Validation, and Application would establish the approved safety analysis methodology at the outset of the project. Neutronics, thermal-hydraulics, and mechanical analysis tools would be improved in conjunction with NRC interactions to support commercial licensing requirements.

PRISM can enhance the development of these reactor types by providing the following:

Gas-Cooled Reactors:

1) Testing of fuel, fuel coatings and fuel cladding systems with the objective of developing a fuel (including high burnup fuel for actinide management) that can withstand high burnup, high damage, and high temperature. Irradiation test assemblies in the PRISM reactor core can provide various flux levels to meet testing needs;

2) Demonstrating the Reactor Cavity Cooling System or the Direct Reactor Auxiliary Cooling System concepts; and

3) Demonstrating Brayton Cycle conversion systems by supplying heat and electrical power generated from PRISM within the energy park.

Lead or Lead-Bismuth Eutectic-cooled Reactors:

1) Testing of irradiated materials in fast spectrum to help evaluate erosion/corrosion mechanisms of irradiated materials so that the important variable (neutron damage) is better understood for the implementation of corrosion control;

2) Testing of both oxide or nitride fuel; and

3) Demonstrating performance of large components in flow loops located within the energy park inputting heat and electrical energy generated by PRISM.

Molten Salt -Cooled Reactors (MSR):

1) Testing of structural materials at elevated temperatures using PRISM's irradiation capability in order to enhance structural material development (subject to additional analysis, it may be feasible to support the thermal to intermediate neutron irradiation in the reflector and shield portions of the test reactor core);
2) Testing of decay-heat-cooling systems for MSR designs that are below ground by modifying the PRISM RVACS so that vertical testing can be done at scale; and
3) Demonstrating Brayton Cycle conversion systems by supplying heat and electrical power generated from PRISM within the energy park.

3.0 Technology Readiness of Design

3.1 Technology Development History

PRISM is heir to the revolutionary work of Walter Zinn, Charles Till, Leonard Koch, and other atomic energy pioneers. The technology investment by the U.S. government in sodium reactors over time has been substantial (Clementine, EBR-I, EBR-II, SEFOR, and FFTF) and has led to the following four key scientific achievements:

- Inherent safety in reactor control and subsequent decay heat removal;
- Metallic fuel for a superior transuranic recycle;
- Ability to use transuranics as a fuel source; and
- Computational tools and standards for licensing.

The GEH PRISM design effort originated in the early 1980's. The first PRISM design was a small moveable reactor that was slightly larger than its predecessor, EBR-II. After the initial GE program, subsequent studies were conducted with DOE support that began in 1984. The PRISM design improved during the DOE-sponsored ALMR program with the introduction of the MOD-A and later the MOD B designs. PRISM MOD-A is 471MWt, with a vessel diameter of ~20 feet so that it could be shipped by rail. However, with the level of power output desired by U.S. power producers at the time, three reactor modules would be required to drive one turbine using MOD-A. Power producers preferred the operational characteristics of a larger, two-reactor unit to the benefits of rail shipment during plant construction. Therefore, the MOD-B design enlarged the reactor vessel to 30-foot diameter, which almost doubled thermal output to 840MWt. With this demonstration reactor mission as a launch platform, however, GEH recommends the smaller MOD-A design.

3.2 Technology Maturity

Technology Readiness Level (TRL)

During the DOE's Global Nuclear Energy Partnership (GNEP), GEH evaluated thirty-two separate technology elements for PRISM. Twenty-one are in the areas of System Technology and System Safety, and these are expected to be addressed in the licensing process.

System Technology consists of: (a) Advanced Components and Systems, (b) Advanced Instrumentation and Control, and (c) Advanced Technology. System Safety consists of Reactor Safety and Fuel Cycle Safety. The TRL scale definition varies. PRISM design development and testing has focused on increasing technology readiness which is most critical for commercialization and safety. The design has been refined through perhaps the largest collection of trade studies ever carried out for a U.S. advanced reactor design underpinned by an extensive testing database. The trade studies and testing program iteratively reviewed and incorporated state-of-the-art technology, which improved competitiveness.

The key PRISM components are: seismic isolation, in-vessel fuel transfer machine, electromagnetic pumps, helical coil steam generator, and core fuel. They have a TRL range of 7 to 9 with an average value of 7.8. Other important components, specifically, reactivity control, reactor module, and materials & structures have an average TRL of 8. It is important to note that due to the non-corrosive nature of sodium, no significant material development program is required for deployment. Additionally the long development and testing history of PRISM systems eliminates technology issues for licensing which would slow the deployment of less mature designs.

Expected TRL Advancement

The PRISM design will advance to TRL 9 upon completion of safety testing. To be specific, the two unique commercial components that are expected to receive the largest "boost" in TRL are the self-cooled electromagnetic pumps and the super-heated steam generator. Both the pump and steam generator would be at commercial scale. The PRISM reactor core is capable of supporting materials and fuel development for other technologies such as the MSR, LBE reactor, and gas-cooled concepts.

Scalability Issues and Fabrication Options

The GEH design removes future scalability issues because the demonstration reactor is at commercial scale. Scaling objectives are anticipated to be replaced by cost, performance, and fabrication efficiency objectives.

4.0 Licensing, Development, and Deployment Plans

4.1 Licensing Strategy

The PRISM reactor can be licensed as a "prototype plant," as specified in 10 CFR 50.43(e), which states:

> *"... applications for a design certification, combined license, manufacturing license, or operating license that propose nuclear reactor designs which differ significantly from light-water reactor designs that were licensed before 1997, or use simplified, inherent, passive, or other innovative means to accomplish their safety functions, will be approved only if:*
>
> 1. *The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;*
> 2. *Interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof; and,*
> 3. *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,*
>
> *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."*

This prototype reactor licensing strategy is advantageous when using PRISM for the demonstration because of the significant design detail, PRA and regulatory review work that has already been performed. This gives the PRISM licensing effort a head start versus other reactor options.

This licensing approach requires the following provisions:

- Prototype Licensing Agreement: An agreement by the NRC, which establishes the rules and guidelines to be adhered to in prototype licensing. Defines administrative matters, the licensing process, and criteria by which the design will be evaluated against. This document draws from policy statements, whitepapers and past non-LWR licensing experience to reduce licensing uncertainty. The document will place particular emphasis on establishing closure on potential "...additional requirements on siting, safety features, or operational conditions for the prototype." As part of defining the licensing process, the agreement will call for submission of a unique Prototype Design Report and Prototype Test Plan which is described below.
- Prototype Test Plan: This plan describes how specific testing or conditional operating procedures will be used to supplement or replace safety analysis to prove that each advanced reactor safety feature and the integrated safety strategy are adequately analyzed. This plan assists Prototype Licensing Agreement in defining the licensing of the prototype and in addition it identifies testing to support commercial design certification.

- Prototype Design Report: This document combines the Preliminary Safety Analysis Report (PSAR) of 10 CFR Part 50 with additional design details traditionally reserved for the Final Safety Analysis Report (FSAR). This document includes the Prototype Test Plan and provides additional details on operations such that the Prototype Licensing Agreement can be signed prior to construction.
- Certification Test Report: This report is issued once all the tests validating the safety features and performance of the demonstration reactor are complete. Since the demonstration reactor is nearly a carbon copy of follow-on commercial plants this report will be referenced in certification of the design under part 52.

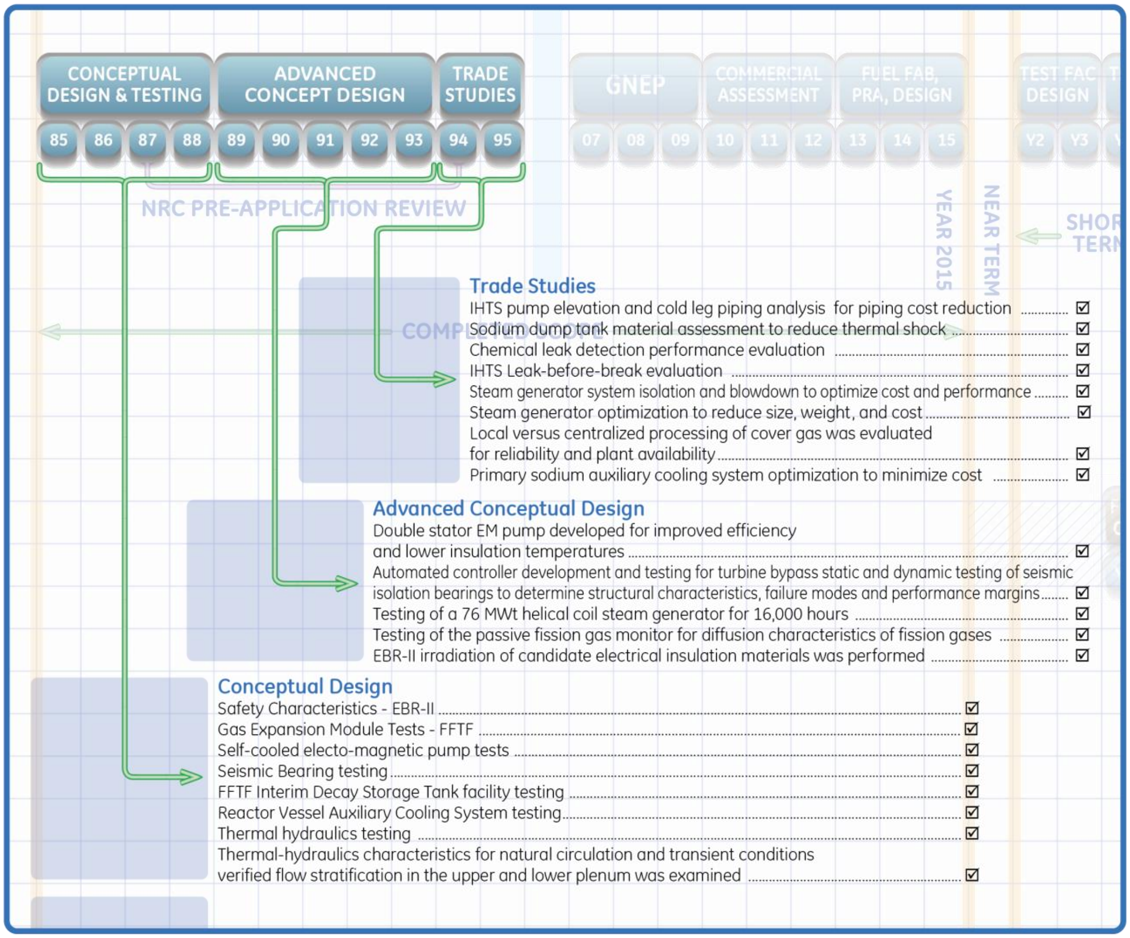
This approach reduces uncertainty by obtaining up front agreement on the licensing process and licensing criteria. It blends the benefits of Part 50 and Part 52 licensing by presenting a well-defined design at the PSAR stage but allows construction to start early. Finally the as built commercial scale demonstration unit will facilitate rapid design certification to be referenced by follow-on units under part 52 licensing.

The NRC's Pre-application Safety Evaluation of the ALMR (NUREG-1368) concluded:

"the staff, with the ACRS in agreement, concludes that no obvious impediments to licensing the PRISM (ALMR) design have been identified."

4.2 Current Development & Licensing Status

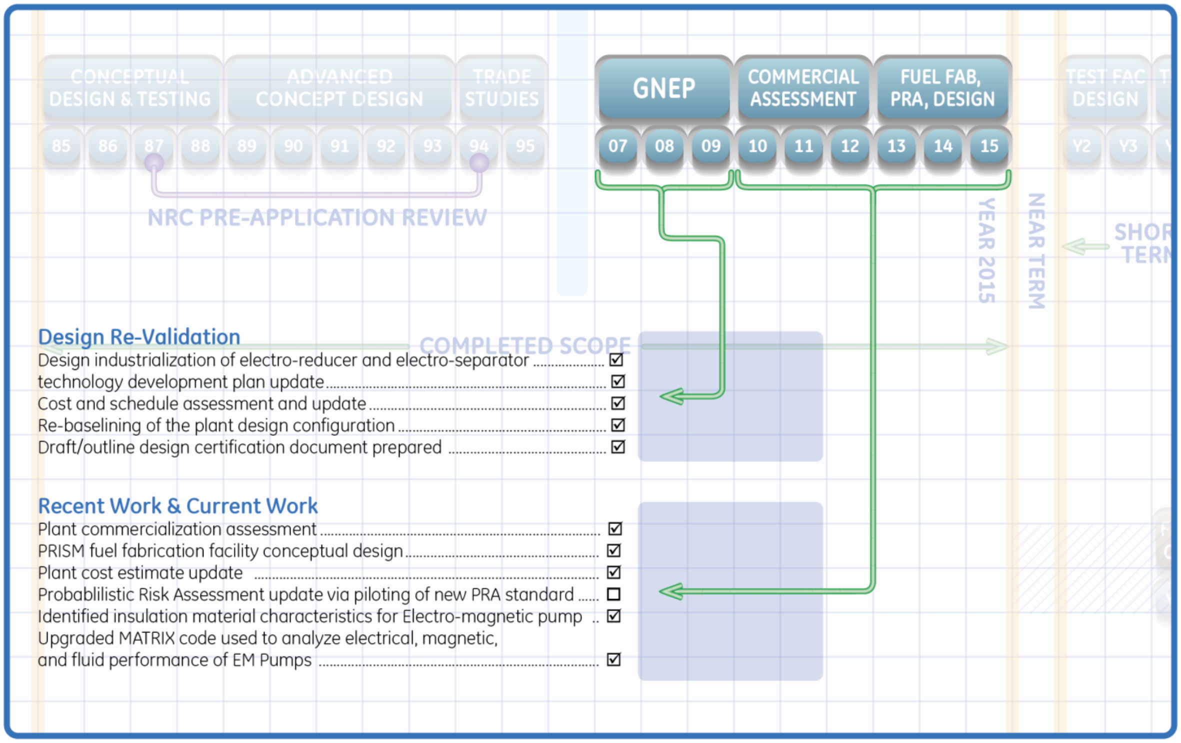
A summary description of PRISM development is shown below:



In 1986, GE submitted to the NRC its Preliminary Safety and Information Document, which included a detailed outline of research and development programs and activities, either underway or planned, necessary to support the licensing effort using 10 CFR Part 50. The majority of the research and development underway was being conducted in cooperation with the U.S. national laboratories. The technology development work for PRISM was identified in four phases: (1) Feasibility tests (conceptual design), 1985-1988; (2) Key features tests (advanced conceptual design), 1989-1993; (3) Components and subsystems tests (preliminary and detailed design); and (4) Systems tests with prototype reactor module. The work performed between 1985 and 1995 was extensive. GE led seven industry companies to refine the design. Work also included a comprehensive, eight year, pre-application review process with the U.S. Nuclear Regulatory Commission (NRC). The NRC released the results of its reviews in a document called, NUREG-1368, in which NRC stated:

> *"… the staff, with the [Advisory Committee on Reactor Safeguards] in agreement, concludes that no obvious impediments to licensing the PRISM design have been identified."*

Pre-application review is an important milestone in preparation for U.S. licensing. The review process resulted in several revisions to the design to address comments and facilitate the licensing stage. This is further time that can be saved versus other designs. It also provided beneficial insights regarding the technical areas that may warrant more or less communication and analysis during licensing. It is based on this R&D that GEH has determined what efforts are necessary to support licensing. In 2007, GEH prepared a draft design control document based on past GE work as a framework of modern licensing wherein future test data and analysis results may be populated.



4.3 Reactor Deployment Timeframe

The most rapid timeframe for deployment of a commercial advanced reactor in the U.S. to the point of initial criticality is approximately 2035 timeframe under current U.S. regulatory conditions. This is based on using PRISM technology with its abundance of research and development already in place. This scenario assumes substantial parallel activities in the areas of analysis, development, testing, licensing, design, and plant procurement. By licensing the first PRISM as a “prototype” under 10 CFR 50.43(e) the associated risks may be minimized.

4.4 Licensing of Test/Demonstration Reactor and Subsequent Commercial Units

With the exception of extra safety features imposed by the NRC, the demonstration reactor will be materially identical to a commercial unit. However, the likelihood of encountering licensing issues is significantly reduced due to the existing pre-application review by the NRC in NUREG-1368 and recent work on sodium fast reactor design criteria (based on the ANSI/ANS 54.1 standard). The design and safety case for commercial units will therefore be essentially complete and confirmed after the prototype safety testing is completed.

Commercial licensing under Part 52 would require essentially the same levels of detail and safety analysis as would be developed for the demonstration reactor. The demonstration unit would continue to be valuable to further advanced reactor technology.

4.5 Development Approach

The safety case for regulatory approval to start the demonstration reactor as a prototype would leverage previous knowledge. The long-lead work required for licensing is to secure an NRC qualified safety analysis methodology embodied in analysis codes. To confidently progress the reactor towards approval of a Design Report, the underlying codes and methodologies must first be reviewed and approved by the NRC as a means of assurance that the principal design criteria for the reactor can be met.

Current guidance needs to be updated to account for advanced reactor characteristics in regulatory guides, standard review plans, codes and standards, reactor oversight process development, and inspection programs. There is a need to renew sodium-specific codes and industry standards (e.g. American National Standards Institute, Inc., American Nuclear Society [ANSI/ANS] 54.1-1989, "General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant," and ANSI/ANS 54.8-1988, (Standard for "Liquid Metal Fire Protection in LMR Plants"). Similar issues were considered during the Clinch River Breeder Reactor Plant (CRBRP) licensing process (which used 10CFR Part 50). The CRBRP application was based on case-by-case agreements between the NRC and the applicant to interpret the regulations and to satisfy regulatory intent with suitable design features and operating procedures. The regulatory structure allows for exemptions from regulations and imposition by the NRC of additional requirements, as deemed necessary, by rulemaking, order, or license conditions.

The PRISM demonstration reactor will start up on a uranium zirconium alloy fuel that possesses an extensive safety limit basis from EBR-II and FFTF operations. This will provide regulatory confidence that the initial criticality and low-power testing can proceed in a safe and controlled manner, with the remaining fuel and materials qualification to occur per the approved reactor safety test program. The key objective of the safety test program to be performed in the demonstration reactor is to confirm the safety performance envelope at the integrated system level. Because of the large margins and high reliability of the systems in the PRISM design, severe transients can be safely induced without significant risk to the public or damage to the plant. The general approach will be to perform transients that can envelope groups of events. The enveloping transient for each group will be initially performed under conservative conditions, and repeated as necessary with gradually more severe conditions. These transients will be analyzed to validate and qualify the analytical codes and procedures used in safety and licensing analyses. The qualified codes and procedures will then be used to predict reactor performance for more severe extrapolated conditions. This will then be used for a robust and low risk commercial design certification safety case.

4.6 Schedule

4.6.1 Startup

Regulatory approval to start up the demonstration reactor is expected to be developed in a phased approach. The initial phase would be based on available knowledge of materials, fuel, analyses, etc. from previous reactors such as EBR-II. The operating license is anticipated to be granted for the demonstration reactor with operational conditions imposed from the prototype test plan that require specific testing and validation processes to be followed. The reactor would be started at low power with designated hold points to permit validation and testing of key safety parameters that satisfy the prototype test plan requirements. The certification test report results should thus confirm the safety analyses that are needed to support full power operation and certification of follow-on units. Any necessary design or administrative changes would then be incorporated into the reactor fuel and component designs.

4.6.2 Reactor Licensing Challenges for Follow-On Units

The demonstration reactor is expected to address licensing challenges for follow-on units through its safety test program described in section 4.5. In addition, the entire reactor plant project would serve as a launch platform to re-mobilize the U.S. nuclear industry's advanced reactor efforts. A U.S. supported advanced reactor project has the potential to manifest lessons learned and to address the challenges of advanced reactor plant development including status of the supply chain, design methods, construction methods, procurement, and operator training, all of which would undoubtedly support commercial licensing for follow-on units and other technologies.

4.6.3 Pre-Conceptual, Conceptual, and Final Design

The terms "pre-conceptual design", "conceptual design", and "final design" are most commonly defined integrally with the process of deciding to conduct, and then executing a specific project in accordance with DOE Order 413.3, "Program and Project Management for the Acquisition of Capital Assets," as well as DOE G 413.3-12, "U.S. Department of Energy Project Definition Rating Index Guide for Traditional Nuclear and Non-Nuclear Construction Projects." Therefore, it is not practical at the present time to identify, "pre-conceptual design," "conceptual design," and "final design," for PRISM without first developing a project plan to design and build PRISM for a DOE demonstration project. The maturity of PRISM's design is best illustrated in the public domain in NUREG-1368 issued by the NRC, which was based on the extensive Preliminary Safety Information Document, which it references. This is underpinned by an extensive collection of PRISM design analyses. In order to deploy PRISM by the 2035 timeframe, significant detail design work would need to progress in parallel with regulatory licensing review in order to achieve readiness of design-for-plant construction at year 15. Licensing the demonstration reactor as a, "prototype," with both U.S. Government support and expertise from industry and national laboratories, should allow a development plan with substantial parallel work.

4.6.4 Components and System Development Schedule and Infrastructure

The work within this scope would support long lead-time items for licensing application and would prepare for larger scale work performed in a non-nuclear test facility, which would include a full size reactor vessel and select reactor components using water as the operating medium to facilitate testing activities.

4.6.5 First Core Load, Low Power Testing, Ascent to Power

Regulatory approval to start up the demonstration reactor is expected to be developed in a phased approach. The PRISM demonstration reactor would be started up on uranium zirconium alloy that has the highest technology readiness of any advanced nuclear fuel based on EBR-II and FFTF operations. This should allow the remaining fuel and materials qualification to occur during the reactor safety test program following initial criticality.

4.6.8 Commercial Scale Demonstration Reactor Supports Licensing of Future Units

As-built design data would have a tremendously beneficial effect on design certification for follow-on units especially in the case of the PRISM design because the demonstration plant design will be commercial scale and prototypical of the follow-on commercial plant design. Full scale data and as-built configuration should avoid the need for a new round of research and development for scaling and redesign, which could require more years. The PRISM demonstration reactor is designed to operate at commercial scale, thus all the key components (electromagnetic pumps, steam generators, in-vessel fuel handing machine, etc.) would not only be tested but proven. The plant can be instrumented to support pre-operational, initial startup, and continued detailed measurements and monitoring (e.g. piping vibration, thermal expansion). Thus the full-scale effects of operation on safety-related system can be confirmed. This may be used for improving safety code validation to reduce safety margin uncertainties. The critical components could continue to be measured during operation and transients for lifetime performance to inform commercial code cases.

4.6.9 Plans for Used Fuel Handling

There are various options for PRISM used fuel handling, which open a range of possibilities for advancing the U.S. nuclear industry's options for used fuel and energy security:

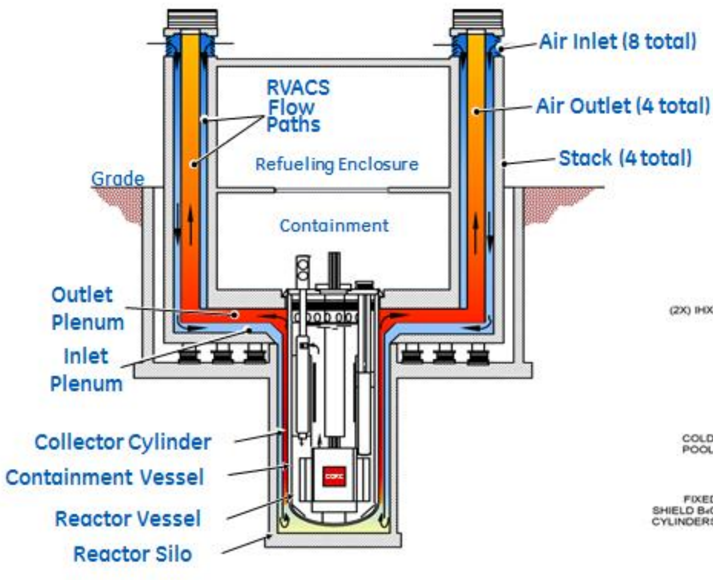
- Direct disposal of sodium-bonded PRISM fuel – Though the decision was made not to do this for used EBR-II fuel in the U.S., a strong case can be made for the technical acceptability of direct disposal.
- Processing of used PRISM fuel for disposal via immobilization as was done for EBR-II fuel (This is well understood and proven but is not evaluated further, as it is beyond the scope of this report);
- Demonstration of used fuel recycling;
- Demonstration of dry cask storage.

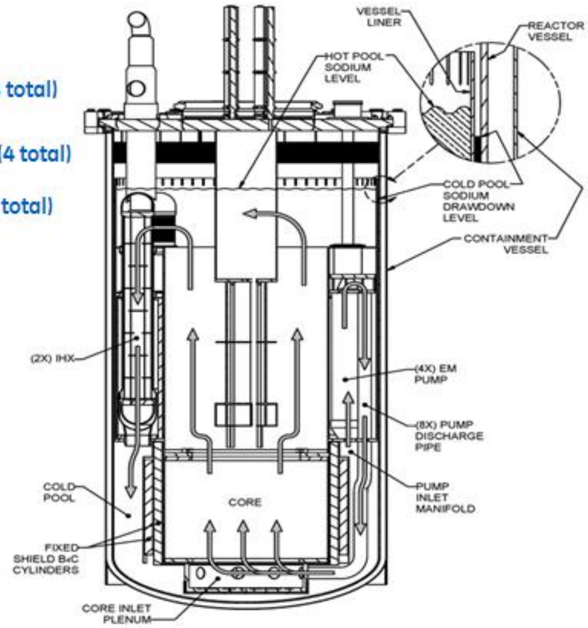
5.0 Reactor System Design

5.1 Discussion on Coolant, Fuel, Major Systems, and Key Components

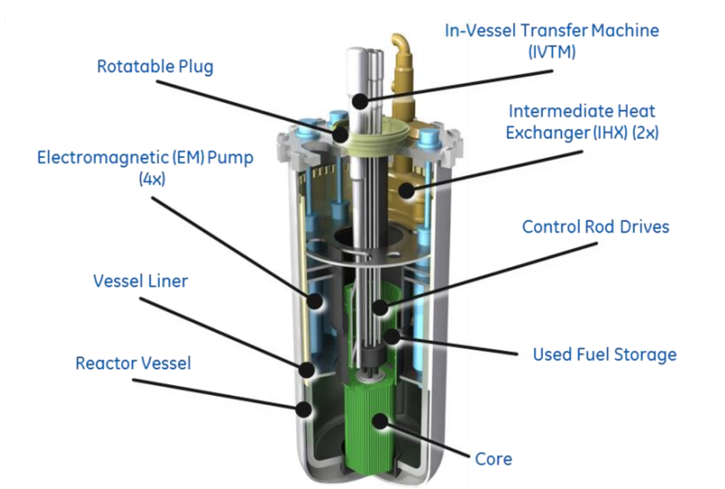
The primary and intermediate systems use sodium as a coolant at core outlet coolant temperatures up to 930 °F. A pool type configuration ensures no primary sodium leaves the reactor vessel during power operation, with no possibility of a loss of coolant accident (LOCA). The intermediate heat transfer system transports heat from the primary system to the steam generator.

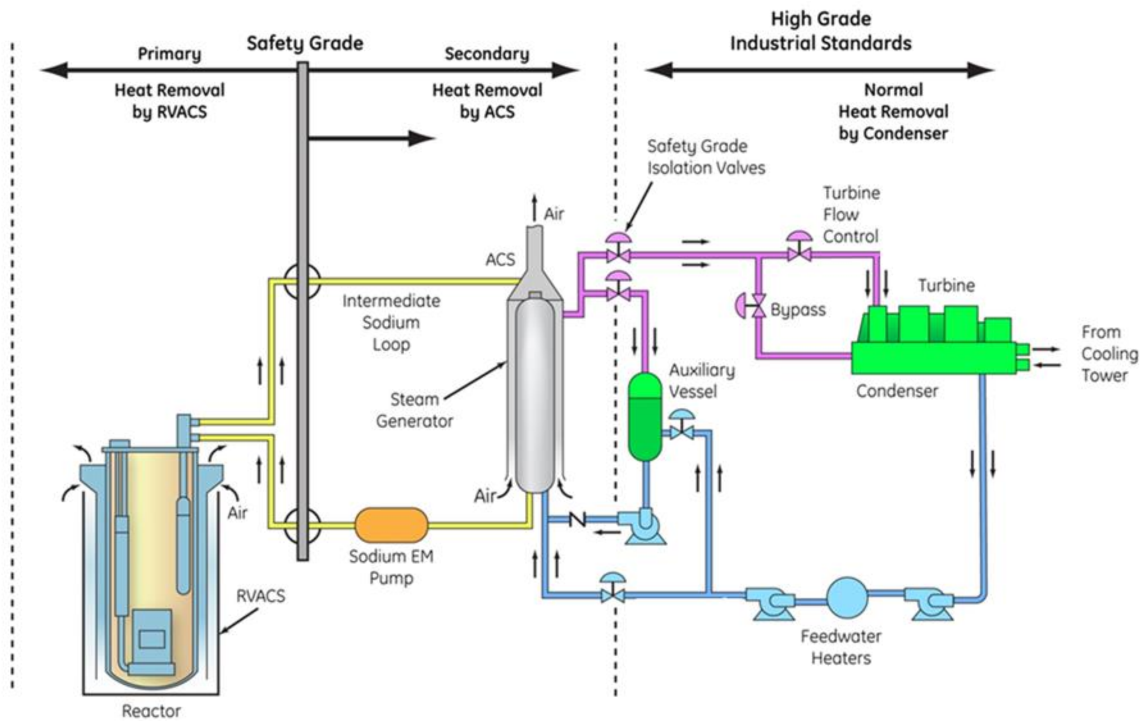
The core is a metal fuel alloy design that includes 99 fuel assemblies with two enrichment zones. The core will be a uranium-zirconium design based upon a vast operational database. The introduction of other metal cores such as: U-TRU-10Zr or U-Pu-10Zr is also possible after appropriate testing is performed to demonstrate safety margins. Refueling is planned every 12-24 months following initial reactor operation.





The reactor module system consists of the reactor vessel, internal structures, internal components, reactor core supports, reactor closure head and containment vessel (see figure above). The reactor vessel is made of two-inch thick 316 SS. Internal structures are welded inside the vessel to support internal reactor components. Key internal components include four primary sodium pumps, two intermediate heat exchangers, cover gas service lines, coolant purification lines, flux detectors and a single fuel assembly transfer port from the reactor core. The reactor core is supported by a redundant beam structure attached at the bottom and the sides of the reactor vessel. A core barrel and support cylinder, extending from the core inlet plenum to an elevation above the core, has storage racks attached to its inner surface for storage of up to 43 used fuel assemblies. The reactor closure head is a stainless steel plate that is approximately 20 feet in diameter with an off-center rotatable plug. The outermost structure is a guard vessel which surrounds the reactor vessel serving as the lower containment. The entire reactor module including steam generator is supported by a seismically isolated structure. Both the reactor vessel and containment vessel are welded to the underside of the closure head therefore there are no penetrations below the sodium operating level, which eliminates the possibility of a loss of coolant accident (LOCA).

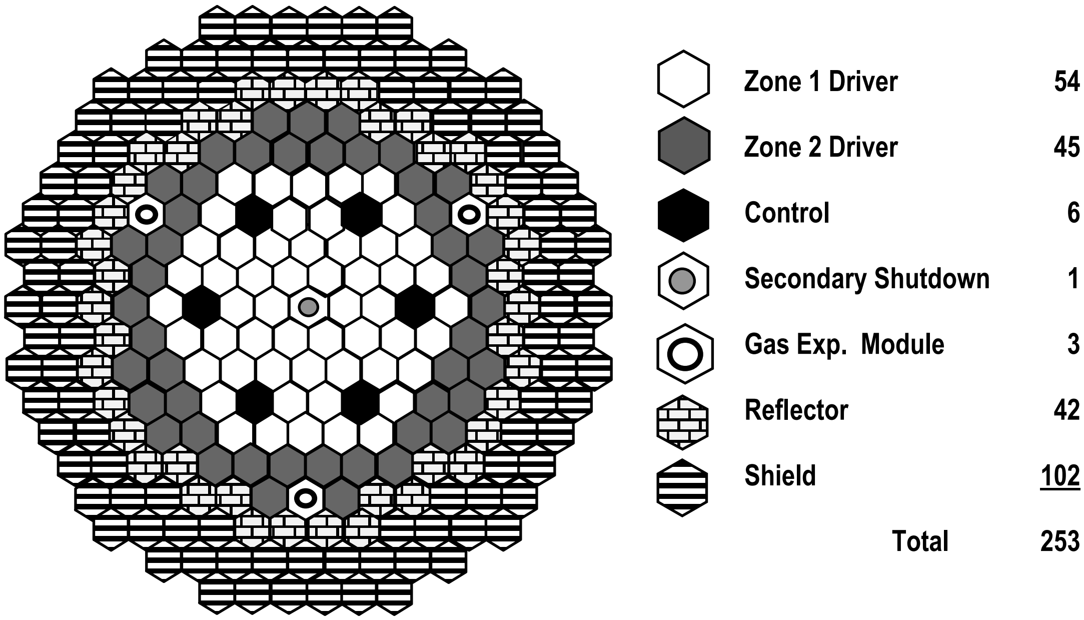



Safety Grade
Primary
Secondary
Heat Removal by RVACS
Heat Removal by ACS
High Grade Industrial Standards
Normal Heat Removal by Condenser
Safety Grade Isolation Valves
Turbine Flow Control
Air
ACS
Intermediate Sodium Loop
Bypass
Turbine
From Cooling Tower
Steam Generator
Auxiliary Vessel
Condenser
Air
Air
Sodium EM Pump
RVACS
Feedwater Heaters
Reactor

5.2 Table Summary of Key Plant Parameters

Demonstration Reactor Summary	
Reactor type	Sodium Fast Reactor, Pool Type
Reactors	1
Seismic isolation	Reactor and steam generator supported by common seismically isolated platform
Containment	Three independent volumes: upper, lower, and intermediate heat transport loop
Net electrical output	165 MWe
Net efficiency	35%
Expected capacity factor	93%
Nuclear Steam Supply System Summary	
Primary System	Sodium pool with EM pumps
Core Thermal Power	471 MWt
Core Outlet Temperature	930 °F
Core Inlet Temperature	665 °F
Total core flow	1.99E+07 lbm/hr
Pumps	4
Primary cover gas pressure (full power)	~0 psig
Intermediate Heat Exchangers (IHX)	2
Intermediate System	Two sodium loops with EM pumps
IHX sodium inlet	574 °F
IHX sodium outlet	864 °F
Total intermediate flow	1.82E+07 lbm/hr
Pumps	2
Intermediate cover gas pressure at SG	5 psig
Steam generator (SG)	Single unit, helical coil, counter flow
Steam cycle	Superheated steam
Shell side/Tube side	sodium/steam
Steam temperature	830 °F
Steam pressure	1800 psia
Feedwater temperature	380 °F
Feedwater pressure	2030 psia
Steam flow	1.61E+06 lbm/hr

Key Safety, Heat Removal and Auxiliary Systems	
Inherent Reactivity Feedbacks	Strong inherent negative reactivity feedback for core reactivity control to maintain a safe state for Anticipated Transients Without Scram (ATWS events) including Loss of Core Flow, Loss of Primary Heat Sink, and Transient Overpower without Scram.
Shutdown Systems	Two diverse and redundant boron control rod systems with passive insertion by gravity or backup by powerful drive-in motors. Gas expansion modules (GEM) used for additional defense in depth negative reactivity.
Heat Rejection System	Normal heat removal via bypass to the turbine condenser or steam blowdown to atmosphere (if necessary).
Auxiliary Water Source	Passive water supply to steam generator in the event pumps are unavailable.
Steam Generator Auxiliary Cooling System (ACS)	Backup heat removal by flow of atmospheric air by the steam generator shell (i.e. cooling via intermediate loop). Fail open damper initiates natural circulation of air but for a more rapid cooldown, forced air flow by fan is available. Flow in intermediate loop is by natural circulation or forced flow if intermediate pumps are available. SGACS increases residual heat removal capacity above that of RVACS to shorten the outages during steam turbine overhauls or in the event that water cooling is lost.
Primary Sodium Auxiliary Cooling System (PSACS)	Cooling of primary sodium via primary sodium cleanup system with nitrogen-air dump heat exchangers. To reduce cost, system shares primary sodium cleanup piping and nitrogen piping used for the primary sodium cold trap. PSACS increases residual heat removal capacity above that of RVACS to shorten outages during Intermediate Heat Transfer System (IHTS) and steam generator overhauls.
Reactor Vessel Auxiliary Cooling System (RVACS)	Backup safety grade heat removal by natural circulation of atmospheric air by the reactor vessel (i.e. lower containment vessel). RVACS is always operating under all plant modes, requires no actuations and can remove all sensible and decay heat in the event no other heat removal system is available.
Primary sodium cleanup system	Sodium cold trap to remove impurities. Corrosion is absent in sodium fast reactors leading to lack of need for coolant chemistry control.
Primary cover gas cleanup system	Vapor traps, charcoal delay beds, HEPA filters
Intermediate sodium cleanup system	Sodium cold trap. Corrosion is absent in sodium fast reactors leading to lack of need for coolant chemistry control.
Sodium water reaction protection	Chemical and acoustic detection system for operational leaks with overpressure protection for large leaks. Isolation and blowdown of water/steam from steam generator halts reaction. Shutdown cooling via ACS, PSACS and RVACS remain available after all postulated sodium water reactions.
Fuel handling in-vessel	Pantograph type
Used fuel storage positions in-vessel	Design of reactor vessel provides for storage of used fuel assemblies during reactor operation. Following a decay period of one reactor operating cycle, the power levels of the assemblies are sufficiently low to permit "dry" handling for transfer and ex-vessel storage. This simplifies the equipment for handling and storing spent assemblies.



Baseline Core Summary	
Cycle length (months)	18
Conversion ratio	0.76
Average fuel burnup (MWd/kg)	~70
Peak fuel burnup (MWd/kg)	~100
Average power density (MW/m^3)	~200
Avg. total neutron flux (10^{15} n/cm^2-s)	~2
Avg. fast neutron flux (10^{15} n/cm^2-s)	~2
Peak total neutron flux (10^{15} n/cm^2-s)	~4
Peak fast neutron flux (10^{15} n/cm^2-s)	~3
Irradiation Volumes	
Active core height (baseline core)	40 in
Irradiation volume per assembly (baseline active core region)	~0.6 ft^3
Irradiation volume per assembly (pin region)	~3 ft^3

GEH has studied many different core designs to meet customer needs. To this end we have intentionally designed the PRISM core envelope to be flexible. All PRISM cores thus far use common assembly designs but with different in-core arrangement of the assemblies to satisfy the various missions. All core assemblies use the same handling socket and have the same external dimensions but the internal configuration can be changed. The PRISM fuel heights, fuel enrichments, and pin sizes may change for some missions, the core centerline elevation remains fixed.

5.3 Plant Security, Safeguards and Proliferation Resistance

GEH has elected not to address this topic in a document without classified material protections.

5.4 Decommissioning and Waste Generation

Decommissioning of the PRISM reactor will nominally start five years after final shutdown to allow reduction in plant radioactivity. The modular construction allows modular/sequential disassembly. The most imbedded structure in the PRISM reactor is the reactor silo which is about 40 feet below ground level and has a calculated activation of $<10^{-12}$ Ci/gm at decommissioning. The silo could be removed, but with this low activity level there is the option to leave the silo in place, thus reducing waste.

Decommissioning steps include removal of all core assemblies, removal of all sodium (using a new Trade Secreted GEH process) for stabilization and low-level radioactive disposal, removal of reactor module equipment, grouting of reactor vessel, removal of structural concrete to depth of 40 feet and restoration of native site vegetation.

5.5 Integration with Energy Conversion and Industrial Process Systems

The energy conversion system for PRISM is Rankine. The PRISM reactor produces 471 MWt, energy that can be transported as heat or electricity (up to 165 MWe) to an energy-park on site. The energy park allows energy conversion test systems or skid mounted industrial process to be tested to obtain operating experience.

5.7 Identification of Non-Prototypic or Non-Scalable Aspects of the Design

The PRISM design proposed for demonstration reactor is at commercial scale therefore there are no non-prototypic or significant non-scalable aspects of this design.

5.8 Instrumentation and Control for Validation of Simulation Tools

The plant control system would be used to collect the data for the validation of simulation tools. Additional diagnostic instrumentation would be incorporated to obtain the information necessary to measure performance, validate the computer codes, perform fuel integrity studies and analyze neutron flux effects on materials. Flowmeters and thermocouples will be the primary instruments added, with the thermocouples having short time constants to permit fast transients. Strain gauges will be added as well. Additional instrumentation details are provided in the PRISM test plan. Features of key systems for data collection include:

Core Performance: Flux, temperature, flow, and vibration. The plant would be monitored during safety tests such as flux trip, flux/flow mismatch, temperature and sodium level. Reactor neutron flux is measured with two different sets of flux detectors ex-vessel and in-vessel. The ex-vessel detectors monitor the neutron flux up 150%. The in-vessel detectors measure flux during shutdown and refueling. Fuel and material testing can have in-situ flux measurements.

EM Pump performance: Inductive, ground, voltage and temperature sensors would be provided. Pump discharge plena pressure is measured for indirect pump flow rate.

Intermediate System Performance: Cold and hot legs temperatures, flow, vibration and radiation sensors would be provided. Sodium pressure is measured at the steam generator inlet so that the hydraulic performance of IHTS components can be validated for analysis codes.

Containment and RVACS Performance: Radiation, sodium aerosol detectors, sodium liquid detectors, air flow, temperature, vibration sensors are provided.

5.9 Prototypic Aspects of Fuel Performance Demonstration

The PRISM demonstration system is at commercial scale. This removes the fuel performance scaling issues and allows follow-on reactors to be licensed more rapidly. Future fuel commercial performance becomes a commercial issue rather than an unknown technology development program.

The PRISM startup core will use a uranium zirconium alloy (U-Zr) fuel leveraging the operating experience in with EBR-II and FFTF. Similar to the way light water reactor (LWR) fuel is tested, qualified, and licensed, new PRISM fuel would be tested using lead test assemblies.

A variety of non-destructive and destructive evaluation techniques are expected to be used to update the knowledge database of metal reactor fuel to underpin future commercial operations. The reactor core should be able to support other advanced reactor fuel testing to investigate phenomena relating to: fuel pin swelling; fuel pin length growth; fission gas release; and fuel/cladding compatibility.

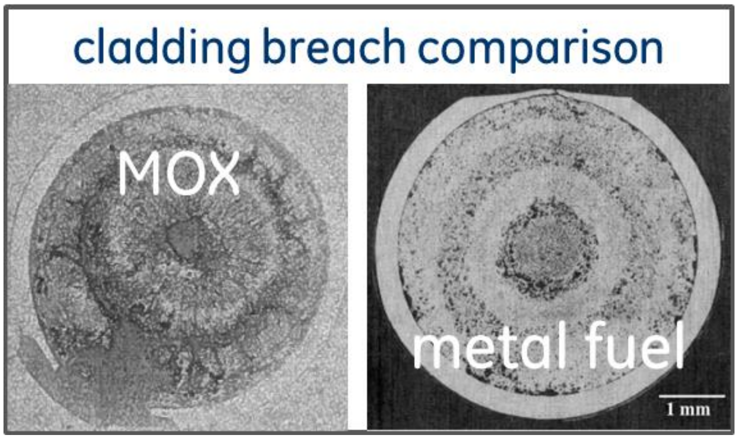
6.0 PRISM Safety Basis

6.1 PRISM Safety Characteristics

6.1.1 Inherent Safety Features of Coolant, Fuel, and Neutron Spectrum

Coolant: PRISM uses liquid sodium as its primary and intermediate coolant. Sodium has high thermal conductivity, high heat transfer at moderate velocities, low pumping power and a high boiling temperature (1,620 °F). This enables the use of a pool design that eliminates LOCA. Its low melting point (208 °F) allows refueling at relatively low temperatures compared to other liquid metal or salt coolants. Unlike the water used as coolant in LWRs, sodium coolant, acts as a getter, precluding chemical attack on steel structures, which is key to the extraordinary longevity of sodium systems. Additionally, the absence of corrosion products results in low dose rates (typically less than one-tenth of those of LWRs). Of even more importance, metal fuel does not react chemically with sodium; in fact, the metal fuel slugs are “bonded” inside the cladding with sodium. It has been demonstrated in run-beyond-cladding-breach tests in EBR-II that fuel pin cladding failure is benign and does not lead to flow blockages and failure propagation. In the event that fuel does fail, the sodium has an affinity for most non-gaseous fission products and even captures iodine and cesium which are problematic in LWR accidents. The low operating pressure of sodium systems and affinity for fission products facilitates a reduced emergency planning zone with a mechanistic source term.

Fuel: Metal fuel has high and constant thermal conductivity; in other words, heat flows easily away from the fuel. Thus metal fuel has a lower temperature gradient than oxide fuel and a lower stored Doppler reactivity. Fuel can operate with a cladding breach with little or no coolant contamination (see example of benign behavior of breached metal fuel below).



Neutron Spectrum: A highly coupled core with high leakage provides for a reliable and less complex shutdown by the use of inherent means of negative temperature/power coefficients. This allows for operator actions for safe shutdown conditions with simple systems. The fast neutron spectrum has a high tolerance for neutron-absorbing impurities.

6.1.2 Potential Coolant, Fuel or Neutron Spectrum Challenges:

Coolant: Sodium is an alkali metal and therefore has exothermic chemical reactions with carbon dioxide, oxygen, and water. Because of these well understood chemical reactions, the PRISM design has sodium fire suppression decks, cover gas piping, and built-in dedicated passive systems such as catch pans and piping jackets to minimize spray fires. Suppression of small fires is accomplished through automatic or manual fire suppression using portable conventional extinguishers.

Metal Fuel: PRISM's fuel does not retain fission product gases, has a lower operating temperature as compared to other chemical forms (e.g. nitride, carbide, or oxide). Also, it has a positive sodium void reactivity coefficient, which is highly unlikely to be challenged (multiple failures and ATWS) and is nonetheless countered by providing EM pump coastdown.

Neutron Spectrum: As with all nuclear reactors, high energy neutrons damage materials via collision cascades that produce dislocations in core structural material. These microstructural changes occurring over time degrade the material properties such as toughness leading to embrittlement and or swelling, which shortens operational life.

6.1.3 Defense in Depth Characteristics and Safety Margins

Prevention of abnormal operations and failures is the first line of defense for maintaining nuclear safety. This is achieved by monitoring the operating characteristics of the reactor to detect abnormal operating conditions.

The metallic core has inherent negative temperature and net power reactivity coefficients. An increasing core temperature inserts negative reactivity immediately and passively. As temperature rises, fuel density decreases, thereby reducing the fission reactions. The reactor simply shuts itself down. This allows the reactor to achieve and maintain a controlled and stable condition without control rod insertion. In the event that all control rods fail to insert, the diverse ultimate shutdown system inserts its control rod to force reactor shutdown.

The next line of defense is the capability to control abnormal operation and failures, if prevention measures are not sufficient. Mitigation measures are designed to restore the reactor to safe and stable conditions, thus preventing damage to fuel, cladding or reactor internals. A significant challenge to a nuclear power plant would be the inability to remove decay heat after being shut down. PRISM's conductive metal fuel and metal coolant is a key enhancement from light water reactors. The high thermal conductivity at each point in the path from fuel to cladding to coolant to vessel dissipates excess heat without damaging the fuel, cladding, or any reactor components. This inherent process uses the laws of nature and known properties of materials instead of relying on human actions and electronic control or mechanical intervention to mitigate a major event.

The PRISM shutdown heat removal system consists of three elements which provide three separate heat removal paths:

1. Main condenser cooling,
2. An auxiliary cooling system (ACS) using the steam generator surface, and
3. The safety-related reactor vessel auxiliary cooling system (RVACS) which removes heat using the surface of containment vessel.

Additionally, a primary sodium auxiliary cooling system (PSACS) operates to remove heat during steam generator and IHTS maintenance operations. Shutdown heat removal system thermal performance analyses show that maximum reactor temperatures are maintained at levels well below design limits.

PRISM design features also mitigate radiological consequences associated with a hypothetical material release. Examples of how the PRISM design mitigates those radiological consequences include the following:

- The PRISM reactor has the ability to safely retain within the reactor vessel, and away from its walls, the molten fuel of the entire core and all of the irradiated fuel assemblies stored in-vessel.
- The reactor design includes completely passive heat removal from the lower containment and relies only on natural circulation of atmospheric air to remove heat from the surface following a postulated accident.
- The absence of concrete and sodium storage tanks inside the containment to prevent traditional containment pressurization accidents resulting from sodium fires and sodium-concrete interactions.
- Significant containment design margin that prevents early containment failure to enhance retention of radioactive gases and the fallout and plate-out of radioactive aerosols within the containment.

6.1.4 Barriers to Release of Radioactive Materials from PRISM Fuel

Barriers to prevent radioactive release include; 1) the soft fuel matrix and gas plenum within each strong fuel pin; 2) the hard steel cladding encasing each fuel pin; 3) high fuel-coolant compatibility inhibits release after fuel pin breach; 4) fission product retention by sodium and deposition after release from fuel pin 5) the reactor vessel; and 6) the upper and lower containments.

6.1.5 Inherent Radionuclide Retention and Source Term

PRISM will have a source term calculated differently than how the source term is calculated for LWRs. There are many features of sodium fast reactor (SFR) accident behavior that differ from LWR severe accident scenarios. Typically in an LWR severe accident scenario, the core becomes uncovered associated with a loss of primary coolant water. Subsequent to fuel failure, only a single barrier exists against release of fission products to the environment and overheated conditions exist for an extended period of time. In contrast, in a typical SFR severe fuel damage scenario, the reactor coolant system is intact at the time of fuel damage. Thus, two barriers exist. In addition, fission products released from the fuel must pass through an overlying pool of sodium. Fission product transport through sodium has been the subject of extensive research in areas such as:

- Chemical affinity, such as between sodium (and other alkali metals), iodine (and other halogens) retains these types of fission products in the sodium;
- Solubility of the fission products such as alkali metals and lanthanide metals also retains fission products in the sodium;
- Buoyancy caused by differences in specific gravities, such as of the fission gas in sodium;
- Volatility of the constituents in the liquid not in thermodynamic equilibrium at given temperature and pressure will cause vaporization of halogens;
- Liftoff by drag forces, such as of aerosol size particles in a sodium vapor bubble; and
- Gravitational settling such as fallout of large aerosol particles in the cover gas space and sedimentation of fuel particulates.

PRISM has superior inherent radionuclide retention compared to currently operating commercial light water cooled reactors. Metallic fuel also melts at a substantially lower temperature than oxide fuel, which is beneficial by suppressing the release of radionuclides with high volatility. Thus, the source terms calculated for the SFR are substantially smaller than those typically used in LWR severe accident analyses.

6.1.6 Shutdown Heat Removal and Accident Mitigation

Reactor shutdown heat is normally removed by the turbine condenser on bypass. PRISM has two auxiliary cooling systems if the turbine condenser is not available: the Auxiliary Cooling System (ACS), which uses natural or forced circulation of atmospheric air past the shell side of the steam generator, and the Primary Sodium auxiliary Cooling System (PSACS). The PSACS is connected to the primary sodium cleanup lines and is actuated by valve operation, were primary heat is rejected to the atmosphere via a heat exchanger.

Whether or not these three systems are functioning, RVACS will passively remove the decay heat. RVACS is the shutdown heat removal system designated as safety-related requiring no power, no valve or damper manipulation, and no time limit for operation.

6.1.7 Emergency Decay Heat Removal

Completely passive emergency decay heat removal is achieved by the continuously operating and monitored RVACS that relies only on the natural circulation of the primary sodium coolant and atmospheric air to remove the decay heat. RVACS has the capacity to meet all shutdown heat removal needs for the reactor module. RVACS performance can be readily monitored and maintained. These features and characteristics include passive shutdown heat removal for loss-of-cooling events and passive reactivity transition to a safe stable state for undercooling and overpower events even with failure to scram.

6.1.8 Size of Emergency Planning Zone

Radiation release probabilities and characteristics are such that detailed off-site evacuation planning, exercises, and early warning will likely not be required for the PRISM design. The source term for PRISM is based on fission product to sodium coolant solubility/affinity, lack of high system pressure, and a robust passively cooled containment. Thus the timing, magnitude, and chemical form of any release is different than LWRs. GEH has performed a containment response and release analysis for justification of an EPZ at 800m. The results show that a smaller EPZ (e.g. <400m) can be achieved.

6.1.9 Maximum Hypothetical Accident

Fast reactor fuel has higher fissile content and thus could possibly re-power if a damaged core re-compacted into a critical mass. This is known as the, "maximum hypothetical accident." PRISM's metal fuel mitigates the energy release from this type of accident. In fact, the EBR-I partial core meltdown demonstrated that during core meltdown metal-fueled cores are naturally dispersive, reducing, decay energy release and avoiding re-criticality. In an 'over-powered' metallic core, fuel dispersal is outward, reducing reactivity and power and preventing re-criticality. A hypothetical core disruptive accident is therefore not physically realistic for PRISM.

6.2 PRISM Demonstration Reactor Safety Performance

6.2.1 Probabilistic Risk Assessment

GEH, in collaboration with ANL, is currently working on a DOE sponsored project to upgrade the PRISM PRA. The objectives of the project are to:

- Develop PRA methods for non-LWRs and derive safety insights from the PRISM design;
- Demonstrate the acceptability of the overall risk of the facility; and
- Ensure a balanced design such that no particular class of accident or feature of the facility makes a disproportionate contribution to the risk target of concern.

As with all potential hazards, risk analysis during the design stage can find potential weak points and help refine the design either eliminating the concern or reducing its importance to risk.

7.0 Economics and Schedule

This section provides a qualitative discussion of the key factors affecting plant development and operational costs.

7.1 Plant Development, Construction, Operating, and Waste Management Costs

Some prototype plans proposed in the past focused only on the nuclear island and excluded the power production or process heat aspects of the commercial design. Producing power and earning revenue help to defray project costs. Secondly, prospects for private sector investment need assurance that the demonstration power plant can be licensed by the NRC for commercial use. The relatively small and modular PRISM reactor design lends itself to an affordable and prototypical demonstration reactor compared with those designs whose costs are more driven by economies of scale, such as large monolithic loop designs.

Producing a well understood cost estimate for the specific program, site, and mission is not a trivial matter – it can be costly and requires considerable effort. But the time and cost is warranted if one is seriously considering the levels of investment of time and money involved in new nuclear plant construction. The cost estimate should mature with the project resulting in a narrowing band of cost uncertainty as the project opportunity develops. GEH noted that Appendix B of the U.S. Department of Energy's 2014, "Report of the Plutonium Disposition Working Group: Analysis of Surplus Weapon-Grade Plutonium Disposition Options," specifies a capital cost of $4B for a single-unit power block reactor similar to the PRISM design. Cost of the proposed demonstration plant is below $4B due to its smaller rating (471 MWt vs. 840 MWt). Since the work producing this report did not include cost estimating, a more specific cost number is not available.

The PRISM design has plant operational features that minimize radioactive waste. The primary source of radioactive, gaseous wastes in PRISM is the reactor cover gas. PRISM uses electromagnetic coolant pumps, which obviate the need for a rotating shaft penetration and associated seal through the reactor closure head. This allows the reactor to operate as a sealed system that is only opened for refueling. Thus, there is no leakage path of reactor cover gas during operation. The cover gas is processed during outages, recycled by a processing substation and then replaced.

PRISM has plant operational features that reduce maintenance and staffing requirements. Projected PRISM staffing levels indicate a substantial reduction in personnel required compared to current LWR nuclear experience. This is primarily the result of the reduced safety role of the operator, the highly automated plant control and management information systems anticipated to be used, the plant's passive safety features, which result in fewer active safety systems, and a small footprint.

7.2 Cost of Demonstration Reactor and follow-on units

PRISM is relatively small, which makes it economically feasible to build the actual commercial design as a test/demonstration reactor minimizing the regulatory, cost, and schedule risks of scaling up a small scale demonstration. The main difference in follow-on commercial units would be a larger yard to accommodate multiple modules for greater economies of scale. Scaling of the yard is not a significant cost or risk concern.

7.3 Plant Capacity Factor, Efficiency and Possible Revenue Generation Options

An availability assessment considered failure and repair rates of design features and their associated uncertainties, and then optimized the schedule of plant operations (e.g. refueling, steam generator inspections, and turbine preventive maintenance), to maximize energy production. The result was a calculated capacity factor above 90%, which exceeded the design goal of 85%. Lower initial capacity factors are assumed for the early years of operation.

Design features contributing to high reliability and availability are:

- Use of electromagnetic pumps with no moving parts;
- Simple, passive, safety systems such as RVACs;
- Redundancy of control system logic; and
- Each reactor has six control rods, controlled by a triply redundant reactivity controller reducing the risk of an inadvertent scram.

7.4 Anticipated Demonstration Reactor Start-Up with Immediate Approval

GEH believes that the most rapid timeframe for deployment of a commercial advanced reactor in the U.S. to the point of initial criticality is the 2035 time frame under current U.S. regulatory conditions. This is based on using PRISM technology with its abundance of research and development already in place. This scenario assumes substantial parallel activities in the areas of analysis, development, testing, licensing, design, and plant procurement. By licensing the first PRISM as a "prototype" under 10 CFR 50.43(e) it may be possible compensate for the risks associated with this parallel work.

7.5 Uncertainty of Economic Estimates

Certainty of cost estimates in megaprojects such as new nuclear plants is of considerable interest for obvious reasons and has been a challenge for the industry in many situations.

With GEH's ABWR design in Japan, a significant portion of nuclear project costs for existing plants was spent on planning before anything was constructed, which has yielded some of the best new nuclear plant project performance records in the industry. This reflects the importance of investing in up-front planning and evaluation to grow a new nuclear plant option.

Since 1955, GE and Hitachi have collectively been designing, constructing, servicing, and fueling nuclear power plants in North America, Asia, and Europe on a continuous basis. This includes four Advanced BWR units recently under construction by the GE Hitachi Nuclear Alliance (the Lungmen 1 and 2, Shimane-3, and Ohma-1 units), as well as four Advanced BWR units that have operational experience (Kashiwazaki Kariwa Units 6 & 7, Hamaoka-5, and Shika-2 units).

GEH recommends that caution be exercised in directly comparing nominal capital cost estimates from different sources because the quantification of uncertainty and confidence levels can vary considerably. A plant with higher nominal cost and lower uncertainty may yield lower as-built cost than a plant with a lower nominal cost claim but higher uncertainty.

Short of performing detailed stochastic calculations, for an understanding of uncertainty level, consider the level of detailed design and prior development.

8.0 Summary of Self-Assessment in separate document

GEH performed a self-assessment of PRISM using the Goals, Criteria, and Metrics from the U.S. Department of Energy Nuclear Energy Advanced Reactor program. The GEH self-assessment score is: **89%.**